MASTER

LIQUID METAL FAST BREEDER REACTORS

1948-1961

Prepared for

Division of Reactor Development and Technology

U. S. Atomic Energy Commission
Technical Information Center

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Experimental Breeder Reactor II
at Argonne National Laboratory's
Idaho Division

Liquid Metal Fast Breeder Reactors

1948—1961

Prepared for the
Division of Reactor Development and Technology

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INTRODUCTION

This bibliography on sodium and sodium—potassium cooled fast breeder reactors has been published in two parts. Part I covers the literature contained in *Nuclear Science Abstracts* from 1948 through 1961, Volumes 1–15. Part II covers the period 1962–1971, *NSA* Volumes 16–25.

Abstracts are divided by year/*NSA* volume number and are arranged by original abstract number. The subject indexes refer to these abstract numbers.

All the literature cited is publicly available. The scientific and technical journals in which much of the literature appears are available in many libraries throughout the United States. The report literature is on file in microfiche form at the libraries listed on the inside front cover of this publication.

The reports listed herein (*NSA* Vols. 1–15) can be purchased from Microsurance, Inc., P.O. Box 3522, Oak Ridge, Tennessee, 37830. A price list appears at the end of this compilation.

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Information is indexed to names of specific reactors and general headings such as critical assemblies, fast reactors, and liquid metal cooled reactors.

Major reactors covered in this bibliography are shown in the chart following this introduction.

MAJOR REACTORS

Reactor name	Location	Type	Output	Fuel	Coolant	Startup/ shutdown dates
USA						
EBR-1 (Experi- mental Breeder Reactor-1)	NRTS, Idaho	First generation fast breeder	1.4 MW(t) 150 kW(e)	Plutonium	Sodium— potassium	1951 startup 1964 shutdown
EBR-2 (Experi- mental Breeder Reactor-2)	NRTS, Idaho	First generation fast breeder	62.5 MW(t) 16.5 MW(e)	PuO ₂ /UO ₂	Sodium	Nov. 1963 startup
EFFBR (Enrico Fermi Fast Breeder Reactor)	Lagoona Beach, Mich.	First generation fast breeder	200 MW(t) 66 MW(e)	PuO ₂ /UO ₂	Sodium	1963 startup
USSR						
BR-1	Obnisk	First generation fast breeder	0 MW(t) 0 MW(e)	Plutonium metal		1955 startup 1956 shutdown
BR-5	Obnisk	First generation fast breeder (based on BR-1)	5 MW(t) 0 MW(e)	PuO ₂	Sodium	June 1958 startup
UNITED KINGDOM						
DFR Dounreay Fast Reactor	Dounreay	First generation fast breeder	72 MW(t) 14 MW(e)	Uranium metal (45.5% enriched)	Sodium— potassium	Nov. 1959 startup

Volume 2, Jan.-June, 1949

- 21 The Importance of Breeding in Nuclear Power Development by Andrew W Kramer Power Generation 52 No.10 86-89 (1948) Oct

The author stresses the importance of a continuous supply of fissionable materials in the future development of nuclear energy. The physics and the economic aspects of "breeding" as a means of augmenting these materials are discussed.

Volume 4, 1950

- 4359 Production of Enriched Nuclear Fuel K H Kingdon Gen Elec Rev 53, 11-12(1950) May

This is a brief review of the advances that have been made in producing enriched nuclear fuel, directly and by the breeding process. The economic significance of its utilization in large-scale industrial application is pointed out.

Volume 5, 1951

5362

National Bureau of Standards

GAMMA RAY SPECTRA OF LOS ALAMOS REACTORS, "WATER BOILER" AND CLEMENTINE J W Metz [nd] Decl July 5, 1951 29p (AECD 3188, LADC-961)

The γ -ray beams emerging through the collimator hole of the Los Alamos Water Boiler (which contained a U^{235} slug in the glory hole) and of the Los Alamos Fast Reactor, were examined with a magnetic spectrometer. The γ rays ejected Compton electrons from a Be radiator in the spectrometer. Only those electrons confined to a narrow cone in the forward direction of the γ -ray beam were detected, so that monoenergetic γ rays yield a "line" for the Compton electron momentum measurements rather than the usual continuous distribution. With each reactor operating at full power, the electron momentum spectrum was measured. From the Klein-Nishina formula and the spectrometer calibration, the observed measurements were converted to a corresponding γ -ray spectrum which expressed the number of photons per unit energy interval incident on the radiator per cm^2/sec as a function of the incident photon energy. An independent check of this conversion to absolute photon intensities was made by measuring the intensities of the γ -ray lines from Cs^{137} (0.661 Mev) and Na^{24} (1.38 Mev, 2.76 Mev) with an extrapolation ionization chamber, and comparing the

calculated and measured values of the number of Compton electrons/sec detected by the spectrometer. The γ -ray spectrum from each reactor covered a range from 0.5 to 12 Mev. At 8 Mev, the photon intensity for each spectrum was reduced by a factor of 10^4 compared to that at 0.5 Mev. The shapes of both spectra appear to be a straight line on a semilog plot, with approximately the same slope for each line. (auth)

Volume 6, 1952

998

National Bureau of Standards

GAMMA-RAY SPECTRA OF THE LOS ALAMOS REACTORS, by J. W. Motz [nd] Dec 26, 1951 9p (AECD-3286, LADC-1060)

The γ -ray beams emerging from the collimator hole of the Los Alamos Water Boiler (operating at 30 kw) and of the Fast Reactor (operating at 25 kw) were allowed to impinge on a Be foil, and the momentum distribution of the Compton electrons ejected from the foil into a small solid angle in the forward direction of the beam was measured with a magnetic spectrometer. For the Water Boiler, the source of the collimated γ -ray was a U^{235} cylindrical slug (diameter 2.22 cm, length 0.426 cm, weight 28.5 g, 95% purity) placed at the center of the sphere, while for the Fast Reactor, the γ -rays originated in the reactor core. The spectrometer measurements covered a range of electron energies from ~ 0.5 to 12 Mev. "Hot wire" measurements indicated that the focusing properties of the spectrometer did not change over this energy range. The measurements for the Water Boiler and Fast Reactor were made with a Be foil 11 mils thick and 33 mils thick, respectively. (auth)

1318

Atomic Energy Commission

THE ATOMIC ENERGY COMMISSION REACTOR PROGRAM, p.34-57 of THE ROLE OF ENGINEERING IN NUCLEAR ENERGY DEVELOPMENT, THIRD ANNUAL OAK RIDGE SUMMER SYMPOSIUM, AUGUST 27 TO SEPTEMBER 7, 1951; by Lawrence R. Hafstad. Dec. 1951. (TID-5031(p.34-57))

A general discussion is given of the reasons for developing the particular reactors in the AEC program. The possibility of developing a breeder reactor is considered. The amount of energy available in the world is compared to the estimated world needs. It is thought that nuclear power is just an interim expedient until that day when the world will be using continuous solar energy. The necessity for security in the AEC Research Program is discussed.

5711

Knolls Atomic Power Lab.

A STUDY OF ELECTRIC POWER GENERATION UTILIZING HEAT ENERGY FROM POWER BREEDER REACTORS, by R. H. Shannon, J. D. Selby, and M. B. Dagan. Nov. 19, 1951. Dec. Aug. 27, 1952. 45p. (AECD-3444; KAPL-591)

The reactor heat capacities, steam conditions, installed electric-power-generating capacity, estimated plant investment capital, and estimated energy cost are tabulated for six types of reactors (500, 1000, and 2000 Mw using either liquid Na or high-pressure water). The following conclusions and recommendations are made. The cost of electric power generation utilizing heat energy from power breeder reactors is economically comparable to the cost of generation of electric power by oil-fired-steam-power plants, provided the capital investment and operating costs of the reactor and basic reactor cooling systems are not

charged to the cost of electric power generation. Electric power generation utilizing heat energy from liquid-Na-cooled reactors is substantially more attractive economically than utilizing heat energy from high-pressure water-cooled reactors. In order to be able to deliver reliable electric power it is recommended that the plant heat sources be divided among two or more reactors, (or that stand-by conventional steam boilers be provided to allow for reactor refueling and outage time). It is recommended that the heat energy available from one reactor in a plant not exceed 500 Mw. *Based on present power consumption the operating electric power systems, with some exceptions, could not easily utilize the power generated by a plant having reactors rated in excess of 500 Mw.*

5712

BASIC PROBLEMS IN CENTRAL-STATION NUCLEAR POWER. W. H. Zinn. Nucleonics 10, No. 9, 8-14(1952) Sept.

Economic considerations in the production of electric power from nuclear reactors are discussed. Some operating characteristics of the Experimental Breeder Reactor at Arco, Idaho, are mentioned.

Volume 7, 1953

2137

DUAL-PURPOSE REACTORS; FIRST STEP IN INDUSTRIAL NUCLEAR POWER DEVELOPMENT. Carroll A. Hochwalt and Philip N. Powers. Nucleonics 11, No. 2, 10-13(1953) Feb.

Design and cost factors of a dual-purpose reactor for the production of Pu and useful power are described. Proposals for the initiation of a program for the industrial development of power breeder reactors are outlined. (L.T.W.)

5854

Oak Ridge National Lab
THE ECONOMICS OF POWER REACTOR PROCESSING.
D E Ferguson [July 1953] 14p (AECD-3550, CF-53-7-46)

The economics of chemical processing for a breeder-type reactor intended primarily for power production is discussed. The purpose of chemical processing is to enable the reactor to produce power as economically as possible. In general, the chemical process must remove the fission products which lower the efficiency of reactor operation and reclaim unspent fuel and source material. The costs that vary with the frequency of processing may be optimized, and the cost of processing per kilowatt of electricity produced may be expressed in terms of several variables. These variables include the value of fissionable material as fuel, the unit cost of chemical processing, and the fraction of material lost during processing. An evaluation of the effect of these variables on the cost of atomic power is presented. (auth)

Volume 8, 1954

6304

ECONOMIC POWER FROM FAST BREEDER REACTORS?
C A. Rennie (Atomic Energy Research Establishment, Harwell, Berks, England). NUCLEAR ENGINEERING, PART II, Chem Eng. Progr. Symposium Ser No 12, 222-27(1954).

The role of fast breeder reactors in a nuclear power program is discussed. It is concluded that a balanced scheme of fast reactors which produce more fissile material than they consume, and thermal reactors which are not quite self-sustained in fissile material offer an attractive line of development for a country without large reserves of uranium, since in this way it should be possible to utilize an appreciable fraction of the source material for power generation. The influence of reactor costs and processing costs on the price of electrical power is discussed, and some estimate is made of the allowable costs in such a system. A value can be assigned to the fissile material once the other costs are known, and this would enable comparisons to be made with other schemes for generating electrical power from nuclear fuels (auth)

6306

APPROXIMATIONS ON THE KINETIC BEHAVIOR OF FAST REACTORS. Richard A. Fayram and Karl Bernstein (Univ. of California, Berkeley and Los Angeles). NUCLEAR ENGINEERING, PART II, Chem. Eng. Progr. Symposium Ser. No. 12, 244-59(1954).

Kinetic equations for fast reactor behavior, delayed neutron yields for Pu^{239} and U^{235} , neutron cross sections, motion of fuel and reflector elements for control, temperature effects on reactivity, and time-dependent reactivity changes are discussed and evaluated. Generalized systems are treated in an elementary way, with attention given to the effects of physical characteristics, design variables, and characteristics which are to some extent both operationally controllable and inherent in a given system. Some calculated results based on representative data are given. The discussion suggests means of making preliminary, rapid, and approximate evaluation of the kinetic behavior of fast reactor systems. (L.T.W.)

6551

FAILURE AND DISASSEMBLY OF LOS ALAMOS FAST REACTOR E T Journey (Los Alamos Scientific Lab, N Mex) Nucleonics 12, No 9, 28-30(1954) Sept

Operations engaged in during the disassembly of the Los Alamos Fast Reactor following fuel element failure are described and illustrated photographically. Disassembly methods described led to a minimum of danger from radiation exposure and from plutonium poisoning (C H)

Volume 9, 1955

753

OPERATIONAL FEATURES OF ZEPHYR. L. R. Shepherd, R. D. Smith, J. E. R. Holmes, H. Rose, and D. D. McVicar (Atomic Energy Research Establishment, Harwell, Berks, England). NUCLEAR ENGINEERING, PART III, Chem. Eng. Progr. Symposium Ser. No. 13, 1-4(1954).

Zephyr, the low-power or zero-energy fast reactor now operating at Harwell has been built with the object of obtaining basic nuclear data needed for fast reactor systems. The reactor has been designed to give as much flexibility as possible from the experimental point of view; it consists of a Pu core surrounded by a U envelope or blanket. The normal operating power is only a few watts; therefore no forced cooling or elaborate shielding is necessary. This makes the problem of access to the reactor much easier. Some results on the relative yields and periods of delayed neutrons obtained with this reactor are given. (auth)

757

THE EXPERIMENTAL BREEDER REACTOR. H. V. Lichtenberger (Argonne National Lab., Lemont, Ill.). NUCLEAR ENGINEERING, PART III, Chem. Eng. Progr. Symposium Ser. No. 13, 139-46(1954).

The basis for the design of the Experimental Breeder Reactor is discussed along with the details of construction and the constants of the reactor. The cooling system and the plant layout are described. The concepts of breeding gain and specific power are discussed along with the significance of the development. (auth)

1083

OPERATIONAL FEATURES OF ZEPHYR. J. E. R. Holmes, D. D. McVicar, H. Rose, L. R. Shepherd, R. D. Smith, and A. M. Smith (Atomic Energy Research Establishment, Harwell, Berks, England). J. Nuclear Energy **1**, 47-52(1954) Aug.

Zephyr is a small low-power fast reactor with a core made up of plutonium and natural uranium sections surrounded by a uranium envelope. The reactor is controlled by moving sections of uranium near the core to vary the fraction of neutrons that may escape from the latter. Power level measurement is carried out by pulse-operated fission chambers with both linear and logarithmic ratemeters as indicators. Initial studies have included the investigation of the delayed neutrons and their influence on the reactor kinetics. (auth)

1701

NUCLEAR POWER PRODUCTION AND UTILISATION. IV. THE LONG-TERM PROGRAMME FOR ATOMIC POWER PRODUCTION. E. W. Titterton (Australian National Univ., Canberra). Atomics **6**, 10-13(1955) Jan.

Technical problems associated with a long-term program for atomic power production, based on the economics of a breeder reactor are discussed. (K.S.)

2445 AERE-RP/M-46

Atomic Energy Research Establishment, Harwell, Berks (England)

THE VARIATION OF NEUTRON FLUX WITH NUMBER OF FUEL ELEMENTS IN THE APPROACH TO CRITICALITY OF ZEPHYR. J. J. Syrett. Aug. 16, 1954. 8p.

An expression is derived by diffusion theory for the neutron flux in a subcritical reactor consisting of a reflected cylindrical core containing a neutron source. By applying this expression to a model of Zephyr, it is shown numerically that for a fixed source at the center of the core, the square root of the number of fuel elements in the core divided by the neutron flux at a fixed point in the reflector, plotted as a function of the number of fuel elements, is approximately a straight line when more than half the critical number of fuel elements are present. The results are compared with those obtained during the approach to criticality of Zephyr. (auth)

2776

TESTING HEAT TRANSFER AND CORROSION OF GLOBE-IRON FOR FUEL CLADDING IN A Na-COOLED FAST BREEDER. R. H. Jones (Hartford Electric Light Co., Conn.) and R. E. Lee (Mine Safety Appliances Co, Callery, Penna.). Nucleonics **13**, No. 2, 70-2(1955) Feb.

Heat transfer and corrosion tests show that Globeiron, an iron alloy, is satisfactory as a thin-walled fuel-cladding material for 30-day exposures in a Na-cooled fast reactor. The tests simulated reactor operation. The testing appara-

tus is described, and data and results are presented. Tests indicated that Globeiron maintained its original over-all heat-transfer coefficient throughout a 30-day period and that there was no fouling of the heat exchanger due to mass transfer in the Na loop. Metallurgical examination showed no significant attack on the Globeiron specimen. (M P G)

3294

NUCLEAR POWER REACTORS. II. Stuart McLain (Argonne National Lab., Lemont, Ill.). Elec. Eng. **74**, 144-8(1955) Feb.

Discussions developed in Part I are extended to the ORNL Homogeneous Reactor Test and the Argonne Experimental Breeder Reactor -II. (K.S.)

Volume 10, 1956

1029 ANL-5491

Argonne National Lab., Lemont, Ill. and Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.

JOINT STATUS REPORT ON FAST REACTOR PHYSICS. D. Okrent and L. S. Shepherd. Oct. 1955. 3p. Contract W-31-109-eng-38.

The status of fast reactor physics in the United Kingdom and U. S. has been examined. Substantial agreement was found in experimental and theoretical values of breeding ratio and other figures. (M.P.G.)

1560 NDA-14-58

Nuclear Development Associates, Inc., White Plains, N. Y. MATHEMATICAL ANALYSIS OF DIFFERENTIAL EQUATIONS ARISING IN THE FAST REACTOR SAFETY STUDY. H. V. Waldinger. Nov. 12, 1954. 25p.

Methods are presented for carrying out the numerical analysis of equations which characterize the dynamical behavior of an idealized reactor. These methods have been applied in specific problems of the APDA fast reactor safety study. (auth)

2507 A-4257

[General Electric Co., Schenectady, N. Y.]. METHODS OF CALCULATION SUITABLE FOR DILUTE FAST AND FAST INTERMEDIATE PILES. PART I. GENERAL THEORY FOR NO SLOWING DOWN IN REFLECTOR AND ONE-VELOCITY APPROXIMATION. Harvey Brooks. Nov. 6, 1946. Decl. Dec. 8, 1955. 21p. (GE-HB-7). \$4.80(ph OTS); \$2.70(mf OTS).

3226 ANL-5513

Argonne National Lab., Lemont, Ill. THE FAST CRITICAL ASSEMBLY. B. C. Cerutti, David Okrent, R. E. Rice, F. W. Thalgott, and H. V. Lichtenberger. Jan. 1956. 29p. Contract W-31-109-eng-38. \$0.25(OTS).

The fast zero power reactor experiment (ZPR-III) is planned to obtain neutron physics information necessary for the design of fast power breeder reactors. The systems to be studied are blanketed cores having a range of compositions in which the volume fractions of U^{235} , U^{238} , Fe, and Na (simulated by reduced density Al) are varied. This is accomplished by assembly of small pieces in drawers which are placed in an array of square tubes mounted on two carriages so that two half-critical sections initially separated may be brought together in a carefully controlled manner. The reaction is controlled by inserting or removing drawers from the back of each half of the reactor with drives which are continually activated for scram under air pressure. Five of these safety-control drawers are pro-

vided in each half and may operate either in the core or blanket. The build up of the first critical assembly was begun October 18, 1955, and the assembly went critical two days later, October 20. (auth)

3384 ANL-5379

Argonne National Lab., Lemont, Ill. THE FAST EXPONENTIAL EXPERIMENT. F. H. Martens. Nov. 1955. 116p. Contract W-31-109-eng-38. \$0.60(OTS).

The Fast Exponential Experiment was set up in order to study the properties of a reactive system utilizing fast neutrons. The experiment consisted of a source pile of U^{235} moderated with H_2O to provide neutrons to the exponential assembly, and the assembly itself. A U slab was located between these two systems to absorb thermal and resonance neutrons from the source pile. Complete descriptions of these systems are included. Data are given from preliminary studies using this system. These include flux distribution measurements, reflector studies, and shielding studies. Descriptions are also given of fission chambers, flux monitors, and catcher foils to be used with the assembly. The theory of buckling is briefly discussed, and results of buckling calculations for an enriched exponential core are given. Exponential measurements of buckling are also described. Measurements of reflector savings and fission ratios are also described. Data on equilibrium neutrons in natural U are included in appendices. (B J.H.)

3650 ANL-4076

Argonne National Lab., Lemont, Ill. EXPERIMENTAL NUCLEAR PHYSICS DIVISION AND THEORETICAL NUCLEAR PHYSICS DIVISION REPORT FOR JULY, AUGUST AND SEPTEMBER. Oct. 13, 1947. Decl. Dec. 15, 1955. 105p. Contract W-31-109-eng-38. \$16.80(ph OTS), \$5.70(mf OTS).

3943 AECD-3712

Chicago Operations Office, AEC HISTORY AND STATUS OF THE EBR. Warner E. Unbehauen. Apr. 15, 1953. Decl. with deletions Nov. 14, 1955. 69p. \$0.40(OTS).

4262 AECD-3831

North American Aviation Inc., [Downey, Calif.]. ISOTHERMAL PUMP LOOP. R. Keen. Oct. 22, 1952. Decl. Dec. 5, 1955. 10p. Contract [AT-11-1-GEN-8]. (NAA-SR-Memo-469). \$3.30(ph OTS); \$2.40(mf OTS).

A centrifugal pump built entirely of graphite and capable of handling liquid metals in the temperature region of 1000°C is described. The pump was designed to meet the requirements of a high-temperature reactor. (J.E.D.)

4392 CF-51-8-151

Oak Ridge National Lab., Tenn.
EBR SLUG DISSOLVING. Quarterly Report for May Through July 1951. E. M. Sampson, Jr. Aug. 17, 1951. Decl. Dec. 21, 1955. 7p. Contract [W-7405-eng-26]. \$1.80(ph OTS); \$1.80(mf OTS).

Procedures for the dissolution of both Al-jacketed and bare slugs from the EBR are summarized. (B.J.H.)

4438 KAPL-494

Knolls Atomic Power Lab., Schenectady, N. Y.
CONSTANT REACTIVITY CONDITIONS FOR THE CONTINUAL OPERATION AND U^{235} START-UP OF AN INTERNAL PLUTONIUM BREEDER. T. M. Snyder. Mar. 12, 1951. Decl. Jan. 3, 1956. 8p. Contract W-31-109-eng-52. \$1.80(ph OTS), \$1.80(mf OTS).

The feasibility of designing a breeder such that its reactivity is independent of burnup (provided that the fuel is mixed with the proper quantity of fertile material) is considered to first and second approximations. A study of the feasibility of startup with U^{235} indicates that there will be a minimal control problem in making a transition from U^{235} to Pu fuel. (C.H.)

4915 AERE-RP/R-1778

Gt. Brit. Atomic Energy Research Establishment,
Harwell, Berks, England.
AN ANALYSIS OF CRITICAL ASSEMBLY AND KINETICS DATA OBTAINED FROM THE AIR-SPACED PLUTONIUM CORE ASSEMBLY (CORE II) OF ZEPHYR. A. R. Baker and D. D. McVicar. Jan. 2, 1956. 13p.

The U-spaced Pu core (Core I) used in the first critical assembly of Zephyr has been replaced by an air-spaced core. The subsequent critical assembly, the reactivity calibration of control elements, and the kinetic behavior of the reactor are described. The decay of the neutron population after shutdown was analyzed into five exponentials with periods corresponding to the five groups of delayed neutrons found by Hughes et al. The group of period 1.1 sec found with Core I is assumed to be groups 4 and 5 of Core II unresolved. With this identification there is no significant difference between the delayed neutron groups obtained with Core I and Core II. (auth)

4916 ANL-5334

Argonne National Lab., Lemont, Ill.
ESTIMATION OF FISSION PRODUCT SPECTRA IN FUEL ELEMENTS DISCHARGED FROM THE POWER BREEDER REACTOR AND THE EXPERIMENTAL BREEDER REACTOR NO. 2. I. G. Dillon and Leslie Burris, Jr. Oct. 1954. Decl. Sept. 23, 1955. 36p. Contract W-31-109-eng-38. \$0.30(OTS).

The basic data necessary for these calculations are the fission yield data as a function of mass number. Data were available from which the fast fission product spectra of Pu fissioned with 2-Mev neutrons and of U^{235} fissioned with 1-Mev neutrons could be estimated. Neither of these neutron energies duplicate conditions in the above two fast reactors in which the average fast neutron energy is estimated to be about 100 kev, but the data were considered sufficiently accurate for this purpose. Furthermore, the errors are not likely to be great and are conservative; that is, the calculated yield of such elements as Ru, Rh, and Pd which are difficult to remove by slagging will probably be slightly higher than actually found in the fuel from these reactors. For the PBR (Power Breeder Reactor) calculations, the fuel element was assumed to be in the pile for 135 days (in accordance with preliminary specifications) with fissions occurring at such a uniform rate that 2% of the total fuel element atoms would have fissioned at the time of discharge. The same bases were used for EBR-II fuel elements. Also, to show the effect of pile irradiation time, cases were calculated for 85 days

pile irradiation and 15 days cooling for both PBR and EBR-II reactors. For this range the changes in fission product spectra were small. (auth)

4917 ANL-5577

Argonne National Lab., Lemont, Ill.
SOME PROBLEMS IN THE SAFETY OF FAST REACTORS. R. O. Brittan. [May 1956]. 67p. Contract [W-31-109-eng-38]. \$0.45(OTS).

A safety survey of fast reactor systems was begun at Argonne National Laboratory by studying the characteristic kinetic behavior of such systems. To make this study possible, the problem of programming and coding the solution to the equations describing this behavior for the available digital computer (AVIDAC) was undertaken. (auth)

5297 ANL-4092

Argonne National Lab., Lemont, Ill.
PILE RESEARCH AND DEVELOPMENT DIVISION REPORT FOR PERIOD MARCH 1, 1947 THRU NOVEMBER 30, 1947. Dec. 22, 1947. Decl. Dec. 13, 1955. 38p. Contract W-31-109-eng-38. \$6.30(ph OTS), \$3.00(mf OTS).

5385 M-3753

[Clinton Labs., Oak Ridge, Tenn.]
FAST AND "MIXED" PILES. PILE TECHNOLOGY LECTURE 24. J. R. Menke. [nd]. Decl. Jan. 10, 1956. 16p. Charge \$3.30(ph OTS); \$2.40(mf OTS).

The basic characteristics of fast, U-diluted fast, and composite fast reactors are briefly considered. The operation of these reactors is compared. Equations for the excellence, E, of a reactor are included. (B.J.H.)

6359 AECD-3925

Argonne National Lab., Lemont, Ill.
EXPERIMENTAL BREEDER PROJECT REPORT FOR THE PERIOD MARCH 1, 1949 THROUGH JANUARY 31, 1950. H. V. Lichtenberger. Mar. 13, 1950. Decl. with deletions Dec. 14, 1955. 31p. Contract W-31-109-eng-38. \$6.30 (ph OTS); \$3.00(mf OTS).

The report covers the status of EBR construction; measurements of Co^{60} , Na^{24} , and fission-product γ -ray attenuation in concrete; examination of NaK-filled fuel rods and ball check rods for temperature distribution and NaK evaporation; examination of various EBR component mock-ups after operation; heat-transfer characteristics of vertical composite tube (NaK- H_2O) falling-film-type steam generator and water heater; tests of a rotating liquid seal for a liquid metal pump with a totally enclosed motor; and an experiment on the effects of pile irradiation on a U slug immersed in NaK. (G.Y.)

6997

A LETTER ON EBR-I FUEL MELTDOWN. W. H. Zinn (Argonne National Lab., Lemont, Ill.) *Nucleonics* 14, No. 6, 35, 103-4, & 119(1956) June.

The controlled experiment, to obtain the temperature coefficient of the fuel in the EBR-1, which resulted in the fuel-meltdown, is discussed. The cause of the meltdown is explained and the information yielded by this unplanned part of the experiment is evaluated. (F.S.)

7300 AECD-3844

Argonne National Lab., Lemont, Ill.
EXPERIMENTAL ... AND THEORETICAL NUCLEAR PHYSICS DIVISIONS REPORT FOR JANUARY, FEBRUARY AND MARCH, [1948]. April 9, 1948. Decl. with deletions Dec. 13, 1955. 83p. Contract W-31-109-eng-38. \$13.80 (ph OTS); \$4.80(mf OTS).

Resonance scattering data are tabulated. Scattering and transmission cross sections of S are reported. Other neutron scattering experiments are discussed. The proper-

ties of the β spectrum of Os^{193} were studied. The β spectrum of Be^{10} is also shown. The depolarization of neutrons is discussed. The magnetic nature of small angle neutron scattering is discussed. Preliminary designs of a neutron chopper are shown. The total neutron cross section curves of polycrystalline compounds CaO , TiC , TiBr , and ZnO are given. Development work on current regulators and vibrator amplifiers is discussed. A metastable state of Sc^{46} was observed. The 1.1-hr and 6.3-hr activities of Hf are discussed. Gamma rays in the Ac^{227} series are discussed. The disintegration of Sb^{124} was studied. The nuclear characteristics of Xe^{137} and Re were also studied. Fission yields in Th^{232} were determined. The recoil Br atoms from pile irradiated ethyl bromide were studied. The use of the reactor as Carlo calculations were made of the flux distribution in CP-4. The theory of closed shells in nuclei is discussed. (B.J.H.)

8678

DOUNREAY FAST REACTOR MODEL. Nuclear Eng. 1, 34-5(1956) Apr.

The construction of accurate scale models assists not only in demonstrating a particular assembly but also in checking theoretical calculations. This model of the fast reactor at Dounreay shows the disposition of the primary heat exchangers and pumping circuits relative to the core. A brief description of the operation of the reactor is appended with some comments on the safety precautions. (auth)

8949

ANL-4481
Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] MARCH 1, 1950 THROUGH MAY 31, 1950. W. P. Bigler. July 1, 1950. Decl. Dec. 13, 1955. 28p. Contract W-31-109-eng-38. \$4.80(ph OTS); \$2.70(mf OTS).

9598

AECU-3195
Atomic Power Development Associates, Inc., Detroit.
PROPOSED CONTROL OPERATION PROGRAM FOR THE POWER REACTOR DEVELOPMENT COMPANY FAST BREEDER REACTOR R. W. Klecker. June 1956 11p. \$0.15(OTS).

The APDA reactor loading, startup, and control are discussed. The methods chosen are believed best for fast breeder reactors. (D.E.B.)

9615

EXPERIMENTAL REACTOR SYSTEMS. PART 2. R. F. Jackson (Atomic Energy Research Establishment, Harwell, Berks, England). Nuclear Eng. 1, 72-7(1956) May.

The discussion begun in this journal Vol. 1, No. 1 on the zero energy reactor ZEUS, a fast breeder for use in research on fuel enrichment is completed. The following high-flux research reactors provide facilities for the chemist, metallurgist and engineer; GLEEP, an air-cooled graphite-moderated reactor, used in the study of neutron cross sections; BEPO, an air-cooled medium-power reactor, produces radioisotopes; DIDO and PLUTO are heavy water reactors, for isotope production and hot-loop studies; and LIDO, a swimming pool reactor for shielding study. (F.S.)

10502

AERE-RP/M-81
Gt. Brit. Atomic Energy Research Establishment,
Harwell, Berks, England.

THE APPROACH TO CRITICALITY OF ZEUS. K. C. Hinos and J. Egerton. May 1956. 12p.

The loading of Zeus was carried out in the following way: an outer ring of fuel elements was inserted first, then loading was commenced at the centre, fuel elements being added in such a way as to close up the gap between the inner cylindrical region of fuel and the outer annular

region. The problem of loading a reactor in this way is considered theoretically on the basis of one-group theory. For a series of values of the radius, "a," of the central fuel cylinder the appropriate eigenvalue of the system is found and from this the flux just outside the core of the reactor is calculated. These flux values as functions of "a" are plotted and compared with the experimentally determined results. (auth)

10528

TID-7506(Pt. I)
Argonne National Lab., Lemont, Ill.
PAPERS PRESENTED AT THE TECHNICAL BRIEFING SESSION ON THE BOILING WATER REACTOR PROGRAM AND THE FAST REACTOR PROGRAM HELD AT IDAHO FALLS, IDAHO, NOVEMBER 1-2, 1955. July 1956. 221p. Contract W-31-109-eng-38. \$1.25(OTS).

Included are 11 papers on boiling water and fast breeder reactor design, operation, and associated technology. (D.E.B.)

10531

ESTIMATION OF DOPPLER EFFECT IN FAST REACTORS. H. Feshbach (Massachusetts Inst. of Tech., Cambridge) and G. Goertzel and H. Yamauchi (Nuclear Development Corp. of America, White Plains, N. Y.). Nuclear Sci. and Eng. 1, 4-20(1956) Mar.

A detailed variation with energy of that part of the neutron cross section of an element which shows resonance behavior is temperature dependent. This dependence, the Doppler effect, arises from the temperature variation of neutron-nuclear relative velocity distribution. An effective cross section (dependent on reactor composition) useful in reactor calculations in place of the rapidly fluctuating actual cross section is defined. Knowledge of the variation of this effective cross section with material temperature is needed for calculation of the temperature coefficient of reactivity. Unfortunately, resolution of present measuring equipment does not permit sufficiently accurate measurement of cross sections in the energy range of interest in fast reactors (100 kev to several Mev), for Doppler effect calculation nor are direct measurements in this energy range available at present. To estimate Doppler effect, it has been assumed that in any energy range containing many resonances the actual cross section is equivalent, as far as reactor behavior is concerned, to a cross section constructed by selecting spacings between neighboring resonances and other resonance parameters independently from probability distributions of these parameters. In this manner, temperature coefficients may be calculated in terms of measured cross sections and various statistical parameters of the probability distributions, the parameters being estimated from low-energy data on actual resonances. In applying the low-energy data to the energy ranges of interest, the predictions of the statistical model of the nucleus, as developed by Weisskopf and others, are employed. (auth)

10534

AN UNREFLECTED U-235 CRITICAL ASSEMBLY. R. E. Peterson and G. A. Newby (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 1, 112-25(1956) May.

An unreflected, spherical U^{235} critical assembly has been in operation at the Los Alamos Scientific Laboratory since August, 1951. A remotely controlled mechanical system is used to assemble subcritical components of the sphere, and reactivity is adjusted with U^{235} control rods. The maximum power level during sustained operation is about 1 kw. Investigations with the assembly include studies of the neutron spectrum, observation of the changes of reactivity produced by inserting foreign materials into the assembly, and determination of parameters such as the temperature coefficient of reactivity. In addition, experiments at

reactivities above prompt critical have been carried out. The assembly has also been used as a source of short, high-intensity bursts of neutrons in the study of delayed neutrons following fission. (auth)

10535

CONSTRUCTION OF THE EXPERIMENTAL FAST-BREEDER REACTOR AT DOUNREAY Nuclear Power 1, 73-4(1956) June.

A description and diagram of the Dounreay reactor are given. Problems of construction are discussed and illustrated. (F.S.)

10961 KAPL-M-JBS-4

Knolls Atomic Power Lab., Schenectady, N. Y.
ESCAPE OF RADIATION FROM BLANKET. John B. Sampson. Aug. 13, 1948. Decl. Feb. 8, 1956. 9p. Contract W-31-109-Eng-52. \$1.80(ph OTS), \$1.80(mf OTS).

Estimates are reported on the escape of radiation from a reactor with a Be-U blanket of 10-in. thickness operating at 10 Mw. Spherical geometry with outside radius of the blanket 21.6 in. is assumed. Data for the sides are for full blanket density and for the ends are for a reduced density of $\frac{1}{3}$ full density. (F.S.)

11467 AERE-T/M-137

Gt Brit Atomic Energy Research Establishment,
Harwell, Berks, England
AN ESTIMATION OF THE DOPPLER EFFECT IN FAST NEUTRON REACTORS A M Lane, J E Lynn, and J S Syory July 1956 45p

An attempt is made to improve the estimates of temperature coefficient of reactivity effects in fast reactors. Numerical estimates are based on assumed statistical distributions of resonance level parameters, using only U^{235} and U^{238} thermal neutron data. Justification for the assumptions is presented. (D E B)

11502

FISSION AND ACTIVATION RATIOS IN THE EBR S. G. Kaufmann, F. S. Kirn, and W. B. Loewenstein (Argonne National Lab., Lemont, Ill.). Nuclear Sci. and Eng. 1, 193-203(1956) July

Miniature fission counters and thin foils are employed to determine the ratios of the effective neutron cross sections

for fission and activation of elements of interest at various locations in the Experimental Breeder Reactor at Arco, Idaho. Experimental results are compared with predictions based on multigroup diffusion theory. (auth)

11515

POWER REACTOR EXPERIMENTS. Nucleonics 14, No. 8, 38-53(1956) Aug.

A report is presented on the power reactors that are either in the process of being built or are already in operation. Operating procedures are outlined along with their designs. (W. L. H.)

12006 AECU-3196

Atomic Power Development Associates, Inc., Detroit.
SOME INHERENT AND DESIGN SAFETY FEATURES OF THE APDA FAST BREEDER REACTOR W J. McCarthy, Jr June 1956 11p \$0.15(OTS)

12040

THE WORLD'S REACTORS NO 5 ZEUS Nuclear Eng 1, 234a(1956) Sept

A data sheet with specifications and cutaway view of ZEUS, a fast heterogeneous reactor for Dounreay, is given. (F S)

12042

SCIENTIFIC PROBLEMS IN THE DEVELOPMENT OF NUCLEAR POWER. John Cockcroft (Atomic Energy Research Establishment, Harwell, Berks, England) Nuclear Power 1, 200-4(1956) Sept.

Scientific problems surveyed are those of reactor physics, solid state physics and chemistry. The prospect of high burnups and the conversion of Pu^{239} to Pu^{240} and Pu^{241} has led to a study of these isotopes. Slight enrichment of fuel for gas-cooled, graphite-moderated reactors is believed to be economical and burnups of 10,000 MWD/t are expected ultimately. On fast Pu reactors, breeding gains of 1.6 to 1.7 are expected. The most difficult problem is the development of a reliable fuel element, but radiation problems of great difficulty occur with all reactor materials. Compatibility problems between fuels, moderators, and coolants are discussed and a new Al alloy is reported. Pyro-metallurgical methods for the processing of irradiated elements show promise and will probably be used for fast reactors. (auth)

Volume 11, 1957

635 AERE-R/R-1949

Gt. Brit. Atomic Energy Research Establishment,
Harwell, Berks, England.

NEUTRON DISTRIBUTION IN THE AIR SPACED CORE
ASSEMBLY OF ZEPHYR. J. E. R. Holmes, M. U. Curry,
and M. Waldegrave. June 1956. 39p.

The nuclear reaction rate surveys which were made in the second core assembly of Zephyr are described. A detailed comparison with the first assembly leads to the following main conclusions: The core neutron spectrum in the air spaced core (Core II) has an average energy of about 1.7 Mev. This is more energetic than the core spectrum of the U spaced core (Core I), where the average energy is about 1.5 Mev. Using revised nuclear data, a neutron balance has been demonstrated for both assemblies. The breeding ratio for Core II is found to be 1.90 ± 0.15 , which is slightly less than the Core I value of 2.00 ± 0.15 . The observed delayed neutron fraction, β , is $0.34 \pm 0.01\%$ for the Core I assembly and $0.29 \pm 0.01\%$ for the Core II assembly. These values are appreciably greater than for Pu^{239} alone ($0.23 \pm 0.02\%$) and can be deduced from the fission rate surveys when the variation of neutron effectiveness with energy is taken into account. (auth)

681

PROCEEDINGS OF THE INTERNATIONAL CONFERENCE
ON THE PEACEFUL USES OF ATOMIC ENERGY, HELD
IN GENEVA, AUGUST 8-20, 1955 VOLUME 3. POWER
REACTORS New York, United Nations, 1955 389p
\$7 50

682

PROCEEDINGS OF THE INTERNATIONAL CONFERENCE
ON THE PEACEFUL USES OF ATOMIC ENERGY, HELD
IN GENEVA, AUGUST 8-20, 1955. VOLUME 5. PHYSICS
OF REACTOR DESIGN. New York, United Nations, 1956.
545p. \$9.00

1123 ANL-5548

Argonne National Lab, Lemont, Ill

SPECIAL HEAT OF AUSTENITIC STAINLESS STEEL
USED IN THE EXPERIMENTAL BREEDER REACTOR
(EBR-1) Work Completed September 1949. Robert A
Noland and David E Walker Oct. 1956 16p Contract
W-31-109-eng-38 \$0 20(OTS)

The heat of austenitic stainless steel which was used in the core structure of the Experimental Breeder Reactor is described. The heat was made by the Crucible Steel Company of America at their Midland, Pennsylvania, mill to conform with a modification of the ASTM specification A-269-47, a Nb-stabilized, austenitic grade stainless steel similar to AISI Type 347. Modification consisted of selecting a narrow range of composition within the broad range ordinarily permitted for most of the elements specified. This was done to promote austenite stability and freedom

from carbide precipitation, without altering the fabricability of the material. Components manufactured from this heat were intended to operate in sections of varying neutron flux density and temperature. They were to be exposed to NaK alloy and/or air under static and dynamic conditions at peak temperatures in the range 325° – 350°C . The steel was mill fabricated into a number of products, including rounds, rods, light and heavy plate, seamless tubing, and foil (0.020"). No difficulty was encountered in machining, grinding, deep drawing, forging, cold drawing, swaging, or welding this material. (auth)

1333

THE PHYSICS OF NUCLEAR REACTORS. A CONFERENCE
ARRANGED BY THE INSTITUTE OF PHYSICS IN LONDON
FROM JULY 3 TO 6, 1956. Brit J Appl Phys. Suppl. No
5, 1956. 110p.

Papers and discussions from the Conference on the Physics of Nuclear Reactors held in London from July 3-6, 1956 are given. Papers were presented on fast reactors; recent developments in nuclear theory; radiation effects on the physical properties of solids, instrumentation, shielding and control of reactors; and the roles of chemistry and metallurgy in nuclear energy. (F.S.)

1342

MEASUREMENT OF NEUTRON SPECTRA OF THE EXPERIMENTAL BREEDER REACTOR. C. Eggler, C. M. Huddleston, V. E. Krohn, and G. R. Ringo (Argonne National Lab., Lemont, Ill.). Nuclear Sci. and Eng. 1, 391-408(1956) Oct.

The neutron spectra at two positions in the core and two positions in the inner blanket of the Experimental Breeder Reactor in Idaho have been measured by means of beams extracted from these positions. The energy range covered was from 75 kev to 4 Mev. A cloud chamber was used for the measurement from 75 kev to 1 Mev and recoil-proton tracks in nuclear plates for that from 0.8 to 4 Mev. Reaction rate ratios measured inside the reactor agree well with those calculated from known cross sections and the spectra found in the beams. (auth)

2103 APDA-111

Atomic Power Development Associates, Inc., Detroit.
TRANSIENT TEMPERATURE SIMULATION OF REACTOR
PRIMARY LOOP. Richard G. Olson and Donald J. Ritchie.
Feb. 1, 1956. 159p. \$24.30(ph OTS); \$7.50(mf OTS).

A simulator study was performed during the months of October and November, 1955, to determine the transient behavior of the power and of the fuel and coolant temperatures of the proposed APDA Developmental Fast Breeder Reactor. Primary loop transient temperatures were also computed. The simulator study was carried out on a Reeves Electronic Analog Computer. A single lumped primary loop was used to simulate the three parallel heat

transfer primary loops of the reactor. Both a lumped single point reactor and a five-section core reactor were considered. (auth)

2104 APDA-115

Atomic Power Development Associates, Inc., Detroit.
ENRICO FERMI FAST BREEDER REACTOR PLANT.
Nov 1, 1956. 107p. \$0.55(OTS).

This report contains a description of the reactor plant portion of the Enrico Fermi Atomic Power Plant, with emphasis on the reactor system. Since this plant contains a developmental reactor, emphasis has been placed on reactor safety and on achieving reliable operation. The design is conservative and is flexible enough to permit incorporation of the results of developmental programs, which include nonnuclear, full-scale component test facility operations, and nuclear experiments, such as critical experiments. The complete reactor plant is being built in accordance with the ASME Code, wherever the code applies. The schedule for construction provides one year for non-radioactive mechanical operations and for making modifications. It also provides for a 9-month pre-operational period after the reactor goes critical. During this time the nuclear characteristics will be checked at low power with the fission product activity at a low level. The design of the reactor system is summarized. (auth)

2121 NP-5997

SOME ASPECTS OF ATOMIC POWER DEVELOPMENT IN THE U S S R I V Kurchatov 1956 12p.

The reactor development program in the U S S R is directed toward the development of power and power breeder reactors for large power station operation. Four reactor types are of major interest, heterogeneous H_2O -moderated and cooled, heterogeneous graphite-moderated Na-cooled, fast Na-cooled Pu breeder, and homogeneous D_2O -moderated Th-breeder. A brief discussion of the physics of fast and thermal breeding is given. (D E B.)

2128

THE PHYSICS OF ADVANCED REACTORS. J. R. Dietrich (Argonne National Lab., Lemont, Ill.). Brit. J. Appl. Phys., Suppl. No. 5, 9-23(1956)

The various promising types of solid fuel power reactors in terms of the neutron-physics characteristics of their moderators, coolants, and fuel cycles are characterized. The neutron-physics design objectives which have the greatest effect on power cost are use of a fertile isotope U^{238} or Th^{232} as a major source of energy, internal conversion ratio high enough to allow long fuel element life, safety characteristics favorable enough to avoid remote location and high containment cost. These objectives are not restrictive enough to define the "best" reactor type, but they are the major factors in determining the composition and geometric arrangement of specific reactor designs once the type has been chosen. The thermal reactors which are at present being designed or constructed utilize U^{235} as the initial fissile isotope, and there is a strong incentive to utilize ratios of fissile to fertile material which depart as little as possible from the "natural" value. With this additional design restriction, the lattice characteristics appropriate to the various moderator-coolant combinations are still more sharply defined. Examples are drawn from the current group of reactor designs to illustrate these characteristics. The possible consequences of a greater latitude in the choice of fissile-fertile ratio are discussed, and the relative characteristics of the possible fuel cycles are considered. The characteristics of fast reactors fit into a quite different pattern. The possible conversion ratio is high on either fuel cycle, and the reactivity changes associated with fuel burn-up are relatively low simply because of the very large fissile-material content of the reactor. The requirement that the average neutron energy be kept high is a rather

restrictive one. It appears now that the range of variation in fast reactor design will be less than that in the thermal reactor field. The characteristics of typical fast reactors are presented, their special problems are discussed, and some recent experimental results are quoted. (auth)

2130

THE ZERO-ENERGY FAST BREEDER REACTOR, ZEPHYR. L. R. Shepherd (Atomic Energy Research Establishment, Harwell, Berks, England). Brit. J. Appl. Phys., Suppl. No. 5, 32-41(1956).

ZEPHYR is a small low-power fast reactor with a plutonium core surrounded by an envelope of natural uranium. No cooling is required since the maximum operating power is about 2 w. This reactor first diverged in February, 1954, and has been used to make a detailed study of the neutron physics of fast reactors. Of the many aspects investigated, particular attention has been given to a determination of the breeding characteristics of the system and the measurement of the nuclear parameters required for a theoretical interpretation of fast reactors. (auth)

2135

THE UNITED KINGDOM RESEARCH REACTORS AND THEIR USES. F. W. Fenning (Atomic Energy Research Establishment, Harwell, Berks, England). Brit. J. Appl. Phys., Suppl. No. 5, 100-6(1956).

A brief description and use of the U. K. research reactors is given. Five are in operation: GLEEP, BEPO, DIMPLE, ZEPHYR, and ZEUS. (auth)

3082

DECAY-HEAT COOLING REQUIREMENTS OF SPENT FUEL FROM EBR-2. C. H. Scheibelhut (Reactor Engineering Div., Argonne National Lab., Lemont, Ill.). Nucleonics 14, No. 12, 57-60, 62(1956) Dec.

Fuel elements will be removed from the EBR-2 reactor after they have served for three months and have achieved 2% burnup. After being removed from the core they will be stored 15 days in the reactor Na so that the initially intense heat from fission-product decay can be removed. Then the fuel subassembly will be transferred to a cell where the irradiated fuel will be removed for processing. (W.L.H.)

3575

EBR-1 CORE: AFTER MELTDOWN. Nucleonics 15, No. 1, 84(1957) Jan.

4110

SAFEGUARD FEATURES OF A FAST-BREEDER-REACTOR POWER PLANT. W. J. McCarthy, Jr. (Atomic Power Development Associates, Inc., Detroit) and F. C. McMath (The Detroit Edison Co., Mich.). Trans. Am. Soc. Mech. Eng. 79, 41-6(1957) Jan.

In the design of a reactor power plant special attention to safety is required in order to safeguard the investment in the plant and to insure that danger to the public is reduced to negligible proportions. Some design features relevant to the safety of a fast-breeder-reactor power plant are described. The decision to enclose the radioactive portions of the plant in a gastight building to provide a final barrier between the reactor and the public is discussed. Criteria developed for locating the reactor on a safe and economically suitable site are presented. (auth)

4660 AERE-RP/M-48

Gt. Brit. Atomic Energy Research Establishment,
Harwell, Berks, England
ZEPHYR PERTURBATION STUDIES. Interim Report

H Rose and R D Smith Sept 17, 1954 Decl Nov 26, 1956 5p

The reactivity changes produced by the introduction of small samples of about 50 elements into the core of the Zephyr fast reactor have been measured Effective capture and transport cross sections are tabulated (auth)

4661 AERE-RP/M-49

Gt Brit Atomic Energy Research Establishment, Harwell Berks, England

FLUX MEASUREMENTS IN ZEPHYR Interim Report J E R Holmes [1954?] Decl Nov 26, 1956 21p

The variation of the fission rates of Pu^{239} , U^{233} , U^{235} , U^{238} , and natural U has been investigated throughout the reactor The fission cross-sections of these materials have been deduced assuming the cross section for fission of Pu^{239} has a constant value of 1.98 barns in the reactor These measurements indicate that an equilibrium spectrum exists in the core and, as the neutrons diffuse outwards, this distribution becomes degraded by inelastic scattering, until a second equilibrium spectrum is reached in the outer parts of the envelope The values obtained for cross sections in the core correspond in most cases, to the cross sections for monokinetic neutrons of about 600 kev. This may be compared with a value of about 1 Mev to be expected as the 'equivalent energy' for an undegraded fission spectrum The corresponding 'equivalent energy' for the equilibrium spectrum in the envelope is about 100 kev The diffusion length in natural U, based on an assumption of spherical symmetry and applying one group theory, has been found to be 11.9 ± 0.4 cms, in agreement with the previously reported value Integrations have been made of the fission rates over the whole of the reactor These integrations, when combined with estimates of capture in U^{238} and neutron leakage from the outside of the envelope, indicate that the conversion factor for Zephyr is approximately 2 (auth)

4664 ANL-5601

Argonne National Lab, Lemont, Ill

REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] APRIL, MAY, JUNE 1956 SECTION II Dec. 1956 89p Contract W-31-109-eng-38 \$0.50(OTS)

Advanced water reactor program Progress on the design of the Experimental Boiling Water Reactor is summarized A mockup of the pressure vessel and part of the shield of the EBWR were installed in the Bathtub Shielding Facility in the CP-5 in order to determine the relative effects of mild steel and boron-stainless steel thermal shields on thermal neutron flux at the pressure vessel wall as a function of the thickness of the water gap between the thermal shield and the vessel wall Data on the compressive strength of concrete shield specimens are tabulated. The construction status of the EBWR is also reviewed The theory of the stability of bubbling reactors is outlined Flow models for boiling water passing upward through heated rectangular channels are discussed The effect of oxygen addition on water decomposition is also studied. Data are included on the effect of fused salt treatment on the corrosion of Zircaloy-2 in static degassed water at 680°F. Data are also given on the corrosion of Zircaloy-2 ingots and Al alloys, the boiling corrosion of Al-Ni clad stainless steel tube, and on the corrosion-irradiation of Al-Ni alloys Irradiation tests on urania-thoria fuel pins are summarized The temperature gradient through the fuel pin was calculated The stress distribution in the pin is also shown Data are tabulated on the dynamic and static corrosion of boron-stainless steel Some data are also included on the static corrosion of miscellaneous structural materials Fast power breeder reactor program The nuclear properties of an EBR-II design study were evaluated on the basis of one-dimensional diffusion theory calculations Irradiation tests of EBR-II type fuel elements

are summarized Heat transfer properties and thermal stresses for the insulation in the EBR-II primary Na system are listed Temperature profiles through the inner pipes of the system are shown The operation of the EBR-II control rod mechanism is also discussed (For preceding period see ANL-5572) (B J H)

4985

DELAYED NEUTRON INVESTIGATIONS WITH THE ZEPHYR FAST REACTOR. PART I PNEUMATIC TRANSFER SYSTEM. R D. Smith, D D. McVicar, and E. A Thorne (Atomic Energy Research Establishment, Harwell, Didcot, Berks). PART II. THE DELAYED NEUTRONS ARISING FROM FAST FISSION IN U^{235} , U^{233} , U^{238} , Pu^{239} , AND Th^{232} H. Rose and R D. Smith (Atomic Energy Research Establishment, Harwell, Didcot, Berks). J Nuclear Energy 4, 133-46(1957) Feb.

A pneumatic transfer system used in delayed neutron experiments is described The system enabled small samples of fissionable materials to be irradiated near the reactor core for preset times of $\frac{1}{2}$ sec or longer and transferred in less than one second to a remote neutron detector The yields of five groups of delayed neutrons resulting from fast fission of U^{235} , U^{233} , U^{238} , Pu^{239} , and Th^{232} were investigated and compared with previous results Some evidence was found that a short-lived precursor contributes to a small extent to the observed twenty-two-second group. (auth)

5608

THE ENRICO FERMI POWER REACTOR. Nuclear Eng. 2, 112-13(1957) March.

6089 NDA-14-107

Nuclear Development Associates, Inc, White Plains, N Y PROGRESS REPORT ON THE STUDY OF LOW PROBABILITY, HIGH HAZARD, FAST REACTOR ACCIDENTS R Daane, R Mela, and J DeFelice Nov. 11, 1955 56p \$0.40(OTS)

The methods being used to analyze highly hazardous but highly unlikely nuclear accidents are described. The assumptions made require more examination and, therefore, the results are probably too pessimistic for the model selected by factors of ten or a hundred Two accidents are described in detail; rapid assembly and slow meltdown In the rapid assembly, the core is considered to compress with a radial velocity of 15 ft/sec The resultant rate of reactivity increase is \$1250/sec As the reactivity builds up, the core temperature increases, fuel is vaporized and eventually pressures are built up in the reactor which force the blanket outward and enable the core to increase in size to the extent that the reactor is shut down This involves the generation of about 5000 megacalories of energy (5 tons of TNT) and pressures of up to 5×10^5 atm These numbers are based on several assumptions which require more examination, among which are the following. The blanket completely surrounds the reactor. In practice, there are sodium header regions between the reactor and the end blankets. Equilibrium thermodynamic relationships are valid. Heat is generated uniformly throughout the core This is probably the only assumption which may not lead to a conservative result The reactivity remains constant from the time it reaches its maximum value until shutoff Actually, the reactivity builds up to some maximum value during the accident, and then decreases until shutoff Following a sudden loss of sodium coolant a slow meltdown of the core occurs in which molten uranium, steel, and perhaps zirconium come to rest, at least temporarily, on the graphite thermal shield beneath the reactor. If the melt then forms a critical mass, large amounts of energy could be released. If it does not become critical, the melt can still overheat the

containment vessel and release fission products to the environment or it could spread over a wide area of the graphite and cool harmlessly. More study is required to determine if all of these possibilities could occur. Two further accidents are considered; hydrogen insertion and autocatalysis. The first is shown to be less serious than the rapid assembly, while an answer to the second is obtained directly from the results of the rapid assembly. The design used in these calculations has changed since these calculations were made. The changes as of this date, however, will not appreciably affect the over-all results, because of the large uncertainty due to the assumptions made in the analysis. (auth)

6102

FAST BREEDER POWER REACTORS. THEIR PROBLEMS AND PROSPECTS. *Nucleonics* 15, No. 4, 62-6(1957) Apr.

The problems of fitting a fast breeder power reactor into a power reactor economy are described. The major drawbacks for this type of reactor are operation at high core-power density, problems of Na technology, high fuel costs because of large inventory and frequent reprocessing, and reactor safety. (W.L.H.)

6103

ENRICO FERMI REACTOR. DESIGN OF FIRST FAST BREEDER. *Nucleonics* 15, No. 4, 68-72(1957) Apr.

The design of the Enrico Fermi fast breeder power reactor is designed to generate 300 Mw of heat and deliver a net output of 90 Mw of electric power. It is fueled with 27%-enriched U and will breed Pu with a breeding ratio of 1.20. Heat is transferred from reactor to steam plant by primary and secondary Na coolant circuits. (W.L.H.)

6521

ANL-5657
Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT—SECTION II [FOR] JULY, AUGUST, SEPTEMBER 1956. Dec. 1956. 69p. Contract W-31-109-eng-38 \$0.40(OTS).

Experimental Breeder Reactor II. A study was made of the coolant flow characteristics of the primary system. A schematic diagram of the system is shown, and flow rates and pressure drops in the system are shown graphically. Data are also included on pump rectifier voltages. Illustrations are given of the pumps for the heat transfer systems. A preliminary test was made of the 5000-gpm Na pump using H_2O as the circulating fluid. A pump efficiency of 79% was obtained at the design flow rate. A test facility was constructed to determine the performance of a low-melting alloy, gas tight seal for the rotating plugs. Supporting and Alternate Design Research and Development. Calculations were made to investigate the reactivity effects of inhomogeneities in the ZPR-III facility. Calculation methods and results are described in detail. Heating tests were made on fission alloy pins and the results are tabulated. Tests were made to evaluate gears and bearings for service in high-temperature Na. The apparatus used is shown and complete test results are tabulated. Experiments were also conducted on the Na-air reaction. (For preceding period see ANL-5601.) (B.J.H.)

6810

DYNAMIC SIMULATION OF A FAST REACTOR. R. G. Olson (Atomic Power Development Associates, Inc., Detroit, Mich.). *Nucleonics* 15, No. 5, 76-9(1957) May.

The analysis of a startup accident and a loss-of-pump-power accident in a fast reactor by an analog computer is described to demonstrate the utility of analogs as reactor simulators. (T.R.H.)

6887

APDA-108
[Atomic Power Development Associates, Inc., Detroit]
DESCRIPTION OF DEVELOPMENTAL FAST NEUTRON BREEDER POWER REACTOR PLANT. Sept. 1, 1955. 104p. \$0.55(OTS).

The general characteristics of Na-cooled fast reactors are briefly reviewed. A complete description is given of the proposed APDA fast breeder reactor, including a tabulation of pertinent technical data and descriptions of the core and blanket subassemblies, the reactor compartment, mechanical handling systems, the reactor control system, the liquid metal and steam-power systems, and the disassembly and shipping facility. Site considerations are also discussed in detail. The physical characteristics, geology, seismology, and population distribution of the reactor plant is also considered. (B.J.H.)

6899

NP-6268
Max-Planck-Institut für Physik, Göttingen
ZUR THEORIE DER REAKTOREN MIT SCHNELLEN NEUTRONEN. (The Theory of Reactors With Fast Neutrons.) [Report] No. 1. R. Schulten. [1954?]. 9p.

6930

<<ZEPHYR>>—ZERO ENERGY FAST REACTOR. T. M. Fry (Atomic Energy Research Establishment, Harwell, Eng.). *Nuovo cimento* (10) 4, Suppl. No. 3, 1324-8(1956).

A general description of the Zephyr is given. Topics given special emphasis are neutron flux, conversion factor, and perturbation measurements. (M.H.R.)

7321

AERE-R/R-2151
Gt. Brit. Atomic Energy Research Establishment
Harwell Berks, England
STUDIES ON ZEPHYR WITH A THORIUM ENVELOPE
R. M. Absalom, B. G. Chapman, J. E. R. Holmes, H. Rose, T. B. Rives and M. Waldegrave. Jan. 31, 1957. 35p.

An account is given of the measurements made on Zephyr with an inner envelope of natural uranium and an outer main envelope of thorium. The main conclusions reached are that the neutron spectrum is more degraded in a thorium envelope than in a uranium envelope, the breeding ratio of the mixed assembly was 1.8 ± 0.2 , and, the breeding ratio of Zephyr with an entirely thorium envelope was estimated at 1.5 ± 0.2 . (auth)

7332

FIRST LARGE SCALE BREEDER REACTOR. THE U.K. A.E.A.'S NEW INSTALLATIONS AT DOUNREAY. *Atomics* 8, 158-61(1957) May.

The Dounreay reactor is fuelled with uranium and U^{235} or plutonium, has a blanket of depleted uranium, and is cooled by means of a sodium-potassium alloy. With an output of 60 Mw thermal, and 20 Mw electrical energy, this will be world's first large-scale breeder reactor, and will assist Britain's electrical power program. The engineering problems involved in the construction of the sphere which is 135 ft in diameter and the system for the disposal of effluents below the sea bed are dealt with in some detail. (M.H.R.)

7547

NAA-SR-1435
North American Aviation, Inc., Downey, Calif.
THE PYROCHEMICAL RECYCLING OF PLUTONIUM FROM SPENT TO FRESH URANIUM FUEL USING MOLTEN UF_4 AS TRANSFER MEDIUM. Archie G. Buyers. Oct. 1, 1955. Decl. Feb. 27, 1957. 31p. Contract AT-11-1-GEN-8. \$4.80(ph.OTS), \$2.70(mf.OTS).

The pyrochemical transfer of Pu from spent to fresh U using molten UF_4 as transfer agent is demonstrated. The resultant U-Pu alloy contained fission products in small fractions of the amounts originally present in the spent

fuel. Distribution of Pu and fission products throughout re-cycling is correlated with thermodynamic data. Application of this method for control of Pu concentration in a Na graphite reactor fuel cycle and fast breeder reactors is suggested (auth)

7629 ANL-5489

Argonne National Lab., Lemont, Ill.
METALLURGY DIVISION QUARTERLY REPORT [FOR] JULY, AUGUST, AND SEPTEMBER 1955. Mar. 1956. Decl. Feb. 12, 1957. 67p. Contract W-31-109-eng-38. \$0.50(OTS).

Progress is reported on the following investigations, fabrication and evaluation of uranium alloy fuel plates for EBWR, radiation damage in EBR-II slugs, radiation effects on U-Pu and U-Si alloys, U powder compacts, U, and UO_2 , corrosion resistance of Nb-U-Zr, U-Si, Al-Ni, and Al alloys, preparation of single crystals of alpha U, phase studies and microstructures of U-Zr and U-Nb-Zr alloys, and vacuum fusion determination of H_2 in Al. (For preceding period see ANL-5465) (C W.H.)

7690 ANL-5175

Argonne National Lab., Lemont, Ill.
PHYSICS DIVISION SUPPLEMENT TO QUARTERLY REPORT [FOR] SEPTEMBER, OCTOBER, AND NOVEMBER 1953 Feb. 1954. Changed from SECRET Nov. 29, 1955. Decl. Feb. 12, 1957. 27p. Contract W-31-109-eng-38. \$4.80(ph OTS), \$2.70(mf OTS).

Experimental Nuclear Physics. An investigation of the isomeric transition associated with Am^{242m} failed to reveal any L-conversion lines for either E3 or M1 transitions. New data are given for the total neutron cross sections (kev region) for Li^6 , Li^7 , Th, U, U^{235} , and Pu. Mass Spectroscopy. Values of α for U and Pu in the EBR have been determined for extended points within the core and blanket. Experimental Reactor Physics. Initial design factors for the source reactor of the Fast Exponential Experiment are outlined. Four irradiated NRX fuel rods were analyzed for change in reactivity (K S)

7707 ANL-4284

Argonne National Lab., Lemont, Ill.
THEORETICAL NUCLEAR PHYSICS DIVISION. Report for Period Ending March 28, 1949. May 2, 1949. Decl. Feb. 11, 1957. 61p. Contract W-31-109-eng-38. \$9.30(ph OTS); \$3.60(mf OTS).

A mathematical analysis is made of neutron attenuation in a conical shield. An analysis is also made of the efficiency of a collection chamber for Br ion collection as a function of the charge, mass, and energy of the Br nuclei, the radii of the outer and inner cylinders of the chamber, and the potential difference between the cylinders. Information on the shell structure of nuclei is summarized. Calculations were made by the Monte Carlo method on the neutron distributions in two fast reactors, the Experimental Breeder Reactor and the Los Alamos Fast Reactor. Results are given in both tabular and graphical form. Calculations were also carried out on a proposed research reactor for the Argonne National Laboratory. Included are studies of two-group constants, critical mass and flux distributions, and adjoint fluxes. (B.J H)

7718 ANL-4274

Argonne National Lab., Lemont, Ill.
EXPERIMENTAL BREEDER PROJECT REPORT FOR THE PERIOD DECEMBER 1, 1948 THRU FEBRUARY 28, 1949. H. V. Lichtenberger. Apr. 12, 1949. Decl. Jan. 17, 1957. 34p. Contract W-31-109-eng-38. \$6.30(ph OTS); \$3.00(mf OTS).

7720 ANL-5009

Argonne National Lab., Lemont, Ill.
CASTING OF BLANKET BRICKS, RING, PLUG, AND CONTROL RODS FOR THE EXPERIMENTAL BREEDER REACTOR (CP-4). Robert E. Macherey. Mar. 19, 1953. Decl. Feb. 12, 1957. 47p. Contract W-31-109-eng-38. \$0.45(OTS).

The cup assembly of the EBR consists of blanket bricks, inner rings, and safety plug, all of natural U. The design and manufacture of the parts are described. Thermal cycling data and manufacturing conditions are presented for the control and safety rods (L T.W)

7727 APDA-102

Atomic Power Development Associates, Detroit
INFORMATION REPORT BY ATOMIC POWER DEVELOPMENT ASSOCIATES COVERING WORK FOR THE PERIOD AUGUST 1, 1954 TO JANUARY 31, 1955 Jan. 31, 1955. Decl. Jan. 7, 1957. 82p. \$0.55(OTS).

Five-group diffusion theory equations were used in calculations performed to assist in selection of a fast breeder reactor engineering design, and results are tabulated. Analyses were conducted to determine whether the rapid expansion of solid fuel elements during a prompt critical accident can stabilize a fast reactor. Data are given on the effect of allotropic transformations on, and grain refinement of Cr-U alloys. Corrosion tests were made on Globeiron, zirconium tubing, and titanium. Dynamic and static corrosion testing was performed on various other materials, and results are given. The effect of Na on the self-welding of two metals is being investigated. The prolonged operation of a piston-type Johnson valve in Na was studied. Progress is reported on fuel element design and development. Analyses of the thermal performance of the elements was made. The fabrication of cast, plate-type, perforated-wafer, and pin-type fuel elements is described, and figures are shown of their assemblies. Testing and economic studies of the elements are being made. Blanket design and specifications are discussed. A cross sectional view of the reactor vessel design is given. Investigations are being made on fuel and blanket element support. Fuel loading and unloading equipment and reactor compartment arrangements have been further studied. The mechanical design of the control rod is discussed and a cross sectional view of the rod is given. The coolant flow characteristics through the reactor core during scram conditions are studied and typical scram histories are shown. A diagram of the liquid and steam-power systems is given. The intermediate heat exchanger is also shown. Diagrams of the once-through-type steam generator and the reactor and steam plant elevation are also shown. A facility for NaK-water reaction tests has been designed. Diagrams of the boiler test facility, the once-through-type test boiler and thermocouple locations for the proposed steam generator unit are shown. An analysis of relative merits of on- and off-site fuel reprocessing has been made. The design of the full-scale reactor and Na loop test facility is also discussed. (For preceding period see DCDE-101.) (B J H)

7825 IDO-14331

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
PROCESSING OF THE CORE FUEL USED INITIALLY IN THE EXPERIMENTAL BREEDER REACTOR (EBR-I) A. E. Erhard, C. L. Murray, R. F. Murray, B. J. Newby, K. L. Rohde, F. P. Schilling, and F. M. Warzel. Dec. 15, 1955. Decl. Feb. 23, 1957. 46p. Contract AT(10-1)-205. \$0.40(OTS).

A detailed description is given of the processing of the enriched U from the first core of the EBR I, together with the cold runs preceding the active processing. The fuel and the Al shipping cans were dissolved in HNO_3 with H_2

catalyst Only two hexone solvent extraction cycles were used on 10 Kg per day flowsheet for decontamination and the product $\text{UO}_2(\text{NO}_3)_2$ as shipped from ICPP was of exceptional radiochemical purity (auth)

7839 LWS-22534

California Research and Development Co., Livermore, Calif.

RECOMMENDED PBR CONCEPTUAL DESIGN PROGRAM.

J. E. Mahlmeister, W. H. Harker, H. L. Hollister, J. B. Stichka, and E. D. Kane. Mar. 10, 1953. Decl. Feb. 15, 1957. 23p. (CRD-A27-43) \$4.80(ph OTS); \$2.70(mf OTS)

A study of the engineering limitations of the core design of the Argonne Power Breeder Reactor showed the metallurgical limitations of high temperature materials, corrosive effects of Pu, and thermal stresses to be the most severe. Preliminary specifications for solid fuel and liquid fuel reactors are included. (F S)

7840 LWS-24612

California Research and Development Co., Livermore, Calif.

FAST REACTOR POISONING BY FISSION PRODUCTS.

C C Old Oct 24, 1952 Decl Feb 26 1957 8p (CRD-R-1) \$3 30(ph OTS), \$2 40(mf OTS)

Estimates are made of fast reactor poisoning by fission products. Data are included on the neutron absorption and production of fission products in reactors. (B.J.H)

7856 ORNL-1137

Oak Ridge National Lab., Tenn.

SOLUTION OF EXPERIMENTAL BREEDER REACTOR SLUGS E M Sampson, Jr Mar 11, 1952 Decl. Mar 5, 1957 45p Contract W-7405-eng-26 \$0 40(OTS)

A full-scale, metal dissolver for Experimental Breeder Reactor fuel was designed, built, and installed for test operation. It was found that the dissolver operated satisfactorily, and feasible operating procedures were established for the dissolution of bare, or jacketed, EBR slugs. Minor modifications of the dissolver design have been required to accommodate a modified EBR slug, but it is believed that this will not significantly affect its operating characteristics. (auth)

7863 TID-5351

Argonne National Lab., Lemont, Ill.

EBR OSCILLATIONS. J. R. Dietrich. Jan. 4, 1956. 8p. Decl Mar. 5, 1957 8p Contract [W-31-109-eng-38] \$0.25(OTS)

The relationship of temperature coefficients to reactivity oscillations in the EBR is discussed. Equations are developed for describing oscillations originating in negative temperature coefficients. (D.E.B.)

7864 TID-7506(Pt II)

Argonne National Lab., Lemont, Ill

PAPERS PRESENTED AT THE TECHNICAL BRIEFING SESSION ON THE BOILING WATER REACTOR PROGRAM AND THE FAST REACTOR PROGRAM, HELD AT IDAHO FALLS, IDAHO, NOVEMBER 1-2, 1955. July 1956. Decl. Jan 17, 1957 40p. Contract W-31-109-eng-38. \$0.40 (OTS).

The processes involved in the fabrication and testing of EBWR fuel elements are briefly summarized. A general discussion is also given on the processing of spent fuel from fast reactors. (B J H)

8180

DOUNREAY. THE FAST REACTOR, THE SPHERE, THE REACTOR VAULT, THE CORE, THE CORE VESSEL, PRIMARY HEAT EXCHANGERS, SECONDARY HEAT EX-

CHANGERS, CIVIL ENGINEERING, AND ELECTRICAL EQUIPMENT Nuclear Eng. 2 230-44(1957) June

8181

THE WORLD'S REACTORS NO. 12. DOUNREAY. Nuclear Eng. 2, 238a(1957) June.

Detailed diagrams are given of the Dounreay Fast Reactor and its core vessel The reactor specifications are summarized (M H R)

8323 ANL-4530

Argonne National Lab., Lemont, Ill.

TRIBUTYL PHOSPHATE SOLVENT EXTRACTION PROCESS FOR RECOVERY AND DECONTAMINATION OF URANIUM 235 FUEL DISCHARGED FROM EXPERIMENTAL BREEDER REACTOR. Leshe Burris, Jr. and Richland C. Vogel. Sept 15, 1950 Decl. Feb. 13, 1957. 144p Contract W-31-109-eng-38 \$22 80(ph OTS); \$7.20(mf OTS).

Laboratory and semi-works studies on the development of a solvent-extraction separations process for enriched uranium are described. The process, using TBP diluted with CCl_4 as the solvent, was developed to meet the decontamination and recovery requirements of enriched fuel discharged from the Experimental Breeder Reactor. 37 figures. (C.H.)

8384 LWS-22654

California Research and Development Co., Livermore, Calif.

APPLICATION OF AQUEOUS PROCESSES TO FBR

BLANKET PROCESSING. M. L. Feldman. June 2, 1953. Decl Feb. 27, 1957. 21p. (D-T-16) \$4 80(ph OTS), \$2.70(mf OTS)

Aqueous processing can be simplified considerably for processing FBR blanket material It is also concluded that in order to achieve the optimum in an aqueous process simplification, a UO_2 blanket and a Pu salt or dioxide core should be used It is recommended that a program be launched to develop an aqueous process utilizing adsorption and precipitation techniques either alone or in combination It is also recommended that a process be investigated using a Purex IA column followed by the precipitation of Pu and U from the organic extractant (auth)

8562 ANL-4798(Del)

Argonne National Lab., Lemont, Ill.

PHYSICS DIVISION QUARTERLY REPORT FOR DECEMBER 1951, JANUARY AND FEBRUARY 1952 Mar 15, 1952 Decl with deletions Feb 12, 1957 80p Contract W-31-109-eng-38 \$12 30(ph OTS); \$4 50(mf OTS)

The scattering of neutrons has been measured at five angles at four energies for U, Zr, Fe, and Cu, all relative to the scattering by carbon at the same angle and energy. The neutron source was the $\text{Li}(p,n)$ reaction; the scattering and detecting setup is shown. New values for neutron resonances in As^{75} , U^{233} , normal U, and normal Zr and neutron transmission cross sections of Al^{27} , Fe, Ni^{58} , Zr, and Bi^{209} are tabulated. A study was made of liquid scintillation counters for neutrons using methyl borate-terphenyl in phenylcyclohexane mixtures. The theoretical aspect of the property of boron esters in such counters is given. The Xe content of the atmosphere was found, by isotopic dilution, to be 0.109 ppm (pressure). The atomic magnetic moment orientation, as found by neutron diffraction, is reported for the antiferromagnetic CrSb . The crystal structure of Np metal, found from x-ray powder photographs is given in detail and interatomic distances are reported for NpC , Np_2C_3 , NpC_2 , HfO_2 , and H_2BO_3 . The fast-fission factor E, the probability of resonance escape p, the thermal utilization factor f, and the multiplication constant k are calculated for a one-dimensional lattice of natural U and H_2O . The critical properties of spherical water-boiler reactors, for use as low-level neutron sources, are discussed; values

are tabulated for a core of $\text{H}_2\text{O}-\text{U}^{235}$ and a reflector of BeO , graphite, or a mixture. An attempt is made to explain the unexpected critical mass of the Experimental Breeder Reactor. Cross sections for neutron inelastic scattering by middle-weight nuclei are calculated and plotted vs. neutron energy. The order of energy levels and the spin of K^{40} and Cl^{36} nuclei are investigated, using jj -coupled wave functions. Finally, theoretical consideration is given to the interaction between the configurations $d^{n-5} sp$ and $d^{n-1} p$ (iron group), a direct method for determining the transformation matrix for the Dirac equation, and an iterative formula for approximating an eigenvalue of a linear 2nd-order ordinary differential equation. (auth)

8665 DOW-NR-55001-1-S

Dow Chemical Co. Nuclear Research Lab., Midland, Mich

PROPOSED REACTOR FOR POWER COMBINING FEATURES OF LMFR-SGR. Feb. 1, 1955. Decl Feb. 25, 1957. 210p. Contract AT(11-1)-329. \$0.40(OTS).

A program is outlined for the design and development of a commercial reactor to produce power and process steam. It is recommended that the basic structure follow the design of the Sodium Graphite Reactor but should be adapted for liquid metal fuel and equipped with a blanket. Ways and means are suggested for reducing developmental steps, simplifying installation, and cutting costs. If all liquid fuel experiments are complete failures, the basic structure can be converted into a sodium graphite reactor (M.P.G.)

8678 K-1173

Massachusetts Inst of Tech, Oak Ridge, Tenn Engineering Practice School

A PARAMETER STUDY OF THE K-49 REACTOR BLANKET. W. A. Stewart, J. R. Buchanan, Jr., R. E. Lueders, and A. B. Reynolds. Oct 5, 1954. Decl. Feb 23, 1957. 56p. Contract W-7405-eng-26, Subcontract 70. \$0.50(OTS).

The ratio of Pu^{240} to Pu^{239} produced in the blanket of the proposed K-49 reactor has been derived as a function of reactor design and certain physical parameters. Two different reactor designs were selected, and a study is presented of the important parameters which affect the isotope ratio. The possibility of studying plutonium chemistry under reactor conditions, and determining plutonium solubility and precipitation rate using an in-pile loop in the LITR was also considered (auth)

8682 LWS-24701

California Research and Development Co., Livermore, Calif.

FAST REACTOR COOLANT COMPARISON. H. A. Brammer. Dec 15, 1952. Decl Feb 27, 1957. 11p (CRD-R-17) \$3.30(ph OTS), \$2.40(mf OTS)

Calculations are made of the variation in critical mass of a fast reactor caused by several possible coolants. The coolants considered are Na, NaK, S, Pb, Bi, and Hg. Each system is calculated for U, Fe, Pb, and Bi reflectors. The derivation of the critical equation for a spherical core with an infinite reflector is included in an appendix. (B.J.H.)

9107 NAVORD-4470

Naval Ordnance Lab, White Oak, Md.

CALCULATION OF THE ENERGY ABSORPTION POTENTIAL FOR THE BLAST SHIELD OF THE ARGONNE NATIONAL LABORATORY NUCLEAR REACTOR, EBR-II. Walter R. Wise, Jr. Sept. 25, 1957. 22p

The blast shield of the ANL reactor, EBR-II consists of cylindrical laminae of steel, celotex, and wood, the function of which is to absorb a portion of the energy which would be released by the reactor in the event of a nuclear excursion

General equations are derived for determining the energy absorption potential for the shield in terms of radial strain of the outermost steel cylinder. For the particular case where this strain is three inches, it is estimated that the energy absorbed by the blast shield would be equivalent to the detonation of about 1200 pounds of TNT (auth)

9110

THE BUILDING OF DOUNREAY. Atomics 8, 201-6(1957) June.

The construction of the Dounreay Fast Breeder Reactor is described. (W.L.H.)

9124

THE DOUNREAY FAST REACTOR. HERE IS HOW IT WORKS. Nuclear Power 2, 240-1(1957) June.

A 3 ft by 2 ft full color cutaway drawing of the Dounreay Fast Reactor is given. The fuel loading and power and fuel production are described. (M.H.R.)

9125

DFR TANK AND HEAT EXCHANGE SYSTEM MANUFACTURE AND INSTALLATION. Nuclear Power 2, 229-31 (1957) June.

The planning for access to parts, the tolerances in fabrication, the quality control, the intricacies of the welding, and other problems in the manufacture of the Dounreay Fast Reactor tank and heat exchange system are discussed. A large diagram of the system is included. (M.H.R.)

9126

DOUNREAY WORKS CHEMICAL AND METALLURGICAL PLANT. Nuclear Power 2, 232-4(1957) June.

Flow diagrams and description of the movement of reactor fuels through the chemical plant at Dounreay are given. (M.H.R.)

9128

CIVIL ENGINEERING AT DOUNREAY. D. Cameron Lindsay (Whatlings Ltd.). Nuclear Power 2, 236-9(1957) June.

Discussions on civil engineering problems such as concreting in the sphere, sea water pump-house, effluent tunnel, safety precautions, combating the weather, and concrete mixing are included. (M.H.R.)

9129

THE DOUNREAY FAST REACTOR. Nuclear Power 2, 224-8(1957) June.

Various phases of the operation of the Dounreay Fast Reactor are discussed. Among these are shutdown heat removal, fuel element discharging, and heat exchanger operation. (M.H.R.)

9465 AI RE-R/R-2150

Gt Brit Atomic Energy Research Establishment, Harwell, Berks, England

THE EFFECTIVENESS OF NEUTRONS AS A FUNCTION OF THEIR ENERGY IN THE NATURAL URANIUM ENVELOPE OF ZEPHYR. H. Rose and R. M. Absalom. Mar 4 1957. 14p. \$0.45(OTS)

The variation of neutron effectiveness with energy has been investigated in the natural U envelope of Zephyr. The experiments involved using the reactor in a steady, sub-critical condition, and measuring the relative changes in reactor flux produced by calibrated neutron sources of different mean energies when inserted into the reactor envelope. In all, five photo-neutron (γ, n) and two (α, n) sources were used, ranging in average energy from 0.03 to 5.0 Mev. Measurements were carried out at five positions, and results obtained for the dependence of neutron

effectiveness upon neutron energy and also upon distance from the reactor core. It is shown that the lower energy neutrons (< about 100 kev) in the envelope contribute very little to the reactivity of the Zephyr system. (auth)

9483

DOUNREAY. PART II. Nuclear Eng. 2, 286-93(1957) July.

The Dounreay Materials Testing Reactor and Dounreay Fast Reactor, their fuel cycles, chemical separation schemes, fuel element manufacture, effluent disposal, and fuel element examination are described. (T.R.H.)

9580 ANL-5494(Del.)

Argonne National Lab., Lemont, Ill.

CHEMICAL ENGINEERING DIVISION SUMMARY REPORT [FOR] JULY, AUGUST, AND SEPTEMBER 1955. Nov. 2, 1955. Decl. with deletions Feb. 5, 1957. 117p. Contract W-31-109-eng-38. \$18.30(ph OTS); \$6.00(mf OTS).

Progress is reported on the following investigations: entrainment of dissolved radioactivity from a boiling water surface; dispersion and coalescence in immiscible liquid systems; fused salt-fluoride volatility process for U-Zr fuel materials including dissolution of U-Zr and Sn-Zr in fused NaF-ZrF₄; design of shielded, vacuum-casting U furnace; dissolution of fissium as stand-in for EBR-II fuel elements; liquid Mg extraction of Pu; corrosive effects of Mg on Fe alloys; spontaneous ignition of U metal; Mo determination by solvent extraction of the xanthate into CaCl₂. (T.R.H.)

9581 ANL-5633

Argonne National Lab., Lemont, Ill.

CHEMICAL ENGINEERING DIVISION SUMMARY REPORT [FOR] JULY, AUGUST, AND SEPTEMBER 1956. Dec. 1956. Decl. Mar. 14, 1957. 147p. Contract W-31-109-eng-38. \$0.80(OTS).

Solvent Extraction: Additional runs have been made in the six-inch, continuous-flow mixing chamber to study the rate of mass transfer between isobutanol and H₂O.

Fluoride Volatilization Separations Process: Work has continued on the processing of Zr-U fuel alloys in fused fluoride systems, corrosion tests on graphite, behavior of Pu during U volatilization step of the fused salt process, crude UF₄ produced from ore concentrates by hydrogenation and hydrofluorination in fluidized bed reactors, and separation of MoF₆ from UF₄ by distillation. Kinetic data were obtained on the gas-phase reaction between F₂ and BF₃ to produce BF₅, and the separation of Mo and V from UF₄. Fluidization: Corrosion tests were made during the reduction and hydrofluorination of several ore concentrates. Installation of the green salt pilot plant was nearly completed. Fabrication of the fluidized-bed calciner for reactor fuel processing wastes was completed. Reactor Chemistry: The absolute fission yield of Cs¹³⁷ has been determined for fast neutron fission of U²³⁵ and Pu²³⁹. Samples of U²³⁵, Pu²³⁹, and U²³⁸ have been exposed in various parts of EBR-1 in order to determine the variation of the ratio of capture-to-fission cross sections, with reactor position. The study of oxidation characteristics of U, Th, Zr, and Pu has been continued. Chemical-Metallurgical Separation Processes: The corrosion of alumina, thorium, and beryllia is discussed. The extraction of Pu from hydride-produced 1% Pu-U powder by molten Mg is presented. Data have been obtained for the partition of Ru between liquid Mg and liquid U-Fe alloy. Testing of pilot plant equipment components was continued. Analytical Research and Services: An x-ray spectrometric method is being developed for rapid determination of Pu in concentration range 3 to 9 × 10⁻³ M. (For preceding period see ANL-5602.) (W.L.H.)

9780 ANL-4811

Argonne National Lab., Lemont, Ill.

FISSION RATE MEASUREMENTS IN THE EXPERIMENTAL BREEDER REACTOR (CP-4). Samuel Untermyer. Apr. 18, 1952. Decl. Feb. 12, 1957. 14p. Contract W-31-109-Eng-38. \$0.25(OTS).

Fission rates of U²³⁵, U²³⁸ and Th²³² were determined in CP-4 using a fission ion current chamber. Typical values obtained are given. These values are in good agreement with measurements at Los Alamos on their fast reactor, and on a fast, tamped critical experiment. This suggests that the amount of extraneous material in CP-4 was not sufficient to degrade the spectrum. It was calculated that the fissions observed in U²³⁸ would contribute 0.28 to the breeding gain in the reactor. (auth)

9782 ANL-4974

Argonne National Lab., Lemont, Ill.

SUMMARY OF RADIATION MEASUREMENTS RELATING TO THE EXPERIMENTAL BREEDER REACTOR SHIELD. J. W. Butler. Feb. 1, 1953. Decl. Feb. 12, 1957. 16p. Contract W-31-109-eng-38. \$0.25(OTS).

A set of selected physical measurements made on the completed Experimental Breeder Reactor (EBR) is presented; the criterion for inclusion was applicability to analysis of shield performance. The information consists principally of fission rate measurements in and near the core and a few photon dose rate observations at the outside of the shield. Also included is a brief statement of plans for future shielding measurements. (auth)

9784 ANL-5095

Argonne National Lab., Lemont, Ill.

DETERMINATION OF THE CONVERSION RATIO OF THE EXPERIMENTAL BREEDER REACTOR BY RADIOCHEMICAL METHODS. "PRODUCTION AND CONSUMPTION PATTERNS IN THE EBR." M. Levenson. Dec. 23, 1953. Decl. Feb. 12, 1957. 63p. Contract W-31-109-Eng-38. \$0.40(OTS).

An evaluation of breeding in the EBR by chemical methods is described. The analytical methods and results for determining Pu production patterns and U consumption patterns involved the accurate measurement of neutron spectra, cross sections for fission and capture, and resulting values of α for U²³⁵ and Pu²³⁹. These procedures are outlined in detail. A net conversion ratio of 1.00 was obtained. Of the total Pu produced, 63.9% occurred in the outer blanket and 35.7% in the inner blanket. (K.S.)

9786 ANL-5222

Argonne National Lab., Lemont, Ill.

A PHYSICAL DETERMINATION OF THE CONVERSION RATIO OF THE EXPERIMENTAL BREEDER REACTOR. C. D. Curtis, S. H. Klein, D. Okrent, W. C. Redman, and S. Untermyer. Aug. 1954. Decl. Feb. 12, 1957. 114p. Contract W-31-109-eng-38. \$0.55(OTS).

Physical methods of measurement have been employed to determine the ratio of Pu²³⁹ production to U²³⁵ destruction in the Experimental Breeder Reactor (CP-4). Highly depleted U²³⁸ foils were irradiated throughout the core, inner blanket and outer blanket (cup), and the U²³⁸ neutron capture rates were determined by scintillation counting of the plutonium K α_1 and K α_2 x rays which accompany the decay of Np²³⁹. The observed activities required corrections for contributions by fission products and the decay of U²³⁷ formed by an (n,2n) threshold reaction of U²³⁸. A U²³⁵-coated fission chamber of the gamma-ray compensated type was used to establish the fission patterns in the core and inner blanket. These measurements were supplemented by the irradiation and counting of copper and U²³⁵ foils placed in those portions of the outer blanket which were inaccessible to the current chamber. The observations for Pu²³⁹ production and U²³⁵ fission were normalized by a comparison in the EBR thermal column of the response of the detectors to a thermal neutron flux for which pertinent

cross sections are known. When combined with values for the ratio of captures to fissions in U^{235} , as determined by mass spectrometric analysis of fuel from the EBR, the measurements yielded a value for the conversion ratio which is in excellent agreement with that determined by radiochemical procedures. It was found that 91% of the U^{235} destruction occurs in the core, 3% in the inner blanket, and 6% in the outer blanket. One-half per cent of the Pu^{239} produced is found in the core, 33-1/2% in the inner blanket and 66% in the outer blanket. For the movable outer blanket in its position of maximum effectiveness, the physically determined conversion ratio is 1.01 ± 0.05 . Estimates based on these measurements indicate that a conversion ratio of 1.3 is possible for a fast converter of the EBR type. (auth)

9788 ANL-LB-SL-1053

Argonne National Lab., Lemont, Ill.
EFFECT OF PROCESSING LOSSES ON BREEDING GAIN AND DOUBLING (CONVERTING) TIME FOR EBR-II AND PBR REACTORS. I. G. Dillon. May 8, 1956. Decl. Mar. 9, 1957. 4p. Contract [W-31-109-eng-38]. \$1.80(ph OTS); \$1.80(mf OTS).

10276 APDA-107

Atomic Power Development Associates, Inc., Detroit.
DESCRIPTION AND SOME SAFEGUARD ASPECTS OF PROPOSED DEVELOPMENTAL FAST NEUTRON BREEDER REACTOR. July 25, 1955. Decl. Jan. 7, 1957. 122p.

The subject reactor is a U^{235} fueled, Na cooled, power breeder designed for a heat output of 300 Mw. Design parameters and descriptions are given. Two types of fuel elements, pins and plates, are evaluated. Reactor control, fuel handling and cycling, nuclear and physical properties of materials, and site location are discussed. (D.E.B.)

10280 CF-57-4-92

Oak Ridge National Lab., Tenn.
MAINTENANCE OF VARIOUS REACTOR TYPES. B. D. Draper. Apr. 8, 1957. 28p. Contract [W-7405-eng-26]. \$4.80(ph OTS); \$2.70(mf OTS).

Maintenance of reactor power plants is more complex than conventional fossil fired steam turbo-generator plants because of the presence of radioactivity either localized in the reactor vessel or scattered throughout the circulating loops depending upon the reactor type. It is obvious that the easiest systems to operate and maintain are those which permit the use of conventional tools and techniques on a large portion of the reactor auxiliaries. Following is a list of reactor power plant types, arranged in order of increasing difficulty of maintenance in the opinion of the author with advantages and disadvantages maintenance-wise of each. (auth)

10306

THE CHEMICAL GROUP AT DOUNREAY. Atomics 8, 255-7(1957) July.

10750

DESIGN AND COST ESTIMATE FOR A PYROMETALLURGICAL REPROCESSING PLANT. Louis Basel (Atomic Power Development Associates, Inc., Detroit, Mich.) and Joseph Koslov (Vitro Corp. of America, New York). Nucleonics 15, No. 8, 56-60(1957) Aug.

A conceptual design is presented of a pyrometallurgical reprocessing plant for reprocessing enriched fuel of the Enrico Fermi fast breeder reactors. The design presented uses the oxide-slagging method, and can handle 18,000 kg/yr of enriched U alloy at an estimated \$79/kg. The highly radioactive product requires remote fabrication. (C.H.)

10921 ANL-4238

Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION REPORT FOR THE PERIOD MARCH 1, 1948 THRU NOVEMBER 30, 1948. Feb. 3, 1949. Decl. Mar. 28, 1957. 67p. Contract W-31-109-eng-38. \$10.80(ph OTS); \$3.90(mf OTS).

Studies reported on equipment for circulating NaK in Experimental Breeder Reactor development include considerable information on wear of bearings in NaK. Blanket blocks of natural U provided with press-fitted finned cooling tubes have been fabricated and jacketed in deep-drawn Ni cans. Which have been welded shut and welded to the cooling tubes to form a gas tight envelope for the blocks. The steps in fabricating the cans and jacketing the blocks are described. A mock-up of a section of the blanket cup formed from the jacketed blocks has been thermal cycled to simulated operating conditions. The blanket cup safety device mock-up and safety rod mechanism are illustrated. One voltage drop method for detecting voids and meniscus in the NaK-filled fuel rods has been developed. Operating characteristics of electromagnetic pumps and a homopolar generator are reported. A NaK level indicator is described briefly. (G.Y.)

10922 ANL-5132(Del.)

Argonne National Lab., Lemont, Ill.
THE MANUFACTURE OF INTERNAL BLANKET AND FUEL BLANKET SLUGS FOR THE EXPERIMENTAL BREEDER REACTOR. R. E. Macherey and S. T. Ziegler. Dec. 1953. Decl. with deletions Feb. 12, 1957. 61p. Contract W-31-109-eng-38. \$10.80(ph OTS); \$3.90(mf OTS).

Manufacturing methods and procedures for fabricating EBR internal and fuel blanket elements are presented in detail. (C.H.)

10923 ANL-5297

Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] APRIL 1, 1954 THROUGH JUNE 30, 1954. July 15, 1954. Decl. Mar. 28, 1957. 206p. Contract W-31-109-Eng-38. \$33.30(ph OTS); \$9.60(mf OTS).

Boiling Experimental Reactor. Revised core and fuel element designs have been established as a result of mockup tests on the removal of heat by natural air circulation at shutdown. A flow sheet for the BER is presented. Power Breeder Reactor. Major work on the EBR-II has been devoted to the design of a working model. Several plan views and perspective drawings are shown. Small Power Reactors. Further design factors for a small horizontal boiling power reactor are discussed. The reactor is primarily designed for operation in remote arctic regions at a power level of $5\frac{1}{2}$ Mw. Design Research and Development. A theoretical analysis was made of the effect of Pu^{239} produced in the BER on the reactor temperature coefficient. For the particular BER considered, it appears that the coefficient may become less negative as Pu^{239} builds up. It is also possible that the cross section temperature coefficient may be positive at reactor operating temperatures. Thermal and fast neutron fluxes were calculated for a given 20 Mw BER core, together with the γ flux. The values obtained for the thermal neutron and γ flux were used to determine the radioinduced heating in the Fe pressure vessel. The accuracy of modified diffusion theory (removal theory) for calculating the photon source distribution from neutron captures in the EBR shield was determined by comparison with measured values. Discrepancies noted are attributed to variations in the U^{238} fission rates and the use of point source geometry. New data have been obtained from a Monte Carlo calculation of neutron transmission and reflection by slabs of H_2O . The time dependent response of a reactor due to localized reactivity variations was treated for the case of low time rate of change where the response is measured some distance from the localized change. Amplitude responses and phase

shifts were made as a function of driving function frequency for a series of short lifetimes. At very low frequencies, the amplitude response is not equally sensitive to all delay groups. The kinetics of a supercritical reactor with no sources is analyzed, and values of Σ , k_{ex} , and λ were calculated as a function of τ . Heat Transfer. A modified electrical-geometrical analogue has been designed to solve temperature distribution problems within fuel elements of complex geometry. Good accuracy for the device is reported. Factors affecting the precision of a given thermal analysis of EBR-II fuel elements are discussed, including uncertainty factors which are helpful in determining the worst possible effects of a number of thermal parameters. A heat transfer analysis was made for 5 different EBR-II fuel element designs. Three radiator-type and 2 pin-type elements were tested, and their comparative performance is summarized. Natural circulation tests were made to determine the steady-state relationship between maximum plate temperature and power density of shutdown heating. It is concluded that the reactor core will not be adequately cooled during shutdown when there is a loss of water from the core area. A pulse-heating test facility was designed to investigate the fundamentals of water expulsion from a Cu tube to which short pulses of a-c current are applied. Fuel Elements. The feasibility of fabricating the reference type EBR-II fuel element was investigated. This design consists of a hexagonal array of fuel tubes which consist of in-line fuel and blanket sections. An experimental ball tower apparatus was constructed for the fabrication of ball-type fuel elements. The results of 4 shotting runs are given. Data are collected on the water-corrosion of unclad U-Nb, U-Si, and U-Nb-Zr alloys at high temperatures. Other corrosion studies include corrosion-irradiation tests on Zircaloy-2-clad U fuel plates, the corrosion of Zircaloy-2 by superheated steam and moist and dry air, and the corrosion characteristics of plates with deliberate defects introduced in the clad. Structure Materials and Compound Tests. Four reactor pressure vessels, fabricated from Carilloy T-1 alloy steel, were tested to destruction by impact testing. Highly satisfactory results were achieved. Further tests are reported on the magnetic jack control rod drive unit described in the previous report. A preliminary investigation is under way to develop a void detector for boiling water systems. Most promising methods involve the absorption of γ or x rays, the dielectric constant of the steam-water mixture, the detection of boiling noise, and the transmission or scattering of visible light. Two types of liquid level indicators were examined for use in pressurized closed reactor systems. One instrument consists essentially of a resistance bridge circuit, whereas the second depends on the changing capacitance of an insulated probe immersed in the water tank. Coolants. The dynamic Na corrosion loop was operated at 400 and 500°C to determine the oxygen content of the Na as an oxide. For operating times of 8 to 55 hr, the content was found to vary between 0.001 and 0.0061 wt. % oxygen. Specific damage to diphenyl by fast neutron flux irradiation was studied. Design and Evaluation. Design details of the BER pressure vessel and control rod drive mechanisms for the BER and EBR-II are presented. (K.S.)

11348 BNL-433

Brookhaven National Lab., Upton, N. Y.
PROCEEDINGS OF THE BROOKHAVEN CONFERENCE
ON RESONANCE ABSORPTION OF NEUTRONS IN
NUCLEAR REACTORS, SEPTEMBER 24 TO 25, 1956
July 1957 135p \$0.65(OTS)

11742 ANL-4554

Argonne National Lab., Lemont, Ill.
EXPERIMENTAL BREEDER REACTOR PROJECT. RE-
PORT FOR THE PERIOD FEBRUARY 1, 1950, THROUGH
MARCH 31, 1951. R. A. Cameron, B. C. Cerutti, L. J.

Koch, H. V. Lichtenberger, D. F. McGinnis, M. Novick,
E. N. Pettitt, H. E. Stanton, E. F. Stone, G. K. Whitham.
Apr. 1, 1951. Decl. Jan. 17, 1957. 60p. Contract W-31-
109-Eng-38. \$9.30(ph OTS); \$3.60(mf OTS).

The design of a pressure gauge system to show pressure changes in the fuel rods throughout the Experimental Breeder Reactor (CP-4) is described. Apparatus has been designed and built for filling the annulus in the lower section of the CP-4 fuel rod with NaK alloy. The procedure for adding NaK to fuel rods is described. Photographs are shown of the Reactor Inner Assembly and of its various parts. Safety and control rods and their drive mechanisms have been constructed and are shown. Operation of the safety plug is described. Velocity, acceleration, and time vs. distance curves of the plug fall are included. A detailed drawing is shown of a die for forming the external ribs on the CP-4 fuel rod jackets. The blanket bricks have been fabricated, autoclaved, and assembled on the lower shield section of the CP-4. The design of a totally enclosed liquid metal pump for the cooling circuit of the CP-4 is described in some detail. Results are given of a test of this pump in a NaK alloy coolant loop to determine its hydraulic characteristics. The design and operation of pressure transmitter unit for the reactor coolant system are described. Tests have been carried out to determine the heat transfer characteristics of a full size steam generator tube assembly under full power conditions. Design details and construction specifications of the tube are shown. Characteristics are shown of a gas circulating blower tested operating on air. (B.J.H.)

11743 ANL-5023

Argonne National Lab., Lemont, Ill.
EXPERIMENTAL BREEDER REACTOR PROGRESS RE-
PORT FOR APRIL 1, 1951 THROUGH JANUARY 31, 1953.
H. V. Lichtenberger, M. Novick, B. C. Cerutti, R. A.
Cameron, D. F. McGinnis, E. N. Pettitt, G. K. Whitham,
R. A. Haroldsen, and L. J. Koch. Feb. 20, 1953. Decl.
Jan. 7, 1957. 46p. Contract W-31-109-eng-38. \$0.45
(OTS).

The following EBR studies are reported: criticality studies, calibration of the control system; neutron flux measurements; fission-rate measurements; over-all temperature coefficient measurements; determination of the sign and magnitude of the metal temperature coefficient; measurement of power distribution in the core and in the internal and external blankets; over-all performance of the reactor cooling system; reactivity losses resulting from thermal expansion of slug elements; measurement of γ intensities around the reactor shielding; and disassembly of fuel rods. The shape of the flux for fission neutrons was determined by measuring the radioactivity of fuel rods at various positions along the rod and using rods from various reactor positions. U^{235} , normal U, and steel were evaluated when placed in the enriched fuel assembly of the reactor (L M T)

11744 ANL-5313

Argonne National Lab., Lemont, Ill.
QUARTERLY PROGRESS REPORT ON REACTOR DEVELOP-
MENT. (4000 PROGRAM). July 31, 1954. Decl.
Mar. 29, 1957. 42p. Contract W-31-109-eng-38. \$7.80
(ph OTS), \$3.30(mf OTS).

Present plans for the Boiling Experimental Reactor (BER) are given. The reactor, auxiliaries, and power plant will be housed in a steel shell designed to maintain tightness against an internal pressure of 15 psig. The reactor pressure vessel is designed to operate at an internal pressure of up to 600 psig. Related experiments with Borax-I provided test information on a Cd fuse element for protection against rapid reactor transients. Such a power surge causes the heating of a U plug with melting of solder, releasing a Cd sleeve into the reactor. After completion of the fuse tests,

a stronger control rod was inserted in the reactor and a short period excursion was initiated. The control rod was worth $4.1\%k_{\text{eff}}$. The resulting surge produced by rapid ejection of the control rod resulted in partial or total destruction of major reactor components. Instrumental data have not been fully analyzed from this incident, but a preliminary estimate of the nuclear energy released is in the region of 120 Mw-sec. The development of U alloys for fuel elements in the BER has been directed to the constitution, fabrication, corrosion, and heat treatment of Zr and Nb-Zr alloys. Fabrication techniques are being developed for Zircaloy-II-clad, natural-U plates. The design study of a small heat and power reactor for installation at remote locations presently calls for a boiling H_2O -moderated reactor operating at 5.5 Mw and a steam pressure of 600 psig. The reactor is located in a horizontal tank, using MTR-type U-Zr fuel plates. Current effort in the breeder reactor program is being directed to a working scale-model of the EBR-II. A general arrangement of the model is presented. Exponential studies on the neutron distributions within enriched assemblies containing U^{235} , U^{238} , Al, and Fe were carried out. A procedure for determining the U^{235} : U^{238} fission ratio was worked out using a 96% enriched and a natural-U miniature fission chamber. Results obtained from Li^6 speck-loaded nuclear emulsions exposed to monoenergetic neutrons from a Van de Graaff generator indicate that the energy resolution of this technique is about 200 kev at 270, 400, and 600 kev. Forthcoming experiments on the EBR-I Pu fuel charge will employ Pu-Al fuel alloy. Tentative values of hardness and coefficient of expansion were determined for this material. Preliminary values are reported on the variation of η of natural-U produced by irradiation in the NRX reactor. Further research is described on the recovery of U from U-Zr and U-Al alloys by fluoride volatility processes. Difficulties with reacting the bulk constituent with BrF_3 have been encountered, including U ignition under given conditions. A smooth and complete dissolution of U was affected by immersing the metal in HCl and bubbling Cl_2 through the solution. The reaction is catalyzed by 0.01 M FeCl_3 or 0.01 M HI. (K.S.)

11745 ANL-5371

Argonne National Lab., Lemont, Ill.

REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] OCTOBER 1, 1954 THROUGH DECEMBER 31, 1954. Jan. 15, 1955. Decl. Jan. 17, 1957. 284p. Contract W-31-109-eng-38. \$1.60(OTS).

Physics. Two-group UNIVAC calculations were made for the BER in order to investigate the safety and operating characteristics of various core designs. Results indicate that considerable control over critical mass, power distribution, and to a lesser extent void and temperature coefficients, can be achieved by suitable arrangement of enriched- and natural-U fuel elements. Void coefficients can be made more negative by reducing the H_2O :U ratio. In all designs considered, the negative reactivity void coefficient of the cold reactor depends strongly on the increased leakage. Good results are obtained in calculating the reactivity held in steam for the operating boiling reactor with uniform fuel distribution, by assuming the steam to be uniformly distributed. Some reduction in reactivity held in the steam in a "flattened" reactor may be produced by reducing the steam voids in the outermost fuel elements. Other calculations are also reported for the EBR-II and the PBR on the following parameters: longterm changes, temperature coefficients of reactivity, control effectiveness, blanket power flattening, and power flattening by radial variation of enrichment. An iteration technique for calculating mutual interactions between steam voids and thermal neutron flux was studied for a uniformly loaded U^{235} , H_2O boiling reactor design. A control-rod analysis of various materials was undertaken to determine material thicknesses equivalent to a black Cd rod, and to establish

burnup allowances for about 10-yr continuous operation. The materials studied were Cd, Hf, boron, B^{10} , B_4C , CaB_6 , and ferro-boron. Burnup data were obtained for some of the Hf isotopes with appreciable neutron absorption cross sections. An extensive study of AVIDAC reactivity and kinetic calculations are presented. The use of diphenyl as a reactor coolant and moderator was studied at different power levels and moderator ratios, using stainless steel as a structural material. These reactor types were investigated for use with control-rod override of maximum Xe and burnup, and borated fuel elements and temperature override of maximum Xe. Temperature override was effective for maximum Xe in these reactors. Experimental work on bucklings, reflector savings, and U^{235} : U^{238} fission ratios in the Fast Exponential Experiment are being completed, and a final report on this project will appear as ANL-5379. A method for measuring the ratio of fission cross sections of two elements in a given flux was developed which is independent of the absolute sensitivity of the equipment. The following values were obtained in the center of the EBR: $\text{U}^{238}/\text{U}^{235} - 0.115$, $\text{Th}/\text{U}^{235} - 0.0305$, and $\text{Pu}/\text{U}^{235} - 1.42$. Design of a very small fission counter for pile measurements is described. Heat Transfer. The expected steady-state nuclear heating of the BER pressure vessel was calculated, relating pressure vessel thickness, heat generation, temperature distribution, thermal stress, and pressure stress. Stresses in a carbon steel vessel without thermal shielding need not be excessive, but 304 and 347 stainless steel are not as desirable. Multi-channel natural circulation boiling density tests at 600 psia, and single-channel forced circulation tests at 260 psia are reported. The apparent thermal conductivity of steel shot in NaK eutectic was measured in an apparatus described in ANL-5134. The experiment is designed to simulate the conductivity of U in liquid Na. Fuel Elements. An alternate fuel element design for EBR-II consists of a radiator-type element in which the coolant flows through channels in a stack of fuel wafers. An experimental procedure for electroplating U with Ni and Ag is given. Unclad high-U alloy samples (U-Si, U-Nb-Zr, and U-Nb) were corrosion tested in 500 and 550°F water. Additional corrosion testing of Ni-plated 2S Al was continued in 600°F water. Weight change data of thorium-uranium bodies in 600°F water were obtained. A test facility is described for the irradiation of EBR-II pin-type fuel elements. A central thimble of the CP-5 reactor is used, and air cooling of the irradiated element is provided. One EBR-II pin-type fuel element (7 U pieces in a 304 stainless steel tube) was thermal cycled 200 times from 300 to 100°F in NaK. The U showed an average length gain of 2%. Structural Materials and Component Engineering Tests. A full-scale test section of a BER-type core was constructed, and air-cold water tests were made to determine the flow characteristics in the top reflector and downcomer. Water pressure-drop tests were made in a modified EBR-II pin-type subassembly. A reduction in pressure drop of 16 psi was effected by a new grid structure and fuel element orientation. A 4-in. Na loop with a d-c electromagnetic pump driven by a homopolar generator is described. The loop will be used to test thin-walled stainless or Nichrome tubing with fast-flowing Na up to 500°C. Certain changes in the control-rod layout of the BER are outlined. A test facility has been built to test the reference BER rod drive mechanism, which is also described. A number of alternate control rod drive mechanisms are evaluated. An apparatus is presented for developing Na-vapor rotary seals, and to test mechanical components, gears, bearings, and couplings immersed in Na vapor or liquid Na. Two newly-developed seal systems are discussed. Coolants. A dynamic corrosion loop was constructed to study the effect of circulating liquid Na(30 gpm) on irradiated and non-irradiated normal U. The content of Na_2O was studied as a function of time. Oxide content rose to a maximum, and dropped off. The maximum

occurred sooner at higher operating temperatures. Two irradiated normal U pins were corrosion tested. Design and Evaluation. A detailed design study (No. 1) of the EBR-II is presented. The analysis includes reactor arrangement, shielding, fuel and blanket assembly designs, neutron and power distribution, Pu distribution, and a thermal analysis. (For preceding report in series see ANL 5345) (K S)

11746 ANL-5471

Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] APRIL 1, 1955 THROUGH JUNE 30, 1955
SECTION II Sept 1955 Decl Jan. 17, 1957 173p Contract W-31-109-eng-38 \$110(OTS)

Development of the thermal power reactor program is summarized. A complete conceptual design study is given for a nuclear power plant for 100-kw electric and 400-kw heat capacity. Calculations were made to evaluate the inherent safety of 20-Mw D_2O -cooled and moderated EBWR-type boiling reactors. The effects of rapid and slow, but large, increases in reactivity are summarized. Methods of calculating reactivity variation as a function of total energy production for enriched fuel, thermal reactors are discussed, and the methods were applied to two reactors. Results of the analysis are given in graphical form. Results of measuring fission cross section ratios in the EBR are also summarized. A brief account is given of neutron flux measurements using Np^{237} foils and of the foil preparation technique. Heat transfer tests include studies of the multi-channel natural circulation boiling density at 600 psig, a discussion of the representation of average voids in boiling length with measurable parameters, normalized plots of void distribution along the boiling length, local boiling pressure drop in round and rectangular vertical channels, and burnout heat flux for wires in crossflow at atmospheric pressure. Fabrication and corrosion test history of EBWR fuel plates are tabulated. Zircaloy-2 specimens which had been heliarc spot welded are shown. The EBR-II core subassembly is shown. Designs are given for a Na-bond filling device for EBR-II pin-type fuel elements. Further attempts were made to produce a protective coating for uranium. The corrosion of high-U alloys in 500 and 550°F H_2O and the corrosion-irradiation of unclad high-U alloy pins are summarized. Water solubility tests on urania-thoria bodies were made. Corrosion data for nickel-plated 2S Al in 600°F H_2O are given, as are data on the dynamic H_2O corrosion of Al-Ni alloys. Static corrosion of anodized 2S Al in H_2O is also reported. Results of experiments of the fission product contamination of NaK bond in contact with U during irradiation are reported. The results of examination of the in-reactor tube tip from the Argonne Water Loop at the MTR are given. Water pressure drop tests of EBR-II core subassemblies were made. Heat capacity characteristics for the EBR-II working model electromagnetic pump are given. The primary EBR-II Na system is shown. Summaries of packing gland tests for liquid metal service are given. A study was made to estimate the rate of water decomposition in the EBWR. Corrosion inhibition studies have been continued. A dynamic cold trap liquid metal purification facility was installed in the EBR-II working model, and the initial results are reported. Schematics are given for the Na transfer system for the EBR-II working model. A manually operated electronic Na level probe is also shown. (For preceding period see ANL-5345) (B J H)

11747 ANL-5561

Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] OCTOBER, NOVEMBER, DECEMBER 1955. SECTION I. Apr. 1956. Decl. Jan. 17, 1957. 169p. Contract W-31-109-eng-38. \$110(OTS).

Experimental Boiling Water Reactor. It was decided to use several types of fuel elements in order to achieve flexibility in the core loading. The structure of these fuel elements is tabulated. Using phenomenological methods, lattice constants, reactivities, and neutron economy of the reference EBWR were calculated, and results are tabulated. Neutron flux distributions in the core are shown. An analytical prediction was made of the natural circulation performance of EBWR at 600 psig. Calculations were also made to determine whether simple changes in the riser length and/or geometry would favorably affect EBWR performance. Complete results are given in graphical form. Design-economics analyses of EBWR fueling are summarized. The boiling water corrosion of Zircaloy-2 is shown. The EBWR control rod scram-time curve at 600 psig is shown. The thermocouples to be used for measurement of vessel temperatures are shown. The final designs for the pressure vessel shock springs are shown, and a load-deflection diagram is given. The gamma-ray flux and corresponding heat generation in the D_2O -moderated EBWR are listed. The status of EBWR construction is discussed. Data are included on the leakage-rate tests of the EBWR steel shell and on compression tests on specimen concrete shielding. Supporting and Alternate Design Research and Development. Boiling heat transfer studies were made. Riser studies indicate that riser height is not too important a parameter in influencing the ratio of velocities of steam and liquid. Data are given on the pressure drop resulting from 2-phase flow. The transient behavior of the multi-channel natural circulation loop is discussed. Single-channel unrestricted and restricted natural circulation boiling studies are summarized in detail. Data showing the relationship between power density, voids, quality, recirculation flow, pressure and geometry for boiling systems are given in both tabular and graphical form. Pressure drop measurements are included, and an analytical method of calculating natural circulation performance is outlined. The sources of error in steam void measurements are evaluated. A heating power source, which is to be used to determine experimentally the physical response of reactor components during a power excursion, is described. Corrosion tests on Zircaloy 2-clad U-Zr plates in 550°F degassed H_2O are summarized. Corrosion test data are also given for high-U alloys in 500°F H_2O . The high-temperature irradiation corrosion of Al-Ni alloys is discussed. Corrosion-irradiation tests were also made on Al-Ni alloy plates. Progress is also reported on solution-irradiation and thermal cycling tests of urania-thoria bodies. A model of the rare earth oxide powder-filled control rod was thermal cycled and exposed to H_2O at 600 psig and 486°F in the EBWR control drive test facility. The mechanical properties of B-contained stainless steel alloys were listed. These alloys are being considered for use as the thermal shield for the EBWR pressure vessel. The corrosion characteristics of boron-stainless steel are also listed. Schematic diagrams are given for a hydraulically operated control rod drive and position indicator. A scram-time curve is given for the hydraulically operated control rod. Diagrams are also given for a magnetic jack control rod drive. Experimental Breeder Reactor-I. Danger coefficient experiments on EBR-I are summarized. Experimental procedures are outlined, and results are tabulated. Experimental Breeder Reactor-II. CP-5 irradiation tests on EBR-II pin type fuel elements are summarized, and photographs of the irradiated fuel elements are shown. An electrically heated model of an EBR-II fuel subassembly was constructed to determine the cooling requirements to remove the decay heat after irradiation. It was determined that a flow of 15 scfm of A gas at room temperature at a pressure of about 3 in. H_2O will maintain a maximum temperature below 1100°F. The activation of Na coolant by a fuel element with a hole in the end closure is discussed. A loop was constructed to test the release of such activity, and it was

determined that in the EBR-II, 0.265 curies of activity would be released to the coolant Na every three months. The possible eutectic point formation of U with Fe in the stainless steel fuel tube was studied. Test conditions and results are listed. Diagrams are given of the EBR-II control rod mechanism. Calculated axial and radial γ and neutron fluxes in the EBR-II are shown. EBR-II Working Model. Design status is described. Supporting and Alternate Design Research and Development. Thermal cycling tests of Mo-U and Zr-U alloys are described. The development of electromagnetic pumps for the Na test loop is briefly discussed. (For preceding period see ANL-5511.) (B.J.H.)

11748 ANL-5571

Argonne National Lab., Lemont, Ill
REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] JANUARY, FEBRUARY, MARCH 1956.
SECTION I. July 1956. Decl. Mar. 28, 1957. 174p.
Contract W-31-109-eng-38. \$1 10(OTS)

Advanced Water Reactor Program Flux distribution studies associated with the use of various combinations of natural and enriched U fuel plates in EBWR are reported. Fuel plate dimensional changes necessitated the calculation of revised thermal data. These new data are given along with the assumptions made in the calculations. The design of fuel assemblies, control rod channel closures, and the grid plate handling assembly are shown. Of the 9 EBWR control rods, 4 will be fabricated from a B-stainless steel system and 5 from Hf. The nuclear and economic considerations of such a rod arrangement are reviewed. The reactor shield and insulation is described and discussed. Reactor loading and unloading, fuel element storage, and the design and operation of the fuel transfer coffin and carriage are considered. A method of control rod removal without draining the pressure vessel is described. The status of fuel element corrosion tests and reactor component construction and fabrication is summarized. Boiling heat transfer studies associated with the EBWR were continued. Natural circulation boiling in single and multiple channels at 600 psig was investigated. Phase velocities, void effects, and friction factors are plotted. An analysis of the electrical characteristics of the power supplies used in these studies is also presented. A portion of the data gathered from Borax III operation is presented. Fission product activity buildup and decay in H_2O and steam, H_2O decomposition, and reactivity changes due to fission products and burnup were analyzed. Additional high-temperature H_2O corrosion data on U and Al alloys are reported for various pH values, H_2 concentrations, and flow velocities. Corrosion and irradiation data on ThO_2-UO_2 type fuel elements are also reported. Studies of the corrosion resistance, weldability, and machining properties of B-stainless steel systems were continued. Corrosion study results for other control materials and structural materials are reported. Fast Power Breeder Reactor Program. Two-dimensional calculations are reviewed which indicate minor changes in reactor component composition. The decay heat characteristics of EBR-II were determined, and a natural convection heat removal system was designed. The heat transfer characteristics of the bayonet cooler used in the system were investigated. In connection with the EBR-II shielding program, the fast neutron constants for Na and graphite were calculated. Results of thermal cycling tests on fission alloys are reported. Liquid metal heat transfer, fused fluoride corrosive effects, and liquid metal pump performance were also studied. (For preceding period see ANL-5561.) (D.E.B.)

11797 KAPL-M-RE-8

Knolls Atomic Power Lab., Schenectady, N. Y.
FURTHER INVESTIGATIONS OF A FAST REACTOR. R. Ehrlich. Mar. 23, 1950. Decl. Jan. 17, 1957. 21p. Contract W-31-109-eng-52. \$3.30(ph OTS), \$2.40(mf OTS)

Two more multi-group calculations, fast reactors 4 and 5, were made. The largest change from the earlier calculations is to reduce the Na to U atomic ratio in the core from 6.4 to 1 to 3.5 to 1. The latter figure is in the neighborhood of the lower limit that the engineers felt they could attain with a pin type fuel structure. FR-4 uses the MgO reflector and FR-5 the Be reflector of the earlier calculations. In addition, the Ni structural material in the core is replaced by Fe and pin void is allowed for. Ni has an abnormally high scattering and hence slowing down cross section at lower energies. Since it is highly unlikely that Ni will be used in the core, its use gives an overestimate of the slowing down to be expected from the structural material. Furthermore, the inelastic scattering is badly treated in fast reactors 1 through 4. That of the structural material was neglected, and that of the fuel in the core was taken too big. The two tend to cancel out to a large extent. The inelastic scattering in the blanket U was taken from LA-140. These values are probably too high. In FR-5 we attempted to treat these cross sections more reasonably and added a further high energy group from 10 to 3.68 Mev ($u = 0$ to $u = 1$). A fission source was assumed in the natural U blanket for FR-5 to get a better idea of the properties of such a blanket. As in the earlier calculations, no allowance was made for burnup, fission product poisoning, or thermal expansion. (auth)

11807 TID 10097

Dow Chemical-Detroit Edison and Associates. Atomic Power Development Project, Detroit.
INFORMATION REPORT ON THE PROJECT ACTIVITIES FOR THE PERIOD DECEMBER 1, 1953 TO JULY 31, 1954. Sept. 1, 1954. Decl. Jan. 7, 1957. 78p. \$0.55(OTS).

The organization and current budget of the Atomic Power Development Project are briefly summarized. Fuel element design and fabrication studies are reported. General corrosion tests are being conducted on a variety of materials for use in liquid metal systems. Gloveiron and Zr show promise as reactor core cladding materials. An evaluation of various mechanical components of liquid metal and steam power systems is being carried out. Studies of reactor control methods have been emphasized. The most promising concept appears to be a control system in which a fast neutron reflector (iron) is exchanged for a fast neutron poison (Li^6) in the reflector region. Temperature coefficients of reactivity have been calculated for the reference design reactor. A conceptual design of the mechanical handling system for fuel and blanket elements in the reference design is shown. Aqueous and nonaqueous methods of fuel processing are being studied and evaluated. Studies of reactor economics, safety, and alternate reactor designs are reported. (M.P.G.)

12210 AECD-4244

Los Alamos Scientific Lab., N. Mex.
NEUTRON SPECTRUM FROM FAST REACTOR. Louis Rosen. [194?] Decl. Aug. 20, 1957. 7p. Contract [W-7405 eng-36]. \$1.80(ph OTS), \$1.80(mf OTS).

A vacuum camera technique was used to determine the neutron spectrum emerging from a fast reactor in the energy range of 0.4 to 15 Mev as detected through a boron steel collimator. (R.V.J.)

12248

DOUNREAY'S FAST REACTOR. Atomics 8, 336-8, 342 (1957) Sept.

The Dounreay Fast Breeder Reactor is discussed. Topics included are the heat cycle, shielding, core and blanket, heat exchanger, charge and discharge, control and instrumentation, power and auxiliaries, and the sphere. (M.H.R.)

12556 KAPL-M-EAL-16

Knolls Atomic Power Lab., Schenectady, N. Y.
FUEL ASSEMBLY FOR OXIDE-FUELED PLUTONIUM

POWER BREEDER. E. A. Luebke. Dec. 17, 1954. Decl. Mar. 29, 1957. 9p. Contract W-31-109-Eng-52. \$1.80 (ph OTS); \$1.80 (mf OTS).

The use of $\frac{1}{8}$ - to $\frac{1}{4}$ -in. diameter fuel elements is suggested for the reactor, thus making possible the consideration of thin-walled tubular spacers between the fuel rods. The Na coolant would be allowed to flow through the separator tubes as well as the usual space between the fuel elements. The fuel rods and separator tubes are supported on one end from narrow support strips. (M.H.R.)

12570 LA-1575

Los Alamos Scientific Lab., N. Mex.

DISASSEMBLY OF THE LOS ALAMOS FAST REACTOR.

E. T. Jurney, G. P. Arnold, E. F. Corpieri, R. B. Leachman, E. H. Pierce, J. Montoya, W. D. Schafer, T. J. Thompson, M. Thorpe, and C. W. Zabel. July 1953. Decl. Apr. 1, 1957. 27p. Contract W-7405-eng-36. \$0.30 (OTS).

The methods and procedures for disassembling the Los Alamos Fast Reactor are described. Some information on the radiation damage of the internal reactor materials is included. (K.S.)

12591 TID-5352

Argonne National Lab., Lemont, Ill.

EBR-I OSCILLATIONS: INFINITIES IN THE TRANSFER FUNCTION WITH POSITIVE AND NEGATIVE THERMAL FEED-BACK FOR SINUSOIDAL POWER VARIATION.

A. J. Ulrich. May 18, 1956. Decl. Jan. 17, 1957. 4p. Contract [W-31-109-eng-38]. \$1.80 (ph OTS); \$1.80 (mf OTS).

12598 TID-10048

Los Alamos Scientific Lab., N. Mex.

THE LOS ALAMOS FAST PLUTONIUM REACTOR. E. T. Jurney, Jane H. Hall, David B. Hall, Avery M. Gage, Nat H. Godbold, Arthur R. Sayer, and Earl O. Swickard. May 1954. Decl. Mar. 13, 1957. 144p. Contract W-7405-eng-36. \$0.65 (OTS).

The Los Alamos Fast Plutonium Reactor is a low-power reactor built primarily to demonstrate the feasibility of a reactor operating on fast neutrons and to serve as an experimental facility. The operating power level of 25 kw produces a fast flux of approximately 4×10^{12} nvt. The reactive region consists of a lattice arrangement of metallic Pu fuel rods surrounded by normal U reflector material and cooled by flowing Hg. Experimental facilities consist of numerous fast neutron ports and a graphite thermal column. (auth)

12601 TID-10085

[Chicago. Univ.] Metallurgical Lab.

DESIGN OF FAST NEUTRON BREEDER TEST PILE.

W. H. Zinn. Jan. 25, 1946. Decl. Mar. 5, 1957. 32p. \$0.30 (OTS).

13005 KAPL-M-JKD-8

Knolls Atomic Power Lab., Schenectady, N. Y.

PROCESSING PLANT FOR THE KAPL FAST OXIDE BREEDER. J. K. Davidson. Mar. 7, 1955. Decl. Mar. 27, 1957. 14p. Contract W-31-109-Eng-52. \$3.30 (ph OTS); \$2.40 (mf OTS).

A processing plant capable of decontaminating the spent fuel from the core of the KAPL Fast Oxide Breeder has been conceptually designed. The process is based on an adaption of the standard Purex solvent extraction flowsheet, and although no detailed data are available, it is felt reasonably certain that such a flowsheet can be made to work. Emphasis has been placed on the preliminary design of an inexpensive physical plant, and the concept of disposal rather than maintenance of the hot equipment is used to

arrive at a plant design having a minimum of shielded volume. (auth)

13094 LWS-24898 (Del.)

California Research and Development Co., Livermore, Calif.

PLUTONIUM ALLOYS AND COMPOUNDS FOR SOLID FUEL OR CONTROL ELEMENTS IN FAST REACTORS.

R. M. Fulrath and D. R. Mash. July 21, 1953. Decl. with deletions Feb. 26, 1957. 6p. \$1.80 (ph OTS); \$1.80 (mf OTS).

Control of fast reactors is best accomplished by the use of shim rods. The properties required in shim rods for use in the FBR are reviewed, and available Pu compounds are considered for application. The use of a PuNi₂ alloy or PuSi₂ system is indicated. (D.E.B.)

13138 CF-55-6-157

Oak Ridge National Lab., Tenn.

BREEDING REACTORS. M. Tobias. June 7, 1955.

Decl. Feb. 23, 1957. 22p. Contract W-7405-eng-26. \$0.35 (OTS).

A general discussion is given on the terminology and theory of breeding. Sodium cooled fast reactors, such as the Experimental Breeder Reactor, and aqueous homogeneous reactors are discussed in more detail. (W.L.H.)

13165 LA-2052

Los Alamos Scientific Lab., N. Mex.

NONWEAPONS ACTIVITIES AT LOS ALAMOS SCIENTIFIC LABORATORY. PART III. POWER REACTOR EXPERIMENTS.

Samuel Glasstone. June 1956. Decl. May 9, 1957. 44p. Contract W-7405-eng-36. \$0.45 (OTS).

The research program at Los Alamos on homogeneous reactors, fast reactors, and the Los Alamos Power Reactor Experiment is summarized. (C.H.)

13167 LWS-22708 (Del.)

California Research and Development Co., Livermore, Calif.

CONCEPTUAL DESIGN OF A ONCE-THRU-CORE, SOLID FUEL FBR.

J. A. Quinville, T. P. Heckman, and R. E. Kelly. July 22, 1953. Decl. with deletions Feb. 27, 1957. 33p. \$6.30 (ph OTS); \$3.00 (mf OTS).

Preliminary design studies on a 500 Mw, Pu fueled, one-pass, high-burnup, power breeder reactor are reported. Nuclear properties, heat balance, fuel processing, and economics are analyzed. (D.E.B.)

13488 ANL-5782

Argonne National Lab., Lemont, Ill.

A MEASUREMENT OF THE TRANSFER FUNCTION OF A FAST CRITICAL ASSEMBLY.

J. F. Boland, R. R. Smith, and R. E. Rice. Sept. 1957. 16p. Contract W-31-109-eng-38. \$0.50 (OTS).

The zero power transfer function of the ZPR-III mock-up assembly of EBR-I, Mark III was obtained over a frequency range from 0.01 to 10 cycles per second. Various oscillator rod materials and shapes were investigated in order to determine the type of oscillator rod that would give the desired amplitude and wave shape for use in future measurements in the EBR-I, Mark III. A null balance system was used to obtain the amplitude and phase data to a reasonably high degree of accuracy. (auth)

13513

DETERMINATION OF THE RATIO OF CAPTURE TO FISSION CROSS SECTIONS IN EBR-1.

P. Kafalas, M. Levenson, and C. M. Stevens (Argonne National Lab., Lemont, Ill.). Nuclear Sci. and Eng. 2, 657-63 (1957) Sept.

The variation of alpha, the ratio σ_c/σ_f , has been determined for U²³⁵ and Pu²³⁹ as a function of position in EBR. Samples of U²³⁵ and Pu²³⁹ were exposed in the core, inner

blanket, and outer blanket; the samples were analyzed radiochemically to determine the fraction fissioned and mass spectrometrically to determine the fraction which had undergone the capture process. The vertical and horizontal patterns of capture and fission probability have been determined. The variation of η in the reactor has been plotted, assuming that $\bar{\nu}$ remains constant. (auth)

13763 ANL-4316(Del.)

Argonne National Lab., Lemont, Ill.
METALLURGY DIVISION REPORT FOR JANUARY, FEBRUARY, AND MARCH 1949. Mar. 31, 1949. Decl. with deletions Mar. 11, 1957. 96p. Contract W-31-109-eng-38. \$15.30(ph OTS); \$5.40(mf OTS).

13864 ANL-5134

Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] JUNE 1, 1953 THROUGH AUGUST 31, 1953. Sept. 15, 1953. Decl. Mar. 28, 1957. 134p. Contract W-31-109-eng-38. \$21.30(ph OTS); \$6.90(mf OTS).

Progress is reported on the following studies design and criticality calculations of the Argonne Power Breeder Reactor and the Central Station Water Reactor, water corrosion of Nb-U alloys, corrosion of welded Zr alloys, boiling heat transfer tests, speed of thermocouple response, control rod drive mechanisms, corrosion of Savannah River fuel slugs, rod drop experiments on ZPR-I, organic coolant studies, and temperatures produced by Na-Hg reactions. (L.T.W.)

13866 ANL-5208(Del.)

Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] SEPTEMBER 1, 1953 THROUGH NOVEMBER 30, 1953. Dec. 15, 1953. Decl. with deletions Apr. 16, 1957. 170p. Contract W-31-109-eng-38. \$27.30(ph OTS); \$8.10(mf OTS).

Power Breeder Reactor The PBR, pilot plant, and EBR are described and compared. An optimum blanket cycle for PBR is analyzed with respect to costs, Pu inventory, Pu quality, and heat generation. A multigroup diffusion analysis of EBR and the pilot mockup was made. Experimental details and preliminary studies in connection with fuel elements fuel media, reflector and structural materials are briefly reported. Thermal analyses of core, blanket, cooling cycle, and steam plant were carried out. Some characteristics of electromagnetic pumps and wetting of an electromagnetic flowmeter were also studied. **Central Station Water Reactor.** A 1,000 Mw reactor design and feasibility study are presented. Criticality and reactivity studies were made, and other reactor characteristics were investigated. Corrosion data on U-Nb and U-Nb-Zr alloys by water are given, and the results of burnout tests on Ni and stainless steel tubes at 1,000 and 2,000 psia are included. **Reactor Physics.** A Monte Carlo analysis of slant transmission and reflection of neutrons by water slabs of varying thickness was conducted, and a method for calculating coolant water activity for a boiling reactor was found. **Corrosion studies.** Corrosion inhibitors for type 347 stainless steel tested were SiO_2 , GeO_2 , and ZnO . **Instrumentation** The volt-ampere characteristic of Si, Se, and Ge diodes vs neutron irradiation was studied. (T.R.H.)

13868 ANL-5260(Del. 2)

Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] DECEMBER 1, 1953 THROUGH MARCH 30, 1954. Apr. 15, 1954. Decl. with deletions Apr. 8, 1957. 185p. Contract W-31-109-eng-38. \$31.80(ph OTS), \$9.30(mf OTS)

Boiling Experimental Reactor (BER). An H_2O -type reactor has been selected for an initial pilot boiling reactor.

Information given here pertains to the D_2O design developed during the course of study. Plate-type and co-extruded tubular fuel element assemblies were considered for this reactor. Several cross section drawings of the proposed design are given. **EBR-II.** Several tank designs are proposed whereby the primary system, submerged in a Na-filled, double-walled tank, may be connected to the reactor vessel. A fuel element loading and unloading mechanism has been designed which incorporates a boom carried by a vertical column, and a moving head along the boom. **Reactor Physics.** Reactor physics calculations are reported on the feasibility of a pilot D_2O boiling reactor of 20 Mw heat output. The basic core design provides for fuel tube clusters, consisting of Zircaloy-clad fuel tubes containing slightly enriched U alloyed with Zr and Nb. Procedures for the determination of lattice constants are discussed, and the critical enrichment was determined for a core without burnup allowance. A similar analysis is made for a supercritical reactor. It was found that an enrichment was necessary for 10,000 Mwd/t exposure. The effects of moderator and fuel region D_2O density, and neutron and metal temperature on the reactivity of a boiling reactor are considered. Criticality changes resulting from small additions of H_2O and D_2O to the reactor were also evaluated. A number of activation cross sections were measured in various locations of the EBR. Tabular data of these cross sections referred to Au are presented for $\text{Ni}^{64}(\text{n},\gamma)$, $\text{B}^{209}(\text{n},\gamma)$, $\text{Fe}^{56}(\text{n},\text{p})$, $\text{Zr}^{96}(\text{n},\gamma)$, $\text{Fe}^{58}(\text{n},\gamma)$, and $\text{Pb}^{208}(\text{n},\gamma)$ reactions. UNIVAC solutions of the multigroup diffusion equations in cylindrical geometry were obtained for a variety of specific problems related to fast reactors. Tables of cross section data and problem parameters are given. **Fuel Elements.** Several new techniques were tested for the fabrication of ball and wire fuel elements. Samples of stainless steel UO_2 fuel plates were corrosion tested in H_2O saturated with O_2 and H_2 . The plates were not corroded during a 16-week exposure at 500, 600, and 680°F. Selected data are given for the water corrosion of Ni, Si, and Zr alloys of U in the unclad condition. Detailed information was obtained on the stages of corrosion penetration for a group of Zircaloy-clad U-Nb fuel plates in 500°F H_2O . **Moderator, Reflector, and Structural Materials** $\text{Zn}(\text{C}_2\text{H}_3\text{O}_2)_2$ and ZnO were found to be effective corrosion inhibitors of 347 stainless steel and 2-S Al. An apparatus was constructed for thermal conductivity determinations of reactor materials up to 1000°F. The embrittling effects of irradiation, H_2 , and temperature on the impact properties of materials were studied. The specimens tested included a variety of stainless steels, Ni-A, Ni alloys, Co alloys, and Zr. **Reactor Coolant, Heat Transfer, and Power Generation.** A new series of tests have been initiated on diphenyl as an organic moderator material, in order to determine the amount of carbonization on heat generating surfaces and the degree of decomposition as a function of neutron dosage. Tar deposits and an 11.1% decomposition was found in a closed loop test operated in the EBR for 73 hr at a power level of 1200 kw. Several thermal test loops for organic coolants are described. A liquid-metal level indicator has been developed which measures a change in the electrical inductance of a 60-cycle circuit, according to the amount of metal surrounding the test element. With a view of reducing high D_2O inventories for boiling D_2O reactors, an air-water atmospheric test facility was constructed in order to study separator configurations. A complete description of the apparatus is given, together with some representative test results and separator designs. A technique has been developed for the thermal analysis of complex fuel element designs under steady-state and transient conditions. An electrical, geometrical, analogue device is used, consisting of a resistive plotting surface of conducting paper, a d-c source, a resistive network coupling the plotting surface to the source, and a null-type voltage measuring device. Comparison of the analogue

data with analytical results shows close agreement. A group of 16 fuel element configurations have been analyzed from the standpoint of maximum permissible temperatures and heat transfer characteristics, to determine the most feasible design for the PBR. The conclusions include other factors besides heat transfer considerations. A study was also made of the thermal conductivity characteristics of laminated fuel elements, where U particles and stagnant Na are arranged in layers within a containing structure. The condition of parallel heat flow is obtained when the laminations are made thin. Preliminary investigation of power plant cycle and reactor core designs using flashed water to produce steam indicates that a more detailed study is justified. Local heat transfer film coefficients of Na and NaK under conditions of high heat flux and non-circular channels will be measured in a heat transfer loop which is described. The construction of a 600-psi multiple channel boiling density test facility has been completed. A detailed description of the system and test section is presented. The facility will be used for the study of boiling H_2O phenomena at pressures near reactor operating conditions. The flow rate of naturally circulating boiling water was studied as a function of additives such as $HC_2H_3O_2$, Dow-Corning anti-foam, and caustic soda. Slow and rapid heat transients in boiling D_2O reactors were simulated in a water expulsion test apparatus, which was used to obtain considerable data on the characteristics of a 7-tube fuel-element cluster. The removal of core decay heat in the event of a sudden reactor shutdown accompanied by a loss of coolant water will be assumed by the natural circulation of cooling air through the core. In connection with such an event several tests were made to determine the relationship between maximum metal temperature and the heat that can be dissipated to ambient air. Four- and five-ft channels, $\frac{3}{4} \times 4\frac{1}{2}$ in. were tested. Instrumentation and Control. A magnetic jack-type control rod mechanism was developed. The device is designed for operation in a pressure shell located in the reactor coolant. The performance of Na and H_2O seals was tested, together with H_2O -lubricated ball bearings mounted vertically under a thrust load. Irradiation Test Facilities. Thermal neutron flux measurements were made in the Argonne Water Loop at the MTR. (For preceding period see ANL-5208) (K S)

13869 ANL-5461

Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT ON THE POWER REACTOR PROGRAM [FOR] JANUARY 1, 1955 THROUGH MARCH 31, 1955. Apr. 15, 1955. Decl. Jan. 17, 1957. 124p. Contract W-31-109-eng-38. \$0.60(OTS)

Construction drawings and specifications have been completed for the building and power plant of the Experimental Boiling Water Reactor (EBWR). The design of the pressure vessel has been evaluated for safety in case of internal explosion. The results of steam pipe rupture studies and tests of impurity carry-over in the boiling process are reported. The design and performance of a fuel assembly grappling tool are described. Design studies of control rods and control rod drive mechanisms are continuing. Tests are being made to observe the corrosion behavior of Zircaloy 2 coupled with types 304 and 347 stainless steels in static and dynamic high-temperature water. The results of stress analyses of EBWR type fuel plates and subassemblies are reported. Data are presented from corrosion and irradiation tests of high-uranium alloys, ThO_2-VO_2 bodies, and Al (Ni-plated or with Ni additions). Theoretical studies have been performed on the effect of the spatial distribution of voids on reactivity, and results are reported. The results of final shielding calculations for the EBWR are presented in graphical form. Calculations have been performed to determine the operating characteristics of D_2O -cooled and moderated, EBWR-type boiling reactors. Construction and pre-installation

performance tests of the Experimental Breeder Reactor (EBR-II) Working Model are described. Low-frequency induction heating is being studied for the Na piping system. Convection heat transfer data are presented on eutectic NaK. Results of an EBR-II design study are reported, including a skeleton flow chart of the entire plant and a temperature-enthalpy diagram for the steam generator. A thermal analysis has been made of radiator-type fuel elements for the EBR-II, and results are presented. Results of irradiation tests of EBR-II pin-type fuel elements are summarized. Preliminary information from a set of UNIVAC multigroup calculations on the EBR-II is reported. (T.R.H.)

13870 ANL-5594(Del.)

Argonne National Lab., Lemont, Ill.

QUARTERLY PROGRESS REPORT ON REACTOR DEVELOPMENT 4000 PROGRAM. Stuart McLain, comp. Apr. 30, 1956. Decl. with deletions Mar. 14, 1957. 66p. Contract W-31-109-eng-38. \$10.80(ph OTS), \$3.90(mf OTS).

Development studies on the EBWR and operating data on the water and steam cycles of Borax III are reported. Operation of the EBR-II Working Model is discussed. Data are reported on thermal cycling of "fission" alloys, U-Fe compound formation at Na-wetted surfaces of U and stainless steel, corrosive effects of diphenyl, corrosion of high-U alloys in 260°C water, corrosion of Al alloys in 290°C water, boiling corrosion of Al-Ni alloy, thorium-uranium corrosion in 316°C degassed water, and B_4C solution in 215 and 260°C demineralized water. Fluoride volatility and pyrometallurgical separations processing studies are briefly discussed. (L M T.)

13879 APDA-105

Atomic Power Development Associates, Inc., Detroit.
FAST REACTOR CONTROL AND SAFETY MEETING [HELD AT DETROIT EDISON COMPANY ON] NOVEMBER 10-11 1954. E. C. Branyan, ed. Mar. 11, 1957. 206p. \$31.80(ph OTS), \$9.30(mf OTS)

The report of the proceedings of a 2-day meeting on fast-reactor control and safety held in Detroit on Nov. 10-11, 1954 is given. The scope of the meeting was limited to the nuclear kinetics of fast reactors. (F.S.)

13942 KAPL-M-KAK-6

Knolls Atomic Power Lab., Schenectady, N. Y.
PRELIMINARY STUDY OF FAST REACTOR SLUG. K. A. Kesselring. May 10, 1949. Decl. Apr. 8, 1957. 18p. Contract W-31-109-eng-52. \$3.30(ph OTS); \$2.40(mf OTS).

The results of a preliminary study on a fast spectrum reactor are summarized. Discussions are included on the core composition, the surface area exposed to the primary flowing coolant, properties of the coolant annuli and flow, and the total fuel content. Heat transfer and coolant flow calculations for the fast reactor pin are included in an appendix. (B J H)

13952 LA-1192

Los Alamos Scientific Lab., N. Mex.

NEUTRON ENERGY DISTRIBUTIONS INSIDE THE FAST REACTOR. Norris Nereson. Dec. 15, 1950. Decl. Feb. 7, 1957. 49p. Contract W-7405-eng-36. \$0.45(OTS).

Various experimental methods utilizing nuclear plates have been used in the determination of neutron energy distribution inside the Los Alamos Fast Reactor. Detailed results are given of determinations obtained both by collimating the neutrons from the pot and by experiments carried out directly at the center of the pot. (B J H)

13953 LA-1614

Los Alamos Scientific Lab., N. Mex.

LADY GODIVA: AN UNREFLECTED URANIUM-235

CRITICAL ASSEMBLY R E Peterson and G A Newby Sept 1953 Decl Apr 12, 1957 49p Contract W-7405-eng-36 \$7 80(ph OTS), \$3 30(mf OTS)

A spherical, unreflected U^{235} critical assembly has been in operation since August, 1951. A remotely-controlled mechanical system is used to assemble subcritical components of the sphere, and reactivity is adjusted with U^{235} control rods positioned in the sphere. In addition to investigations of the neutron spectrum of the assembly, observation of the changes of reactivity produced by inserting foreign materials into the assembly, and the determination of parameters such as the temperature coefficient of reactivity, studies have been made of the behavior of the assembly at reactivities above prompt critical. (auth)

13954 LA-1653

Los Alamos Scientific Lab, N Mex
NEUTRON DETECTOR TRAVERSES IN THE TOPSY AND GODIVA CRITICAL ASSEMBLIES G A Linenberger, L L Lowry, J A Grundl, and R N Olcott June 1953 Decl Apr 1, 1957 24p Contract W-7405-eng-36 \$4 80(ph OTS), \$2 70(mf OTS)

Neutron detector traverses of the untamped U^{235} (Godiva) and the U^{238} - or Ni-tamped U^{235} (Topsy) metal critical assemblies have been obtained by counting γ -activity of U^{235} and U^{238} fission products, fission fragments of Np^{237} in a spiral chamber, and β -activity of Au and S. At a few positions within the assemblies, and for the U^{235} fission spectrum, cross section ratios of a number of pairs of fissionable isotopes were determined by means of a comparison fission chamber. At the center of Godiva, $\sigma_f(25)/\sigma_f(28) = 6.2$, and this ratio for the Topsy $U^{235}-U^{238}$ assembly ranges from 6.8 at the center to 76 at a radius of $8\frac{1}{4}$ ". Results of radiochemical analyses for fission and other reaction products are listed. (auth)

13955 LA-1708

Los Alamos Scientific Lab., N. Mex.
MATERIAL REPLACEMENT MEASUREMENTS IN TOPSY

AND GODIVA ASSEMBLIES. L. B. Engle, G. E. Hansen, H. C. Paxton, J. C. Hoogterp, and D. S. Young. July 1954. Decl May 9, 1957 98p Contract W-7405-eng-36 \$15 30 (ph OTS), \$5 40(mf OTS)

This report brings together an extensive accumulation of material replacement (danger coefficient) data for the various oralloy critical assemblies at Pajarito. Corresponding values of effective absorption and transport cross sections are derived. In certain favorable cases, inelastic scattering contributions to the effective absorption cross sections are estimated. Special applications of material replacement data include computation of the relationship between oralloy critical mass and concentration of a diluent, and hydrogen isotopic analysis of heavy water and heavy polythene. (auth)

13957 LWS-22710

California Research and Development Co., Livermore, Calif.

CONCEPTUAL DESIGN OF A MOLTEN FUEL FBR. R. E Kelly, L B Robbins, and J B. Stichka July 2, 1953 Decl Mar 4, 1957 33p (P-F-13) \$15 30(ph OTS), \$5 40(mf OTS)

Design drawings and an economic evaluation is presented for a power reactor having a molten $Pu^{239}-Ni-U^{238}$ fuel in a pot-type core. The nuclear properties, heat balance, and fuel processing costs for the reactor are considered. Solutions to the many metallurgical problems are assumed. (D E B)

13989 TID-10077

Dow Chemical-Detroit Edison and Associates. Atomic Power Development Project, Detroit.

INFORMATION REPORT TO THE PROJECT COMPANIES OF THE DOW CHEMICAL-DETROIT EDISON AND ASSOCIATES, NUCLEAR POWER DEVELOPMENT PROJECT. Dec. 1, 1953. Decl. Mar. 11, 1957. 164p. \$0.70(OTS).

The development and design of an economic power reactor and associated processing plant are presented. (L.M.T.)

Volume 12, 1958

655 KAPL-1784

Knolls Atomic Power Lab., Schenectady, N. Y.
THE FAST OXIDE BREEDER—FUEL IRRADIATION
EXPERIMENTS. W. M. Cashin. Aug. 26, 1957. 22p.
Contract W-31-109-Eng-52. \$0.50(OTS).

A fuel concept applicable to a fast breeder reactor is described. Mixed PuO_2 — UO_2 fuel elements were irradiated. Central void formation was observed, but no settling of the fuel occurred. Details of the irradiation experiments are given. (auth)

749 ANL-5529(Del.)

Argonne National Lab., Lemont, Ill.
CHEMICAL ENGINEERING DIVISION SUMMARY REPORT
[FOR] OCTOBER, NOVEMBER, AND DECEMBER, 1955.
Feb. 10, 1956. Decl. with deletions Feb. 12, 1957. 130p.
Contract W-31-109-eng-38. \$19.80(ph OTS); \$6.30(mf OTS).

Progress is reported on the following investigations. direct-cycle boiling-reactor entrainment studies, theory of solvent extraction, fused-salt process for U recovery from STR-type fuel elements; melt refining of EBR fuel elements; volatilization of PuF_6 from Pu—U alloys, and x-ray spectrometric determination of U (C.W.H.)

750 ANL-LB-SL-1016

Argonne National Lab., Lemont, Ill.
REVISED EBR-II FLOWSHEET FOR FUEL CYCLE WITH
FUEL PURIFICATION BY OXIDATIVE SLAGGING. L.
Burris, Jr. and I. G. Dillon. Feb. 23, 1956. Decl. Mar.
11, 1957. 15p. Contract [W-31-109-eng-38]. \$3 30(ph OTS), \$2.40(mf OTS).

780 KAPL-M-SSJ-3

Knolls Atomic Power Lab., Schenectady, N. Y.
RADIATION EFFECTS IN THE CHEMICAL PROCESSING
OF FAST OXIDE BREEDER FUEL. Samuel S. Jones.
May 31, 1956. Decl. Mar. 12, 1957. 19p. Contract W-31-
109-Eng 52. \$3.30(ph OTS), \$2.40(mf OTS).

In processing spent fuel from the KAPL Plutonium Power Breeder Reactor in the Purex Process the effect of radiation damage becomes prominent. The effect of the radiation on the following process factors is discussed: temperature, gas evolution, viscosity, corrosion, solute oxidation states, solvent deterioration, and phase interface behavior (T.R.H.)

801 ORNL-1941(Del.)

Oak Ridge National Lab., Tenn.
ORNL METAL RECOVERY PLANT PROCESSING CLEMENTINE REACTOR FUEL ELEMENTS. Terminal Report.
J. L. Matherne. Sept. 7, 1955. Decl. with deletions Feb.
28, 1957. 19p. Contract W-7405-eng-26. \$3.30(ph OTS), \$2.40(mf OTS).

The processing of Los Alamos Fast Breeder (Clementine) Reactor fuel elements to recover Pu and Am is described. Chemical and equipment flowsheets are included. (C.W.H.)

887 ANL-4399(Del.)

Argonne National Lab., Lemont, Ill.
METALLURGY DIVISION QUARTERLY REPORT FOR
OCTOBER, NOVEMBER AND DECEMBER 1949. Dec.
31, 1949. Decl. with deletions Feb. 12, 1957. 99p.
Contract W-31-109-eng-38. \$16.80(ph OTS); \$5.70(mf OTS).

944 KAPL-1302

Knolls Atomic Power Lab., Schenectady, N. Y.
REPORT OF THE NUCLEAR PHYSICS SECTION FOR
DECEMBER 1954, JANUARY, FEBRUARY 1955. Decl.
Mar. 30, 1957. 68p. Contract W-31-109-eng-52. \$0.45 (OTS).

A new U^{235} fission chamber which is ready for installation is described. A 200-channel time-gated coincidence time analyzer has been developed. Mass spectrometry studies include brief discussions of a new spectrometer tube and better magnetic field control. Results are presented from studies on determining upper limits of "nonexistent" isotopes of the following elements: Li, Na, Mg, K, V, Mn, Nb, Ru, In, Cs, La, Pb, Ta, and U. Procedures for determining trace amounts of U and Pu in the atmosphere are briefly discussed. The contribution from the Compton effect in a γ spectrometer is canceled out by surrounding the spectrometer crystal with many NaI crystals which detect the Compton scattered quanta. The fuel assembly for the KAPL Plutonium Power Breeder is discussed. Most probable values for the parameters of the 3-ev resonance of Pu^{239} were derived. Radioactivity problems for open-cycle boiling water reactor systems were studied (L.M.T.)

969 ANL-4847(Rev.)

Argonne National Lab., Lemont, Ill.
THE MANUFACTURE OF ENRICHED URANIUM FUEL
SLUGS FOR THE EXPERIMENTAL BREEDER REACTOR.
Arthur B. Shuck. Apr. 20, 1953. Decl. Mar. 18, 1957.
54p. Contract W-31-109-eng-38. \$0.50(OTS).

The specifications, materials, and sequence of operations used to found and fabricate the first charge of enriched U fuel in the Experimental Breeder Reactor are described. Deviations from the standard process are discussed. Quantitative data are presented to show the over-all yield of fuel from the process and the quantities of material in the residues. The method of ignition of turnings is described together with quantitative data on the operation. (auth)

970 ANL-5012

Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT FOR DECEMBER 1, 1952 THROUGH FEBRUARY 28, 1953. Mar. 15, 1953. Decl. Apr. 8, 1957. 34p. Contract W-31-109-eng-38. \$6.30(ph OTS), \$3.00(mf OTS).

Progress is reported on the following: design of a boiling reactor, the power breeder reactor (physics, thermal analysis, and experimental and development program),

corrosion of Al coating, irradiation of wax and lubricants, effect of long-term irradiation on pile reactivity, caused by changing isotope ratios and fission products, Ni plating of Zr-U alloys, design and operation of a high-pressure, continuous-flow pH meter, stress-rupture tests on nuclear fuses for emergency release of nuclear poisons, and design of miniature fission pulse counters. (J.E.D.)

971 ANL-5060(Del.)

Argonne National Lab., Lemont, Ill.

REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] MARCH 1, 1953 THROUGH MAY 31, 1953 June 15, 1953 Decl. with deletions Jan 17, 1957 157p. Contract W-31-109-eng-38 \$24.30(ph OTS); \$7 50(mf OTS)

Power Breeder Reactor Conceptual design studies for the Power Breeder Reactor indicate the feasibility of a contained primary coolant system in a package arrangement. One- and two-group calculations are presented in detail for preliminary PBR estimations. A technique for impregnating UO_2 with NaK has been developed. The resulting putty-type fuel was successfully extruded through a $\frac{3}{8}$ -in. hole. Data are presented on the fuel element design factors imposed by heat transfer and cooling limitations for fuel plates, pins, and spheres. Fission heat liberated in the PBR blanket was estimated to be 7.7% of the total core power. Design criteria resulting from this estimate are analyzed. Heat exchanger size for the transfer of 500 Mw of heat from Na to NaK has been calculated for the intermediate exchanger and for the NaK-to- H_2O steam generator. A theoretical treatment is presented for the thermal stress analysis of stainless steel-clad U fuel plates. Boiling Reactor Heat Transfer Experiments. Further results have been obtained from natural-circulation boiling-density tests and investigations of natural-circulation flow rates. A special thermocouple design, meeting the rapid-response requirements of high-power plate transients, is described. Feasibility studies on heavy water boiling reactors are presented for spherical boiler, pressurized tube, and horizontal pressure tank configurations. Reactor Physics Detailed theoretical methods are developed for determining long-time changes in reactors. Application is made to a one-group calculation of such changes in a spherically symmetric fast converter. Further data are reported from investigations of irradiation effects on the reactivity of U^{238} - U^{235} and Th^{232} - U^{235} systems. New data are available on the efficiency of the upper shield of the EBR, determined by neutron flux traverses taken on axes through the core center and inner blanket. Fuel Element Development Attention has been directed to the study of U alloys with lower corrosion rates in water. Corrosion characteristics are given for UO_2 , Zr-clad U plates, U, and Mg in diphenyl, and free energies for chemical reactions commonly occurring in reactor studies are tabulated. A study is reported on the pressure-drop characteristics of fuel elements with variations in flow rate, metal-to-water ratio, pitch, and water density. The increased resistance of such fuel wires was found to be less than anticipated. Radiation Studies. A comparison is made between the radiation effects of Dow-Corning high-vacuum silicone stopcock grease and Standard Oil Co. Lg-0147 grease. (C.H.)

974 ANL-5345(Del.2)

Argonne National Lab., Lemont, Ill.

REACTOR ENGINEERING DIVISION QUARTERLY REPORT FOR PERIOD JULY 1, 1954 THROUGH SEPTEMBER 30, 1954. Oct. 15, 1954. Decl. with deletions Mar. 29, 1957. 127p. Contract W-31-109-eng-38. \$19.80(ph OTS); \$6.30 (mf OTS).

Reactor Programs. The design of the Boiling Experimental Reactor is described. The Borax-I reactor was intentionally destroyed by suddenly adding 4% excess re-

activity. Design of the Borax-II reactor is described. It will be used for studies of transient behavior and steady-state operation as a steam producer. A preliminary design configuration has been established in EBR-II employing central, inner, and outer blankets. Fuel geometries including a smaller pin-type element are being investigated. Supporting Design, Research, and Development. The accuracy of diffusion theory for fast reactors was investigated. Water corrosion tests have been performed on high-U alloys, Zr- and Ti-base alloys, and Ni-plated Al plates. Thermal cycling tests were performed on fuel rods and pin-type fuel elements. Results are reported on tests on a current conductor connection for a high-temperature d-c electromagnetic pump, on Na valves with conventional packing, and on a rod-gripper test facility for the EBR-II. A vacuum cup sampler for high-temperature Na and NaK systems is described. Lifting and loading mechanisms for the EBR-II working model are described. (T.R.H.)

975 ANL-5398(Del.)

Argonne National Lab., Lemont, Ill.

REACTOR DEVELOPMENT QUARTERLY PROGRESS REPORT. Jan 31, 1955. Decl. with deletions Jan. 21, 1957. 63p. Contract W-31-109-eng-38. \$10.80(ph OTS), \$3.90(mf OTS).

The design and construction program of the Boiling Experimental Reactor is reviewed. A number of preliminary experiments were performed with Borax-II at pressures between 75 and 300 psi. The most corrosion-resistant U-Zr-Nb alloy developed so far is produced by heating in a vacuum to the γ phase, quenching, and aging for 2 hr at 400°C. Special attention is given to the removal of H_2 from the material. Unclad and unirradiated samples of U-Nb and U-Nb-Zr alloys were corrosion tested in H_2O . Corrosion rates were also measured under irradiation conditions in CP-5. Elongation measurements of irradiated wrought and cast U-Zr material suggested no way for treating the wrought fuel so that stability comparable to the cast material could be obtained. Natural circulation boiling density tests at 600 psia were made in order to determine the effects of channel cross section and subcooling on the steam void fraction. Results of autoclave and dynamic corrosion studies of 2-S Al in H_2O are reported. These results include the testing of Ni-clad samples. A large number of criticality calculations were performed for the EBR-II and the PBR. The solubility of ThO_2 pellets containing various concentrations of U_3O_8 was tested in water at 316°C for periods of 672 to 744 hr. None of the samples disintegrated, although at least one sample developed cracks. Solutions of reactor kinetic equations were attempted for the purpose of studying transients in reactors with lifetimes of 7×10^{-8} , 10^{-7} , and 6×10^{-5} sec. Ignition experiments were performed on Th, Cu, Al, Fe, Mg, Zr, and fluorothene when contacted with fluorides. Except for Zr and fluorothene, the materials did not ignite. (C.H.)

976 ANL-5492(Del.)

Argonne National Lab., Lemont, Ill.

FAST REACTOR PHYSICS CALCULATIONS. Robert Avery. Jan. 1956. Decl. with deletions Jan. 17, 1957. 80p. Contract W-31-109-eng-38. \$13.80(ph OTS); \$4.80(mf OTS).

Multigroup diffusion theory calculations performed in connection with the fast power breeder reactor development program at Argonne are summarized. The problems investigated relate to steady-state properties; long-term changes; reactivity coefficients associated with temperature changes; effectiveness of various control methods; evaluation of neutron lifetime; various methods of radial power flattening; various parametric studies to determine optimum design points; accuracy of diffusion theory for systems under consideration, and analysis of experimental results. (auth)

977 ANL-5511

Argonne National Lab., Lemont, Ill.
 REACTOR ENGINEERING DIVISION QUARTERLY REPORT
 [FOR] JULY, AUGUST, SEPTEMBER 1955. SECTION II.
 Jan. 1956. Decl. Jan. 17, 1957. 186p. Contract W-31-109-
 eng-38. \$1.10(OTS).

Experimental Boiling Water Reactor. The boiling water corrosion of Zircaloy-2 clad stainless steel thimbles is shown. Designs of the EBWR control rod and drive mechanism are given. The deceleration curve for the drive shaft at 600-psig operation is given. The prototype test facility for the control mechanisms is also shown. Calculated heat generation and thermal stresses in the pressure vessel wall for each type of thermal shielding are tabulated. The mechanical arrangement of the thermal shielding is given.

The pressure vessel containment structure is described in detail, and calculated impact loading stresses are summarized. A redwood, steel, and celotex blast shield for the pressure vessel is also shown. Fast and thermal neutron flux calculations for the heavy water-reflected EBWR in the adjusted shield are summarized. The physical arrangement of this shielding is also described. An emergency boric acid injection system to reduce the reactor core activity below criticality was designed. The reactor water purification system is discussed. A schematic flow diagram of the air drying and fluid recovery system is given. Estimated flows in this system are also given, and the operation of the system is briefly discussed. A schematic plan view is shown of the EBWR power plant and control areas. A simplified line diagram is also given of the EBWR electrical system. The schematic for the emergency air-lock doors for the EBWR is shown. Supporting and Alternate Design Research and Development. The design of a

boiling D₂O reactor is briefly discussed. Data given on boiling heat transfer studies include the ratio of steam-to-liquid velocity as a function of the variable inlet temperature, power, flow, and pressure. The analytical and experimental studies made of the 600 psi, unrestricted, single-channel circulation loop (Armadilla) are described, and preliminary data are plotted. These plots show modified Martinelli-Nelson factors as a function of slip ratio and exit and local void fraction. Data from single channel forced and natural circulation boiling density tests at 150 psig are summarized. Slip ratios are also plotted as a function of inlet water velocity at 150 and 250 psig, indicating that there is no significant pressure effect on the slip ratio pattern in going from 150 to 250 psig. The corrosion of Zircaloy 2-clad, U-Zr fuel plates in 550°F degassed water. The corrosion behavior of Al-clad, natural U plates is compared to that of EBWR reference alloy plates. The corrosion of Al-Ni alloy in 500°F H₂O is also shown. Corrosion data are also reported for austenitic stainless steel, and its weight changes are plotted. Corrosion and irradiation tests were continued on urania-thoria bodies. Several methods were tested for the detection and location of ruptured fuel elements in the core of a reactor. These included comparisons of fission product activities in various sampling devices and of γ activities in H₂O samples. It was, however, concluded that neither of these methods would be useful. A proposed alternate control rod for the EBWR is described and discussed. Progress in the development of boron-stainless steel alloy is discussed. Boron-stainless steel corrosion tests are summarized. Results of experiments on the removal of boric acid using Amberlite IRA-400 resin are summarized. Experimental Breeder Reactor I. The conversion ratio was measured in the second U fuel loading of the EBR-I. U²³⁸ capture distributions and U²³⁵ fission distributions in the reactor are shown. The conversion ratio with the cup about $\frac{1}{4}$ in. below its uppermost position was found to be 1.03 ± 0.05 . Activation cross section ratios at various locations in the outer blanket of the EBR were surveyed. Experimental and theoretical values are compared. Experimental Breeder Reactor II. The dissipation of

simulated decay heat generated in the EBR-II core sub-assembly is shown. Water pressure drop tests were conducted on the complete EBR-II core subassembly, and the results of the tests are summarized. Calculations of hold-down forces in the core are also given. Irradiation tests were made on EBR-II fuel elements. EBR-II Working Model. A general description is given of the control rod and the cover raising and rotating mechanisms. A diagram of the fuel unloading system is given. Performance characteristics are given for a model tank pump in Na at 370°C. Supporting and Alternate Design Research and Development. The composition, grain size, and appearance of Mo-U and Zr-U alloys, which were thermal cycled in NaK are shown. Equipment is shown for the induction heating of the Na test loop, and the induction heating time for this loop is shown. Reactor Safety and Kinetics. Machine solutions of the kinetics equation are described in some detail. Analytical solutions of kinetics equations are also discussed. These methods were used to investigate the oscillatory behavior of the kinetic system for a reactor with a lifetime of 4×10^{-8} sec, and resultant calculations are plotted. The solution to the general problem of obtaining temperature distributions in two-region cylindrical geometry for an exponentially rising heat source in the central region is also given. The time variation of stored heat in a Na-cooled U rod during transient is also calculated, and the fraction of heat remaining in the fuel rod is shown as a function of the reciprocal period of the heat source. Fuel and Structural Materials. The failure in the tube assembly for the MTR loop water-to-water heat exchanger is discussed. Instrumentation. The pulse height distribution of fission signals in B¹⁰ and Al-U-Al neutron-sensitized photomultiplier tubes is shown. (C.H.)

978 ANL-5568(Del.)

Argonne National Lab., Lemont, Ill. and Nuclear Metals, Inc., Cambridge, Mass.

THE FABRICATION OF PROTOTYPE FUEL ELEMENTS FOR THE EXPERIMENTAL BOILING WATER REACTOR AND THE EXPERIMENTAL BREEDER REACTOR. Yearly Report: July 1, 1954 to June 30, 1955. H. F. Sawyer, P. Loewenstein, W. C. Paynton, and P. Corzine. May 1956. Decl. Mar. 14, 1957. 41p. Contracts W-31-109-eng-38 and AT(30-1)-1565. \$0.40(OTS).

The methods used for producing prototype fuel elements for the Experimental Boiling Water and Experimental Breeder Reactors are described. Foundry and fabrication techniques and problems are discussed; the results of product evaluation are given. Extrusion constants for several U alloys and for Zircaloy-2 are included. Thermal expansion and transformation characteristics for the U-5 w/o Zr-1.5 w/o Nb alloy are given. (auth)

991 CF-3582

[Chicago. Univ.] Metallurgical Lab.

DESIGN OF THE FAST NEUTRON BREEDER TEST PILE. PART II. Report for January-June 1946. Aug 7, 1946. Decl. Feb. 15, 1957. 122p. Contract W-7401-eng-37. \$0.60(OTS).

Part I was issued as CF-3414 (deleted version: TID-10085).

996 CF-52-8-230

Oak Ridge School of Reactor Technology, Tenn.

PRELIMINARY DESIGN CALCULATIONS FOR A FAST U²³⁸ CONVERTER. Reactor Design and Feasibility Problem. W. E. Davis, C. H. Bergsland, Harold Brammer, J. H. Kinginger, J. C. Nance, W. E. Schortmann, and P. H. Wackman. Aug. 20, 1952. Decl. Mar. 14, 1957. 142p. \$22.80(ph OTS); \$7.20(mf OTS).

A study was made on the nuclear behavior of a fast reactor based on a liquid U-Fe fuel system, with particular emphasis on the region of low U²³⁵ enrichment (10 -

20%) An enriched core- U^{238} blanket system was briefly investigated also. The U-Fe system was found to be operable from a nuclear standpoint but would present extremely difficult engineering problems because of the high temperatures involved (M H R)

1031 KAPL-131

Knolls Atomic Power Lab., Schenectady, N. Y.
PRELIMINARY INVESTIGATION OF A FAST REACTOR.
R Ehrlich Mar 15, 1949 Decl Mar 5, 1957 20p
Contract W-31-109-eng-52 \$3.30(ph OTS), \$2.40(mf OTS)

Brief multi-group analyses were made of the size and power distribution of reactors having no slowing-down material other than Na and structural material in the core. Investigations were made of assemblies with a 6-cm Be reflector and an infinite U blanket, a 12-cm MgO reflector and an infinite U blanket, or an infinite U reflector. Results of the analyses are tabulated. (B.J.H.)

1034 KAPL-334

Knolls Atomic Power Lab., Schenectady N. Y.
SECULAR EQUATIONS FOR BREEDING. Harvey Brooks.
Apr. 28, 1950 Decl Mar 27 1957. 20p Contract W
31-109-Eng-52 \$0.25(OTS)

Equations are given for an elementary analysis of breeding, simultaneous breeding and conversion, and the influence of higher isotopes on breeding. A numerical example is included. (M.H.R.)

1045 KAPL-M-HAB-2

[Knolls Atomic Power Lab., Schenectady, N. Y.]
SUMMARY OF BLANKET CALCULATIONS. H. A. Bethe.
July 9, 1947 Decl. Mar 29, 1957 4p. Contract [W-31-
109-Eng-52] \$1.80(ph OTS); \$1.80(mf OTS).

1047 KAPL-M-HB-31

Knolls Atomic Power Lab., Schenectady, N. Y.
EFFECTIVENESS OF BLANKET FISSIONS. H. Brooks
and H A Bethe Apr. 28, 1950 Decl. Mar. 12 1957 8p
Contract W-31-109-eng-52 \$1.80(ph OTS); \$1.80(mf
OTS)

Calculations are made to show the influence of a primary U^{238} or U^{235} fission in the blanket on the conversion or breeding ratio (W L H)

1065 KAPL-M-RSM-3

Knolls Atomic Power Lab., Schenectady, N. Y.
ESTIMATED INTERNAL PRESSURES IN FAST OXIDE
BREEDER FUEL ELEMENTS DUE TO FISSION GAS RE-
LEASE. R. S. Miller. Dec. 5, 1956. Decl. Mar. 19,
1957. 19p. Contract W-31-109-Eng-52. \$3.30(ph OTS);
\$2.40(mf OTS).

An estimate of the increase in the internal pressure of the KAPL Plutonium Power Breeder Reactor fuel pins due to the accumulation of fission gases was made for ranges of specific power, fuel burnup, and fuel density. Families of curves are presented which show the variation of internal pressure with fuel diameter for various values of the parameters listed above. Allowable upper limits on fuel diameter have been determined for the various cases considered as a function of the volume fraction of cladding iron present in the core region (auth)

1072 KAPL-M-WRK-3(Rev.) (Del.)

Knolls Atomic Power Lab., Schenectady, N. Y.
CONFERENCE ON THE EXPERIMENTAL BREEDER
REACTOR AT THE ARGONNE NATIONAL LABORATORY
ON APRIL 25, 1952. M. Davis, H. Hurwitz, Jr., W. R.
Kanne, T. M. Synder, H. E. Stevens, Jr., A. D. Tevebaugh,
and C. E. Weber. [nd]. Decl. with deletions Mar. 12, 1957.
17p. Contract W-31-109-ENG-52. \$3.30(ph OTS); \$2.40
(mf OTS).

Information obtained from and experiments planned for the EBR are discussed by ANL and KAPL personnel. (L.M.T.)

1073 LA-1525

Los Alamos Scientific Lab., N. Mex.
MATERIAL REPLACEMENT EXPERIMENTS: THEORY
AND MEASUREMENTS FOR THE LADY GODIVA ASSEM-
BLY. Gordon E. Hansen and Clifford Maier Apr. 23,
1953. Decl Apr. 1, 1957. 58p Contract W-7405-Eng-36.
\$9.30(ph OTS); \$3.60(mf OTS).

The perturbation theory for material replacement experiments in a bare assembly is given through second order, thus permitting corrections for sample size. Computed flux and adjoint distribution functions are tabulated for the Lady Godiva assembly, enabling the observed danger coefficients for U^{238} and U^{235} to be compared with corresponding predicted values. Consistency of this data is checked by its use in three independent combinations, each yielding the same value for the effective fraction of delayed neutrons from fast fission. Reactivity contributions associated with inelastic scattering were calculated and comparison made between central danger-coefficient ratios and ratios for the Topsy assembly. Evaluation of transport cross sections by means of replacement measurements in Godiva is illustrated for the several elements C, Cu, and Au. (auth)

1075 LA-2029

Los Alamos Scientific Lab., N. Mex.
TIME BEHAVIOR OF GODIVA THROUGH PROMPT
CRITICAL. T F. Wimet, L. B. Engle, G. A. Graves,
G. R. Keepin, Jr., and J. D. Orndoff. Apr. 1956 36p.
Decl Apr. 2, 1957 36p Contract W-7405-eng-36 \$0.40
(OTS)

Studies were made of the time behavior of a bare spherical U^{235} metal assembly (Godiva) following sudden increases in reactivity with essentially zero initial power level and no external neutron source. The highest final reactivity attained under controlled conditions was about 10 cents above prompt critical. In the region above prompt critical the observed fission rate increased exponentially until the decreasing reactivity from thermal expansion caused it to pass through a bell-shaped maximum or "burst." Following this burst, the fission rate dropped to a low-amplitude "tail" sustained by delayed neutrons and finally terminated upon automatic scrambling of the assembly. Burst amplitudes corresponding to instantaneous power levels of 10,000 Mw were observed with durations under a millisecond. Excellent agreement was found between measured reactor periods and calculations based on recent delayed neutron data. Over-all burst behavior also was found to agree with theory based on a space independent one-group model. (auth)

1076 LAMS-733(Del.)

[Los Alamos Scientific Lab., N. Mex.]
EFFECT OF TEMPERATURE AND REACTIVITY
CHANGES IN OPERATION OF THE LOS ALAMOS PLUTO-
NIUM REACTOR. David B. Hall and Jane H. Hall. May
28, 1948. Decl. with deletions Mar. 18, 1957. 40p. Con-
tract [W-7405-eng-36]. \$7.80(ph OTS), \$3.30(mf OTS).

The operation of the Fast Reactor is considered in terms of normal equilibrium conditions and normal shut-down. The proposed loading, control rod adjustment and subsequent "floating" operation are discussed. Safety devices and interlocks are described. Temperature and reactivity changes are examined with respect to various system failures, phase changes, and "flashing" of the reactor. Slow changes due to faulty slug cooling are also considered. The calculations were initially based upon 10 kw operation. Performance tests of the mercury system now indicate that 20-kw operation may be feasible. (D.E.B.)

1078 LWS-24934 (Del.)

California Research and Development Co., Livermore,
Calif.

CONCEPTUAL DESIGN OF THE ANL-FCR. S. E. Warner, W. B. Myers, J. C. Ekvall, and B. Lund. July 8, 1953. Decl. with deletions Feb. 26, 1957. 36p. \$6.30(ph OTS); \$3.00(mf OTS).

An initial design and economic analysis of the Fast Converter Reactor, a forerunner of the EBR, is presented. This design employs stainless steel-clad plate type fuel and breeding elements, liquid Na cooling, and has a power rating of 500 Mw. General reactor structure and operation are discussed, and fuel element assembly, installation, and removal details are given. (D E B)

1660 TID-7532(Pt 1)

Division of Reactor Development, AEC.

REACTOR CONTROL MEETING HELD IN LOS ANGELES, MARCH 6-8, 1957. Oct 1957. 245p. \$6.00(OTS).

Papers are included on the reactor control meeting summary remarks, a review of methods used in control rod analysis for reactor design at Bettis, formulas for control rod worth in thermal reactors, half-group theories for control rods, variational calculation of control rod absorption area, thin regions in diffusion theory calculations, flux distribution for a finite cylindrical bare homogeneous thermal reactor with a partially inserted control rod, dependence of control rod worth on reactor temperature and fuel exposure, multiple control rod calculations, heterogeneous reactor critical conditions using the small source theory, control rod worth studies on seed and blanket reactors, limitations of scram transient calibrations, APPR-1 control rod experiments and calculations, APPR control rods—theory and experiment, control rod worths vs temperature in LAPRE I, effects of rod shape on control worth, lattice experiments on thin black control rods, theory and experiments on rods containing rare earth metals, an empirical determination of fast absorption constants of control rods, the control of fast reactors, the general application of burnable poisons, experiments with control rods and voids in a pool type reactor, investigation of absorbing membranes in slab reactors, chemical control experience in boiling reactors, central control rod calculations for SPERT reactors, some control rod studies at the Materials Testing Reactor, a calculation of the effect of voids in large boiling reactors, reactor control work at General Electric APED. (M.H.R.)

1682

THE ENGINEERING DESIGN OF EBR-II. L. J. Koch (Argonne National Lab., Lemont, Ill.) Chem Eng Progr 52, Symposium Ser No 19, 37-47(1956)

The EBR-II is one of the five prototype industrial power reactors selected by the AEC for development. A plutonium-fueled, unmoderated, sodium-cooled reactor with a power rating of 60,000 kw of heat, it is a complete plant including the reactor, the steam-electric power plant, and an integral fuel-reprocessing and -fabrication facility. The plant includes engineering features which are believed to be applicable to large central-station power plants and employs system components large enough to be directly applicable to large central-station power plants and employs system components large enough to be directly applicable to large plants. (auth)

1889 LWS-22679(Del.)

California Research and Development Co., Livermore, Calif.

RADIATION EFFECTS IN AQUEOUS PROCESSING OF THE FBR. R. E. Elson and B. Rubin. May 6, 1953. Decl. with deletions Feb. 27, 1957. 10p. (D-F-14). \$1.80(ph OTS); \$1.80(mf OTS).

Aqueous processing is possible for the FBR core and blanket or the core alone. The most stringent conditions were considered: nearly saturated UNH, no loss of activity in off gases, and no cooling. A more realistic evaluation of these factors will lower the levels of activity, however,

there are a large number of unknowns whose effects must be known and methods of overcoming deleterious effects will have to be determined. (F.S.)

1976 BMI-T-6

Battelle Memorial Inst., Columbus, Ohio

THE BONDING OF URANIUM TO NICKEL. H. A. Saller and J. R. Keeler. Feb. 1, 1949. Decl. Feb. 13, 1957. 26p. Contract W-7405-eng-92. \$0.35(OTS).

A method of metallurgically bonding U inside Ni tubes was developed for use in jacketing the fuel rods in the Argonne Fast Neutron Pile. The bond was obtained by Ag plating the surfaces to be joined and diffusion bonding the mating Ag surfaces with the aid of an amalgamating agent such as Hg. Silver was found to be an excellent barrier between Ni and U at temperatures up to 600°C. Heating for 150 hr at 600°C caused practically no change in the bond, and only slight diffusion occurred in 125 hr at 700°C. The bond was homogeneous, having no cracks or voids, and withstood 60 cycles between 100°C and 550°C without failure. Indium could be substituted for Hg as an amalgamating agent, but Sn was unsatisfactory because it formed brittle compound layers with U on extended heating. Molten solders were investigated for joining U to Ni, but in all cases were unsatisfactory because circumferential cracking occurred in the bond when the joints cooled to room temperature. (auth)

2042 LA-1234

Los Alamos Scientific Lab., N. Mex.

ENERGY DISTRIBUTION OF FAST NEUTRON BEAM.

Norris Nereson, Ester Allison, Julia Carlson, Pearl Norwood, and Dorothy Squires. Feb. 15, 1951. Decl. Mar. 26, 1957. 37p. Contract W-7405-eng-36. \$0.35(OTS).

Experimental techniques are described for the spectral measurement of a collimated fast-neutron beam. A H₂-filled cloud chamber, proton-recording nuclear plates, and threshold fission foils were used as neutron detectors in the measurements. As an application of these techniques, the energy distribution and absolute flux of the fast neutron beam emerging from the Los Alamos fast reactor was measured from 0.1 to 18 Mev. (D.E.B.)

2047 TID-10073

Los Alamos Scientific Lab., N. Mex.

ENERGY SPECTRUM OF NEUTRONS FROM THERMAL NEUTRON FISSION OF U²³⁵ AND FROM AN UNTAMPERED MULTIPLYING ASSEMBLY OF U²³⁵. Glenn M. Frye, Jr., Juanita H. Gammel, and Louis Rosen. May 1954. Decl. Mar. 18, 1957. 18p. Contract W-7405-eng-36. \$0.25(OTS).

The neutron fission spectrum of U²³⁵ was measured from 0.3 to 12.0 Mev using the method of proton recoils in nuclear emulsions. The resulting neutron spectrum may be represented by the formula $N(E) = 1.058 \cdot 10^3 e^{-E/0.066} \sinh \sqrt{2.29E}$. The Godiva leakage spectrum was also measured and shows a higher proportion of low energy neutrons. (auth)

2055 ANL-FWT-105

Argonne National Lab., Lemont, Ill.

HAZARD EVALUATION REPORT ON THE FAST REACTOR ZERO POWER EXPERIMENT (ZPR-III). R. O. Brittan, B. Cerutti, H. V. Lichtenberger, R. Rice, and F. W. Thalgott. Apr. 1955. Decl. Feb. 13, 1957. 75p. Contract W-31-109-eng-38. \$2.00(OTS).

A fast zero power assembly is planned to obtain neutron physics information for fast power breeder reactor designs. The systems to be studied are blanketed cores having a range of volume fractions of U²³⁵, U²³⁸, Fe, and Na. Volume fractions are varied by arranging small pieces in drawers of two square arrays mounted in such a way that they may be brought together for criticality. The assembly is described, the extensive safety measures designed to prevent prompt criticality are discussed, and the effects of the maximum foreseeable accident are evaluated. (D.E.B.)

2058 CEPS-1121(Del.)

Commonwealth Edison Co. CEPS Group, Chicago.
A THIRD REPORT ON THE FEASIBILITY OF POWER
GENERATION USING NUCLEAR ENERGY. June 15, 1953.
Decl. with deletions Feb. 7, 1957. 104p. \$1.05(OTS).

A technical evaluation was made of four reactor power
plant designs. The evaluation was based on design data and
cost estimates furnished by the originator of each design.
In some cases the data were incomplete and estimates were
substituted for data. The designs considered were the So-
dium Graphite Reactor (North American Aviation); the Plu-
tonium Power Breeder Reactor (Knolls Atomic Power
Laboratory); Aqueous Homogeneous Reactor (Oak Ridge
National Laboratory); and the Boiling Reactor (Argonne
National Laboratory). (C.H.)

2063 CF-52-11-164

Oak Ridge National Lab., Tenn.
SCOPE OF P.B.R. STUDY. P. C. Zmola. Nov. 21, 1952.
Decl. Feb. 15, 1957. 9p. Contract [W-7405-eng-26].
\$1.80(ph OTS); \$1.80(mf OTS).

2081 KAPL-294

Knolls Atomic Power Lab., Schenectady, N. Y.
THEORETICAL ESTIMATE OF MAXIMUM POSSIBLE NU-
CLEAR EXPLOSION. H. A. Bethe. Jan. 31, 1950. Decl.
Feb. 22, 1957. 31p. Contract W-31-109-Eng-52. \$6.30
(ph OTS); \$3.00(mf OTS).

The maximum nuclear accident which could occur in a
Na-cooled, Be moderated, Pu and power producing reactor
is estimated theoretically (T.R.H.)

2082 KAPL-908(Del.)

Knolls Atomic Power Lab., Schenectady, N. Y.
KAPL STUDY OF HETEROGENEOUS PLUTONIUM
POWER REACTORS. R. Ehrlich and E. A. Luebke. May
1, 1953. Decl. with deletions Mar. 15, 1957. 58p. Con-
tract W-31-109-Eng-52. \$9.30(ph OTS), \$3.60(mf OTS).

Results of nuclear calculations for a variety of composi-
tions of fast, heterogeneous, sodium-cooled, U-235-fueled,
plutonium- and power-producing reactors are reported.
Core compositions typical of plate-, pin-, or wire-type
fuel elements and with uranium as metal, alloy, and oxide
were considered. These compositions included atom ratios
in the following range: U-238 to U-235 from 2 to 8; sodium
to U-235 from 1.5 to 12; iron to U-235 from 5 to 18; and
vanadium to U-235 from 11 to 33. Calculations were per-
formed to determine the effect of lead and iron reflectors
between the core and blanket. Both natural and depleted
uranium were evaluated as the blanket fertile material.
Reactors were compared on a basis of conversion ratio,
specific power, and the product of both. The calculated
results are in general agreement with the experimental
results from fast reactor assemblies. An analysis of the
effect of new cross-section values as they became avail-
able is included. (auth)

2089 KAPL-M-JHS-2

Knolls Atomic Power Lab., Schenectady, N. Y.
IMPROVEMENT OF CPC THREE-GROUP, THREE-
REGION REACTOR CALCULATIONS TO INCLUDE HIGH-
LY ABSORBING REFLECTORS AND BLANKETS. Jack
H. Smith and Charles D. McKereghan. Mar. 22, 1954.
Decl. Mar. 27, 1957. 11p. Contract W-31-109-eng-52.
\$1.80(ph OTS); \$1.80(mf OTS).

The CPC coding as originally employed to handle three-
group, three-region reactor calculations has been revised
to provide an accurate treatment for highly absorbing re-
flectors. As now instructed, the machines will correctly
handle configurations in which the reflector region is com-
pletely unlimited with regard to geometric size and both
scattering and absorption properties. (auth)

2122 LRL-68(Del.)

California Research & Development Co. Livermore
Research Lab., Livermore, Calif.
PLUTONIUM FAST POWER BREEDER REACTOR
STUDY. H. A. Brammer. Dec. 1953. Decl. with

deletions Feb. 27, 1957. 61p. Contract AT(11-1)-74.
\$9.30(ph OTS); \$3.60(mf OTS).

Preliminary nuclear physics calculations are made on
several reactor systems to determine a feasible configura-
tion for a Pu fast power breeder reactor, with particular
emphasis on achieving positive breeding gain, high specific
power, and good over-all efficiency of the system. Non-
blanket reactors of both single fuel pass (once-through) and
fuel recycle types are investigated, but none of the systems
calculated appears to be practical from the standpoint of
critical mass requirements. Blanketed reactor calculations
however, indicate that this type of system offers a feasible
approach to the power breeder reactor problem (auth)

2500 FRDC/P-252

Gt. Brit. Atomic Energy Research Establishment,
Harwell, Berks, England
REACTIVITY EFFECTS CONTROLLED BY THE BORON
SHUT-OFF AND BORON OSCILLATOR IN THE FAST
REACTOR. A. R. Baker. Sept. 1957. 5p

A new estimate was made for the reactivity controlled
by three shut-off rods in the Dounreay Fast Reactor,
namely $(0.82 \pm 0.08)\%$. The increase of 50% on the
previous figure is mainly due to an over-estimate of self-
shielding in the earlier work. Rotation of the proposed B¹⁰
oscillator will produce a peak-to-peak oscillation in reac-
tivity of 0.010% (with a possible error of a factor of two).
This will be satisfactory for the Dounreay equipment for
measuring the phase and amplitude of the power oscilla-
tion (auth)

2543

ROLE OF THE FAST POWER REACTOR IN DEVELOP-
MENTS OF NUCLEAR ENERGY. Mieczyslaw Taube.
Nucleonika 2, No. 1, 9-30(1957). (In Polish)

The economic potentials of nuclear energy in electrical
power production are analyzed. The commercial aspects
of U and Pu fuels are evaluated, and the distribution and
supply of nuclear and conventional fuels in the world are
estimated. (R.V.J.)

3204 ANL-5715

Argonne National Lab., Lemont, Ill.
HAZARDS SUMMARY REPORT ON THE OXIDE CRITICAL
EXPERIMENTS. W. C. Redman, J. A. Thie, and L. R.
Dates. Apr. 1957. 87p. Contract W-31-109-eng-38.
\$2.25(OTS).

A new zero-power reactor facility, designated as ZPR-
VII, which is under construction at the Argonne National
Laboratory is described. This facility will be used to
supply physics design information required in connection
with several phases of Argonne's boiling reactor program.
A description of the first reactor system to be investigated
in this facility and an assessment of the hazards associ-
ated with the planned program of measurements are
included. The first system to be investigated will employ
fuel in the form of pellets of a fired mixture of enriched
UO₂ and ThO₂. These pellets will be approximately 1/4 in.
in diameter, stacked to a height of 5 ft within an Al jacket
of 5/16-in. outer diameter. These fuel rods will be grouped
in a variety of arrangements within a 6 3/4-ft diameter tank
which can be filled with D₂O. The construction, instru-
mentation, and control system of the ZPR-VII conform
with past practice at Argonne. This reactor system is
unusually safe as a consequence of large negative tempera-
ture and void coefficients of reactivity. Large reactivity
effects are virtually impossible to achieve other than
through deliberate sabotage, and the inherent safety
mechanisms make the reactor completely self-limiting
with no damage for a 1% excess reactivity step and
probably for a 2% step. In addition, the public hazard
associated with an appreciably greater reactivity step is
extremely small. (auth)

3205 ANL-5731

Argonne National Lab., Lemont, Ill.

THE EBR-I MELTDOWN—PHYSICAL AND METALLURGICAL CHANGES IN THE CORE. Final Report—Metallurgy Program 7.2 18 J. H. Kittel, M. Novick, and R. F. Buchanan, Nov 1957 81p Contract W-31-109-eng-38 \$2 25(OTS)

As a result of the partial meltdown which occurred in EBR-I on November 29, 1955 it was necessary to remove the core assembly from the reactor and to separate the enriched fuel section from upper and lower unenriched blanket sections. A temporary cave was constructed on top of the reactor in order to remove the core assembly, and at this time about one-fourth of the fuel elements were removed. In order to perform further disassembly operations under less hazardous conditions, the core assembly was shipped from the Idaho Division of Argonne National Laboratory, at the National Reactor Testing Station, to the Lemont, Illinois, site of the Laboratory, where disassembly was completed in a protective atmosphere. It was found that approximately 40 to 50% of the core had melted and reached temperatures ranging between approximately 850 and 1400°C, and that the molten portion had separated into three clearly defined zones characterized by different porosities. Densities of the zones ranged from 2.5 to 15.4 gm/cm³, depending upon the degree of porosity. It was also found that molten fuel alloy had traveled upward 5 inches and downward 3 inches between the blanket rods. Chemical and mass spectrographic analyses indicated that relatively little mixing occurred in the core during the period in which it was molten, that the fuel alloy which penetrated the blanket sections originated primarily from outer part of the molten zone, and that the blanket did not enter the molten phase. Observations during disassembly of the core and subsequent simulated meltdown experiments indicated that the porous structure which formed in the molten core could have resulted from the vaporization of entrained NaK. (auth)

3206 ANL-5775

Argonne National Lab., Lemont, Ill.

THE COUPLED ASPECTS OF A FAST-THERMAL CRITICAL ZPR-V. B. J. Toppel, Oct. 1957. 20p. Contract W-31-109-eng-38. \$0.75(OTS).

ZPR-V is a zero-power, coupled fast-thermal critical assembly involving a dilute fast core and a H₂O-moderated, enriched U thermal annulus. The two regions are coupled through a natural U isolation blanket. This facility is useful in studying some of the properties of dilute fast reactor systems as well as those of coupled fast-thermal power breeders. Recent emphasis has been on the latter aspect of the problem and an investigation of the coupled aspects of the system has been made. An important parameter in the coupling concept is the division of reactivity. This can be determined by means of fast fuel substitutions and 1/v poisoning of the thermal annulus in conjunction with theoretical calculations. Auxiliary information obtained in such a procedure is the fast fuel worth throughout the core and the prompt neutron lifetime of the whole system. A second important parameter, S_1/S_2 , is the relative number of neutrons emitted per unit time due to fast and slow fissions. Information regarding this quantity can be obtained with the aid of U²³⁵ and U²³⁸ fission distributions. Measurements of the above parameters were made for the case of a U²³⁸:U²³⁵ atomic ratio of 5.1 in the fast core and a 1-in. isolation blanket. The fractions of reactivity in the fast core, blanket, and annulus are, respectively, 24, 14, and 61%; the prompt lifetime is 39.0 ± 3.0 μsec; the ratio S_1/S_2 is 0.43. (auth)

3238 KAPL-M-JBS-9

Knolls Atomic Power Lab., Schenectady, N. Y.

MULTI-GROUP ANALYSIS OF PuO₂ POWER BREEDERS;

PRELIMINARY RESULTS J. B. Sampson and E. A. Luebke Aug 17, 1954 Decl Oct 23, 1957 18p Contract W-31-109-Eng-52 \$3 30(ph OTS), \$2 40(mf OTS)

Several Pu-fueled power breeders have been analyzed by the multi-group method. The results indicate that a feasible reactor can be made having fissile and fertile material in the form of an oxide in both core and blanket. One possible reactor with 6 atoms U²³⁸ per Pu in the core and 43 volume % Na has a critical mass of about 450 kg, a breeding ratio of about 1.4 of which 0.66 is internal, and a power of about 800 Mw. Replacing the metal U blanket with U oxide results in no significant reduction in breeding ratio but makes possible more economical reprocessing of blanket material. Further calculations are planned (auth)

3866 ANL-5514(Del)

Argonne National Lab., Lemont, Ill.

REACTOR DEVELOPMENT QUARTERLY PROGRESS REPORT 4000 PROGRAM Stuart McLain, comp Oct 31, 1955 Decl Nov 18, 1957 50p Contract W-31-109-eng-52 \$0 45(OTS)

The status of Experimental Boiling Water Reactor development is reviewed. Shell testing, control mechanism development, fuel corrosion, and dimensional stability studies, and design consideration to facilitate D₂O operation are discussed. Fuel plate and subassembly fabrication, corrosion, and radiation damage studies were continued. Reactor water filtration, purification, and induced activity are considered. High-temperature operational tests were run on Experimental Breeder Reactor mechanisms. Though several minor malfunctions were encountered, the basic operation is considered satisfactory. Studies of the physical and metallurgical properties of U-Mo, U-Zr, U-Pu, and U-Pu-Mo alloys for fuel elements were conducted. Conversion ratio determination for the second loading of EBR-I gave a value of 1.03 ± 0.05. Alloys and systems were irradiated and the effects of the irradiation on corrosion determined. Included in the irradiation were U, U-Si, Nb-Sn, UO₂, and Al-Ni. (For preceding period see ANL-5476.) (D.E.B.)

4118 KAPL-1758

Knolls Atomic Power Lab., Schenectady, N. Y.

THE FAST OXIDE BREEDER—RADIATION EFFECTS IN CHEMICAL PROCESSING OF FAST OXIDE BREEDER FUEL. S. S. Jones, June 14, 1957. 38p. Contract W-31-109-eng-52. \$1.25(OTS).

Various ways in which KAPL Fast Oxide Breeder fuel processing may be affected by absorption of fission-product radiations are discussed. Results of experimental investigations are presented. While radiation will affect the process significantly, no prohibitive difficulties are known. However, amid the various complexities of such a system, only a mixer-settler run at full radiation level and solute concentrations can give a completely satisfying answer in a reasonable length of time. (auth)

4430 ANL-5719

Argonne National Lab., Lemont, Ill.

EXPERIMENTAL BREEDER REACTOR II (EBR-II), HAZARD SUMMARY REPORT L. J. Koch, H. O. Monson, D. Okrent, M. Levenson, W. R. Simmons, J. R. Humphreys, J. Haugsnes, V. C. Jankus, and W. B. Loewenstein May 1957. 401p Contract W-31-109-eng-38 \$6 50(OTS)

The EBR-II consists of an enriched core surrounded by a fertile blanket of depleted U. The fuel pins fit loosely in a thin walled tube filled with static Na for a heat transfer bond. Heat is removed by the primary Na flowing along the outside of the fuel tube. The fuel element lends itself to remote control fabrication methods. Effort was made to achieve a rigid, close-packed arrangement of the fuel and to produce preferential bowing such as to effect a probable

bowing coefficient which is essentially zero or negative. Reactor control is effected by movement of fuel into and out of the reactor core. Heat is removed by the primary Na coolant system and transferred to the secondary Na system in a shell-and-tube heat exchanger. The reactor and entire primary system, including heat exchanger, are contained in a large vessel (primary tank) and operate completely submerged in coolant. The large volume of Na in the primary system provides a reliable source of constant temperature coolant. Shutdown cooling is accomplished by natural convection. This bulk volume of Na is also used as coolant during reactor unloading. The reactor, primary coolant system, and all associated equipment are contained in a building in the form of a gastight cylindrical steel shell (with a normal safety factor of approximately four). Two potential sources of accidental energy release are treated separately. The major design and operating features of the EBR-II plant are summarized. (M H R)

4441 KAPL-1756(Pts I & II)

Astra, Inc., Milford, Conn

THE FAST OXIDE BREEDER-REACTOR ANALYSIS PART I NEUTRON YIELDS, CROSS SECTIONS, GROUP CONSTANTS, AND MACHINE ROUTINES G A Baraff and R B Mallon PART II REACTOR CALCULATIONS D F Molino and J K Davidson June 30, 1957 (Pt 1, 57p and Pt II, 35p) For Knolls Atomic Power Lab., Contract W-31-109-Eng-52, Subcontract K-314 (Pt 1, \$1 50(OTS), Pt II, \$1 00(OTS))

These two parts were issued separately, but are cataloged as a unit

Advanced Scientific Techniques Research Associates (ASTRA) has performed work for KAPL in the area of reactor physics with specific application to the KAPL Fast Oxide Breeder. The results of the first phase of this work, an evaluation of existing information on cross sections in the energy range of interest and on neutron yields for fissionable isotopes, are reported. The methods and techniques developed for reducing the raw data to group constants for PROD II (the code chosen for multigroup calculations) and modifications made to the PROD II code are discussed. The results of a new series of multigroup criticality calculations for the Fast Oxide Breeder reactor are reported. A new compilation of nuclear data prepared by ASTRA personnel was used as input. Among the more important differences between the new results and the original calculations of Luebke and Sampson are an increase in fuel concentration (U/Pu ratio from 5/1 to 4/1) for the same reactor size and power output, reduction in the total breeding ratio obtainable from about 1.4 to 1.3, and use of fuel movement rather than reflector movement for control. (auth)

4461 NDA-14-127

Nuclear Development Corp of America, White Plains, N Y

INTERPRETATION OF AN EXPERIMENT TO MEASURE DOPPLER EFFECT G Goertzel and C Klahr June 29, 1956 21p For Atomic Power Development Associates, Inc \$4 80(ph OTS), \$2 70(mf OTS)

An experiment was devised for measuring the Doppler reactivity effect in fast reactor mockups operating at zero power. Calculations are developed which relate the experimental reactivity values obtained in this manner to the actual Doppler temperature coefficient in a hot reactor operating at full power. (D E B)

4824 ANL-5717

Argonne National Lab., Lemont, Ill

METALLURGY DIVISION QUARTERLY REPORT [FOR] JANUARY, FEBRUARY, AND MARCH 1957 Mar 31, 1957 67p Contract W-31-109-eng-38 \$2 00(OTS) Advanced Water Reactor Program The second fuel loading for the EBWR is planned as $\text{ThO}_2\text{-UO}_2$. Some proper-

ties of $\text{ThO}_2\text{-UO}_2$ systems are presented. Fast Power Breeder Reactor Program. The development of casting techniques for EBR-1, Mark III extrusion billet cores are described. The investigation of methods for the remote refabrication of irradiated reactor fuels was undertaken with the objective of designing, constructing, and testing techniques and equipment for the refabrication of EBR-II fuels. Advanced Engineering and Development. From observations and photographs made during disassembly operations, and from chemical, metallographic, and density measurements, it was possible to reconstruct the probable physical structure of the core after the meltdown of the EBR-1 core. A boiling experiment was performed to give preliminary information as to whether thermal decomposition or volatility would complicate the use of H_3PO_4 solution in boiling reactors. Due to the possible use of Mg as a cladding material a few preliminary corrosion tests were made with distilled water. Cylindrical, NaK-filled Zr irradiation capsules, which suddenly broke open during decanning operations following irradiation, were examined in an effort to determine the role of irradiation in these unexpected failures. Production, Treatment, and Properties of Materials. Elastic constants of alpha-U single crystals in three directions were obtained. Recrystallization of heavily cold rolled U sheet was studied. Two ingots of high-purity U were used to study the melting point of U. The eutectic temperature measurements of U-UC alloy are presented. The $\alpha\text{-}\beta$ transformation temperatures of high-purity U were studied. Phase diagrams are presented for the U-Ru, U-Ru-Mo, and U-Mo alloys. (For preceding period see ANL-5709) (W L H)

5013 TID-7548

Argonne National Lab., Lemont, Ill., Los Alamos

Scientific Lab., N. Mex., and Power Reactor Development Co., Detroit.

FAST REACTOR PROGRAM INFORMATION MEETING, CHICAGO, ILLINOIS, NOVEMBER 20-21, 1957. Summaries of Papers. Nov. 20, 1957. 53p. \$1.50(OTS).

Papers are presented on: the ANL, LASL, and PRDC fast reactor programs, the engineering design of the Experimental Breeder Reactor-II- the reactor and power cycle and the fuel process and fabrications cycle; the design of the Enrico Fermi atomic power plant, fast reactor safety in the EBR-I; meltdown and analysis, and analysis of prompt excursions in simple systems, reactor design and analysis of LAMPRE I, EBR-II, and the Enrico Fermi Reactor; nuclear analysis by critical experiments with ZPR-III and static analysis, fuel element design; the EBR-II fuel cycle-fuel cycle plant and process and remote fuel element fabrication and processing; irradiation and thermal stability; Na technology, pumps, and piping system components; heat exchangers and steam generators; high purity Na systems; advanced Pu reactor concepts; fluid fuels; Pu-U alloy metal systems; fuel element concepts; fuel and blanket process development; and a coupled fast-thermal reactor. (M.H.R.)

5639 KAPL-1837

Knolls Atomic Power Lab., Schenectady, N. Y.

THE FAST OXIDE BREEDER-SINTERING STUDIES OF $\text{PuO}_2\text{-UO}_2$ J N Frank. July 15, 1957 26p Contract W-31-109-Eng-52 \$0 75(OTS)

The sintering characteristics of PuO_2 and UO_2 pellets were studied in a statistically designed experiment. An equation relating sintered density, weight loss or dimensional changes of the pellets with time of sintering, temperature, sieve size of powder material, and composition ratio was determined. When this equation is plotted against actual data, the density of the sintered pellets is predicted within a maximum error of 3% (auth)

5650 NAVORD-5747

Naval Ordnance Lab., White Oak, Md.

CONTAINMENT STUDY OF THE ENRICO FERMI FAST BREEDER REACTOR PLANT. E M Fisher and W R. Wise, Jr Oct. 7, 1957. 31p.

A study was made on the ability of the Enrico Fermi fast breeder reactor plant to contain a nuclear excursion equivalent to the violence produced by 1000 pounds of TNT. The results of the study indicate that the reactor plant can contain shock waves developed in the air and in the Na and also fragments from the cylindrical covering materials surrounding the reactor core. The rotating shield plug, however, is a serious hazard when projected by the gun action of the internal blast pressure in the reactor compartment. This problem is analyzed and a recommendation was made to lessen the danger. A greater hazard from the rotating plug can result from a slower (nonexplosive) energy release, however, it has not yet been determined by APDA whether a slower energy release as postulated in this report is possible (auth)

5676

ADVANCES IN NUCLEAR ENGINEERING. VOLUME I John R Dunning and Bruce R. Prentice, eds. New York, Pergamon Press, 1957. 530p.

Papers are presented on manufacture, production, recovery, and economics of nuclear fuels, spent fuel processing, plant containment concepts and design, plant components, waste disposal, protection and safety measures, and radiation processing. (M H.R.)

5677

ADVANCES IN NUCLEAR ENGINEERING. VOLUME II. John R Dunning and Bruce R Prentice, eds. New York, Pergamon Press, 1957. 587p

Papers are presented on reactor design, reactor core design, operation and maintenance of reactors, metallurgy, natural resources, instrumentation and control, coolant systems, heat transfer and heat evolution, educational uses of reactors, and standardization planning in the nuclear field. (M H R)

5922 IDO-14372(Del)

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.

EVALUATION OF ICPP FOR DETROIT-EDISON COMPANY FBR II FUEL PROCESSING. H. Schneider. Oct. 18, 1954. Decl. with deletions Feb. 25, 1957. 22p. Contract AT(10-1)-205. (CPP-54-97(Del.)). \$4.80(ph OTS); \$2.70(mf OTS)

The processing of a 2% Zr-U alloy type fuel is feasible. Recovery of core and blanket can be accomplished at rates compatible with reactor discharge rates with only minor process and equipment modifications and additions. Cost data are summarized (F S)

6218

COUPLED FAST-THERMAL POWER BREEDER Robert Avery (Argonne National Lab, Lemont, Ill) Nuclear Sci and Eng 3, 129-44(1958) Feb

Coupling a fast and thermal assembly in a power breeder reactor affords the possibility of obtaining the relatively long neutron lifetime characteristic of a thermal assembly and the high breeding ratio characteristic of a fast assembly. General properties of such mixed systems are discussed. A suggested design is discussed and compared with a prototype all-fast system. The coupled system considered consists of a 400-liter Pu^{239} fueled, Na-cooled, fast core surrounded by a 10-cm inner blanket annulus containing natural U, Na coolant, and structural material, but no moderator. Outside the inner blanket is a 30-cm annulus of Be surrounded by an outer blanket consisting primarily of depleted U. The inner blanket serves as core for the thermal system, as barrier for low-energy neutrons between moderator and fast core, and as reflector for the fast

core. Its construction is essentially the same as the first part of the blanket in a fast power breeder, so that the transition from an all-fast system to the coupled system involves only the replacement of blanket material by moderator and the use of natural rather than depleted uranium in the inner blanket. The properties of the system described are thereby changed: neutron lifetime increases from $\sim 1.5 \times 10^{-7}$ sec to $\sim 2 \times 10^{-5}$ sec, breeding ratio reduced $\sim 10\%$, fast core critical mass decreased $\sim 10\%$, multiplication constant of the system without the contribution of thermal fissions ~ 0.95 , thermal fissions generate $\sim 13\%$ of total power, and the radial power distribution in fast core flattened, maximum to average ratio reduced from ~ 1.5 to ~ 1.3 (auth)

6816

CF-55-5-103(Rev 2)

Oak Ridge National Lab, Tenn

A BIBLIOGRAPHIC FILE OF TECHNICAL REPORT ABSTRACTS USEFUL TO REACTOR CONTROL SYSTEM ENGINEERING Francis P Green Mar 1, 1958 18p Contract [W-7405-eng-26] \$0.50(OTS)

Report abstracts from more than 40 AEC and foreign facilities, AEC contractors, and technical periodicals are represented, published throughout the years 1941 to 1958. Cards for current publications are placed in the file no later than six weeks after the abstracts appear in the Abstracts of Classified Reports (auth)

6851

THE FAST REACTOR AT DOUNREAY H Grumm. Atomkern-Energie 3, 28-32(1958) Jan (In German)

The design, construction, and operation of the Dounreay Fast Reactor are reviewed (J S R)

6865

FAST-BREEDER POWER REACTORS WHERE DOES U S PROGRAM STAND? L J Koch (Argonne National Lab, Lemont, Ill) Nucleonics 16 No 3, 68-74(1958) Mar

The present activities in the U S program on fast breeder power reactors are outlined. The development of fuel materials and designs, reactor systems and components and fuel processing and fabrication are discussed. A brief discussion of the fast reactor projects under construction—the Experimental Breeder Reactor-II, Los Alamos Molten Plutonium Reactor Experiment-I, and the Enrico Fermi Fast Breeder Reactor—is given. Brief paragraphs are devoted to a comparison of the nuclear parameters for fast and thermal power reactors and to the U K fast reactor program (J S R)

7026

FAST NEUTRONIC REACTOR A H Snell (to U S Atomic Energy Commission). U S. Patent 2,815,319 Dec. 3, 1957

This patent relates to a reactor and process for carrying out a controlled fast neutron chain reaction. A cubical reactive mass, weighing at least 920 metric tons, of uranium metal containing predominantly U^{238} and having a U^{235} content of at least 7.63% is assembled and the maximum neutron reproduction ratio is limited to not substantially over 1.01 by insertion and removal of a varying amount of boron the reactive mass being substantially freed of moderator

7055

IMPROVEMENTS IN OR RELATING TO NUT AND SCREW MECHANISMS J A F Glass (to United Kingdom Atomic Energy Authority) British Patent 786,486 May 11, 1954 Nuclear Power 3, 442(1958) Mar

In the published descriptions of the Dounreay Fast Reactor it was not made particularly clear how the reactivity control system operates because the information was withheld by the AEA at the time. Now a patent is published

which seems to suggest how it is done. As disclosed last year, control of the DFR is by withdrawal from below the unmoderated core of certain fuel elements, the mechanism for doing this being partly immersed in the molten sodium primary coolant. The object of this invention, one form of which is probably used at Dounreay, is to provide in one unit both fine control and emergency control and the specification describes a nut and screw mechanism driven through a magnetic clutch. Fine control is obtained by moving the fissionable material into or out of the core by the screw mechanism. When scram action is required, de-energizing the clutch sends the movable elements out of the core under gravity. The invention makes use of a 'ball-bearing' thread (British Patent 707,D17) (auth)

7454 AECU-3648

Detroit Edison Co., Detroit

THERMAL CYCLE TEST OF AN APDA FUEL PIN
Engineering Lab. and Research Dept. Report 56E30 J. J. Bodzin, F. T. Rath and C. A. Anderson Jan 31 1957
32p Contract [AT(11-1)-476] \$6.30(ph OTS), \$3.00(mf OTS)

An investigation to determine the effect of thermal cycling in sodium of a fuel pin is reported. The pin submitted for the test was composed of uranium-2 wt % zirconium sheathed in zirconium. The test consisted of the following thermal cycle: heating to 1250°F, holding 5 min then cooling to 250°F and holding at 250°F to prevent the freezing of the sodium. The duration of the test was 25 cycles. At the completion of the thermal cycling the pin was examined visually to determine whether any physical damage was evident and what dimensional and metallurgical changes had occurred. (auth)

7455 AECU-3650

Detroit Edison Co., Detroit

THERMAL CYCLE TEST IN SODIUM OF AN APDA FUEL PIN
Engineering Lab. and Research Dept. Report 57CO5-1 J. J. Bodzin and T. A. Alessi June 13, 1957
25p Contract [AT(11-1)-476] \$4.80(ph OTS), \$2.70(mf OTS)

An investigation was undertaken to determine the effect of thermal cycling in sodium of various fuel pins. The pin submitted for this test was a depleted uranium-10 wt % molybdenum fuel pin sheathed with sponge zirconium commercial grade. The pin was not end capped. The test consisted of the following thermal cycle: heating to 1375 ± 25°F, holding, cooling to 250 ± 25°F. The temperature of the test specimen exceeded 1200°F for at least 30 min for each cycle. At the conclusion of 25 cycles, the pin was examined visually to determine whether any physical damage was evident and what dimensional and metallurgical changes had occurred. (auth)

7456 AECU-3658

Allis-Chalmers Mfg. Co., Milwaukee.

FUEL ELEMENT SUBASSEMBLY NOZZLE TEST IN HOT SODIUM FOR ATOMIC POWER DEVELOPMENT ASSOCIATES [1957], 17p, 2 illus. (ACNP-5728) \$4.80 (ph OTS), \$2.70(mf OTS).

A test to determine the effect of rubbing contact on various surfaces of a fuel element nozzle when submerged in 750°F Na is described. (M.H.R.)

7458 ANL-5843

Argonne National Lab., Lemont, Ill

TEMPERATURE COEFFICIENT MEASUREMENTS ON A CRITICAL MOCK-UP OF EBR-I John K. Long Mar 1958 28p Contract W-31-109-eng-38 \$1.00(OTS).

A rough mock-up of the EBR-I reactor was built in the ZPR-III critical assembly machine, with resistance heaters in the core and inner blanket. The variation of reactivity with core temperature was observed. Lateral expansion

was measured as the core was heated. A temperature coefficient of -0.42 inhour per degree Centigrade was observed when the core alone was heated, and -0.72 inhour per degree Centigrade was observed when the core and inner blanket both were heated. Essentially all of these coefficients are attributable to volume expansion. (auth)

7460 APDA-120

Michigan Univ., Ann Arbor. Engineering Research Inst. REPORT ON THE POSSIBLE EFFECTS ON THE SURROUNDING POPULATION OF AN ASSUMED RELEASE OF FISSION PRODUCTS INTO THE ATMOSPHERE FROM 300 MEGAWATT NUCLEAR REACTOR LOCATED AT LAGOONA BEACH, MICHIGAN. Henry J. Gomberg, Thomas Bassett, and Carlos Velez. Foreword by Alton P. Donnell. July 1957. 109p Project 2506. For Atomic Power Development Associates, Inc. \$16.80(ph OTS), \$5.70(mf OTS).

7461 APDA-123

Atomic Power Development Associates, Inc., Detroit. RESEARCH AND DEVELOPMENT ACTIVITIES QUARTERLY REPORT [FOR] APRIL-JUNE 1957. 17p \$3.30 (ph OTS), \$2.40(mf OTS).

A general review of Fermi Fast Breeder Reactor research and development is presented. Attention is directed primarily to the development of less expensive shielding material and more satisfactory fuel materials and fuel pin fabrication methods. (D.E.B.)

7465 CF-57-8-14

Oak Ridge School of Reactor Technology Tenn. FLUIDIZED BED REACTOR STUDY Reactor Design and Feasibility Study C. L. Teeter, T. Ciariariello, B. L. Hoffman, D. H. Jorgensen, F. D. Judge, L. J. King, D. A. McCune, N. R. Scheve and H. E. Zellnik Aug 1957 203p \$31.80(ph OTS), \$9.30(mf OTS)

A feasibility study of a fluidized bed reactor system was undertaken. Initial studies were conducted on a fast, sodium cooled system, a gas cooled system, and organic and water moderated thermal systems. The organic moderated and light water moderated systems showed promise. The light water system was chosen for a detailed study to enable comparison with existing pressurized water reactors. The detailed study resulted in the selection of a slightly enriched uranium reactor with a heat production capability of 465 Mw. If the plutonium produced is recycled, natural or depleted material can be utilized as feed. The economic study indicated fuel costs between 1.8 and 2.7 mills/kw-hr assuming a plant load factor of 0.85. (auth)

7475 NDA-27-16

Nuclear Development Corp. of America, White Plains, N. Y.

SUMMARY OF APDA FAST REACTOR SAFETY STUDIES W. H. Jens Oct 1956 34p \$0.30(OTS)

In the evaluation of the safety of a reactor system it is usual to hypothesize many series of events which would lead to serious accidents. These events are then analyzed to determine quantitative results. The assumptions, important considerations, techniques, and results of a number of such studies which were made for the APDA fast breeder reactor are described. In general the initial assumption is made that reactivity is inserted at a linear rate into a reactor that has a neutron source and a negative temperature coefficient. (auth)

7838 BW-7065

Babcock and Wilcox Co. Research Center Alliance, Ohio. SODIUM BONDING OF APDA BLANKET ELEMENTS Paul C. Thys and Dale W. Koch Nov 29, 1957 29p \$4.80(ph OTS), \$2.70(mf OTS)

A method was developed for satisfactorily bonding stainless steel cladding to uranium rod for use in the Fermi Fast Breeder Reactor blanket. The process is described, equipment used is shown, and a production flow diagram is proposed. (D.E.B.)

8048 AECU-3652, AECU-3653, and AECU-3654 Atomic Power Development Associates, Inc., Detroit. SUMMARY REPORTS—APDA ACTIVITIES [FOR] JULY, AUGUST, SEPTEMBER 1957. (July, 9p., Aug., 10p.; and Sept., 11p.) (AECU-3652 and AECU-3653 \$1.80(ph OTS); \$1.80(mf OTS) ea. and AECU-3654 \$3.30(ph OTS); \$2.40(mf OTS)).

These three reports were issued separately, but are cataloged as a unit.

Brief summaries of Fermi Fast Breeder Reactor design, development, and component procurement progress are given. Pressure vessel fabrication and fuel pin testing are discussed. (For preceding period see APDA-315.) (D.E.B.)

8077 PRDC-TR-4 PRDC-TR-5, and PRDC-TR-6

Power Reactor Development Co., Detroit. MONTHLY TECHNICAL REPORTS REPORT PERIOD, OCTOBER, NOVEMBER, DECEMBER 1957 (Oct, Dec 18, 1957 12p, Nov 10p, and Dec 12p) Contract AT(11-1)-476 (PRDC-TR-4 and PRDC-TR-5, \$3.30(ph OTS), \$2.40(mf OTS) ea; PRDC-TR-6, \$4.80(ph OTS); \$2.70(mf OTS))

These three reports were issued separately, but are cataloged as a unit

Fermi Fast Breeder Reactor design and major component fabrication and procurement are reviewed. Brief statements on fuel pin testing are included (For preceding period see AECU-3654) (D.E.B.)

8078 PRDC-TR-7

Power Reactor Development Co., Detroit. MONTHLY TECHNICAL REPORT: REPORT PERIOD, JANUARY 1958. 10p. Contract AT(11-1)-476. \$1.80(ph OTS); \$1.80(mf OTS).

Fermi Fast Breeder Reactor design progress and major component fabrication status are summarized. (For preceding period see PRDC-6.) (D.E.B.)

8079 PRDC-TR-8

Power Reactor Development Co., Detroit. MONTHLY TECHNICAL REPORT: REPORT PERIOD, FEBRUARY 1958. 15p. Contract AT(11-1)-476. \$3.30(ph OTS), \$2.40(mf OTS).

Design data for the fuel element transfer cask are given. Design progress for the Fermi Fast Breeder Reactor is briefly summarized, and major component fabrication is reviewed. Results of fuel pin testing are discussed. (For preceding period see PRDC-7.) (D.E.B.)

8709 AECU-3678

Atomic Power Development Associates, Inc., Detroit. REPORT ON CORE AND BLANKET DESIGN AND TECHNOLOGY C M Ladd, R J Hennig, O E Homeister, D O Leaser, R A MacLeod, R G Rateick, and L P VanMaele Jan 1, 1958 45p \$7.80(ph OTS), \$3.30(mf OTS)

A summary is presented of the information developed to date on the core and blanket fuel materials for the Fermi Fast Breeder Reactor. The specification requirements for the core and blanket subassembly fabrication are discussed, along with special materials accountability and shipping problems (W L H)

8713 ANL-5836

Argonne National Lab, Lemont Ill.

EBR-I MARK III—DESIGN REPORT R E Rice R N Curran F D McGinnis M Nevick and F. W Thalgett Mar 1958 42p Contract W-31-109-eng-38 \$1.25(OTS).

Under severe operating conditions, the Experimental Breeder Reactor was observed to exhibit oscillatory power characteristics and periods of prompt positive reactivity. Investigations of these characteristics resulted in the meltdown of Core-II and the design of a more suitable Core-III with rod bowing eliminated and greater coolant flow control. The design of the core rods, assemblies and matrices is presented, and its control and cooling systems are described (D E B)

8735 NDA-14-182

Nuclear Development Corp of America, White Plains, N Y

CALCULATED CRITICALITY OF THE ENRICO FERMI REACTOR DURING A HYPOTHETICAL SLOW MELTDOWN ACCIDENT J H Ray and A. D. Krumbein July 1, 1957 22p \$4.80(ph OTS); \$2.70(mf OTS)

The progress of a slow meltdown accident in the Enrico Fermi Reactor has been calculated, and a series of two-dimensional multigroup calculations has been performed to ascertain the time at which the melted reactor forms a critical configuration. The results of this series, as modified by the procedure described in this report, show that the melted configuration becomes critical approximately 5.2 sec after initiation of the accident. The rate of reactivity insertion at this time is approximately \$26 per sec. It is noted that instantaneous flow of molten uranium from the core and subsequent refreezing at the top of the lower blanket have been assumed. The molybdenum in the core was not assumed to melt with the uranium. These assumptions while not the worst that can be imagined, are sufficiently pessimistic so that a more realistic calculation of the progress of melting is expected to lead to a lower reactivity insertion rate (auth)

9468

POWER REACTOR TECHNOLOGY Technical Progress Review, Vol. 1, No. 2. Dunedin, General Nuclear Engineering Corp., 1958. 72p. Available from U. S. Government Printing Office, Washington for \$0.55.

Pertinent data from current studies in reactor development are reviewed. The subjects covered are: reactor physics, reactor dynamics and safety, fuel elements and other reactor materials, fuel cycles, Pu recycle, fluid fuels, UO_2 -NaK slurry, components, aqueous homogeneous reactors, pressurized water reactors, boiling water reactors, fast reactors, organic moderated reactors, gas cooled reactors, and molten fluoride reactors. Extensive bibliographies are included. (M.H.R.)

10083 WAPD-P-145(Rev.)

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

MINUTES OF ADVANCED REACTOR TECHNICAL MEETING NO. 8. Robert J. Creagan, ed. Nov. 15, 1951. 17p. \$3.30(ph OTS); \$2.40(mf OTS).

Conservation of D_2O in a heterogeneous advanced reactor was discussed on the basis of losses scaled up from those predicted for similar reactors in the past. The conclusion was that losses could be reduced to 0.05% of those predicted, by the use of reasonable equipment. This could mean a net loss of 0.1 pounds per hour. The over-all heat power considerations for using liquid metal and aqueous coolants for reactors were compared. It was concluded that over-all efficiencies could be expected as follows: 20% for liquid metal coolant, 14 to 17% for a water cooled reactor. Plant layout comparisons were discussed for the case of liquid metal and aqueous type reactors and it was concluded that while the liquid metal temperature reduces the size of

the steam equipment, the shielding requirements make the liquid metal power plant heavier than an aqueous reactor to furnish the same power. The shielding problems associated with aqueous and sodium cooled reactors were compared and it was concluded that the greater specific activity in the sodium coolant required more shielding for the primary loop than in the case of aqueous coolant. One result of the primary loop activity is to preclude maintenance inside the secondary shield for minutes in the case of one previous design and for weeks in the case of another previous design. Reactor calculations of various liquid metal reactors were discussed and it was concluded that a sodium cooled, graphite moderated thermal reactor would be the most practical of this type with slightly enriched uranium used to reduce the size. Fuel costs were discussed and it was concluded that for similar fuel elements and the same burn up, fuel costs per kwh of power were higher in the case of highly enriched reactors. It was further concluded that if ORNL-1096 cost figures are correct the fuel cost figures given on page 6 of WAPD-P-130 are high. The new estimates are tabulated. A visit to Argonne, during which work on fuel plates for an aqueous heterogeneous reactor was observed, was reported on. The conclusion was that plates are not as feasible as rods for a fuel element. (auth)

10096

INSTRUMENTATION, 1958 Nuclear Power 3, 148-86 (1958) Apr

An over-all picture of the instrumentation requirements of modern nuclear power and research reactor installations and a review of the latest British instruments available to meet them are given. Five review articles by the country's leading authorities are included. These cover physical or non-nuclear instruments needed in a power station, lines of development for instrumentation, methods of neutron detection by means of gas ionization devices, scintillation counting methods, and methods of finding neutron flux variations. Diagrams are included showing the main instrumentation features in a gas-cooled power station, a high-flux reactor, and a fast breeder reactor. Preview features of two exhibitions in London—the Physical Society and the Instruments, Electronics, and Automation exhibitions—are included. (M H R)

10968 AECU-3697

ASTRA, Inc., Milford, Conn.

THE EFFECT OF SODIUM VAPOR FORMATION ON THE OPERATION OF THE ENRICO FERMI REACTOR—AN ESTIMATE. H R Kroeger and J W Stephenson. July 1957. 53p. (ASTRA-471-E-1.1) \$9.30(ph OTS), \$3.60(mf OTS)

The results of calculations concerning two-phase sodium flow in the Fermi Fast Breeder Reactor are presented. The large specific volume of sodium vapor coupled with the resistance of the top axial blanket section resulted in large pressure drops for small values of net vapor. Once the coolant reached the saturation temperature the pressure drop resulting from vapor formation exceeded the added pressure drop from flow reduction. Although there appears to be a region of stable operation with vapor formation, the lowest value of pressure drop in this region is much greater than that at the initiating flow rate. It is concluded that any vapor formation in the reactor produces an unstable condition which would lead to burnout. (D E B)

11032

CONSIDERATIONS IN THE DESIGN OF THE NUCLEAR SAFETY INSTRUMENTATION SYSTEM FOR THE ENRICO FERMI ATOMIC POWER PLANT. Presented

at Nuclear Engineering and Science Conference held at Chicago, March 17 to 21, 1958. Preprint 42, Session 21. V S Underkoffler and R G Olson (Leeds and Northrup Co., Philadelphia and Atomic Power Development Associates, Inc., Detroit). New York, American Institute of Chemical Engineers, 1958. 21p.

The conceptual design of the nuclear safety instrumentation system for a central station nuclear power plant provides considerable insight into the complexity of the instrumentation requirements for such a system. The proposed safety instrumentation system for the Enrico Fermi Atomic Power Plant is used as an illustration of this complexity. Emphasis is placed on the safety instrumentation and its associated equipment. The design is traced through its many aspects and the problems encountered are indicated. The supporting experimental and simulation programs are briefly discussed to further illustrate the vast amount of information required for the conceptual design. (auth)

11941

DESIGN OF EBR-II PRIMARY TANK SUPPORT STRUCTURE. Presented at Nuclear Engineering and Science Conference, held at Chicago, March 17 to 21, 1958. Preprint 73, Session 36. R W Seidensticker and S H Fistedis (Argonne National Lab., Lemont, Ill.). New York, American Institute of Chemical Engineers, 1958. 15p.

The EBR-II primary tank support structure consists of a system of columns and beams which transmit the loads incurred on the structure of the main internal building foundation. In combination with the biological shield, it forms a "pressure vessel" surrounding the primary tank which contains the reactor and all of the primary system components. The bottom "closure" of the "pressure vessel" is the internal building foundation, which is mass concrete reinforced with steel bars to act as a pressure vessel head. The material proposed for the columns and beams is a high strength steel, USS Carilloy T-1, which has a yield strength of 90,000 psi and a tensile strength of 105,000 psi at room temperature. Since the primary tank contains approximately 600,000 pounds of sodium at 700°F, it is imperative that the primary tank support structure contain this sodium in the event of a postulated explosive-type nuclear incident in order to avert the secondary, and undoubtedly more severe effect the sodium-oxygen reaction in the reactor building. As a consequence of this requirement, the structure is designed not in the conventional manner of proportioning members to resist static gravity loads, but to provide sufficient strength to safely resist high pressures of short duration and to some extent, to provide a blast-resistant structure in order to preserve the integrity of the "pressure vessel" enclosure. (auth)

11946

MODEL STUDIES OF THE FLOW CHARACTERISTICS OF THE FUEL ELEMENTS FOR THE ENRICO FERMI REACTOR. Presented at Nuclear Engineering and Science Conference, held at Chicago, March 17 to 21, 1958. Orville F Homeister, Weymouth N. McDaniel, and Raymond A Yagle (Atomic Power Development Associates, Inc., Detroit and Univ. of Michigan, Ann Arbor). New York, American Institute of Chemical Engineers, 1958. 21p.

Laboratory measurements were made on radial blanket subassemblies of the type proposed for use in the Enrico Fermi nuclear reactor to determine head losses, correct sizes for distributing orifices, and fluid mixing characteristics. A coefficient of eddy diffusivity was measured for the complex geometry so that the distribution of temperature in the proposed nuclear reactor

could be estimated more accurately. Head loss measurements were also made on various core subassemblies representing five different arrangements or methods of spacing the small diameter core fuel pins. Several possible methods of reducing these losses were also investigated. For both the radial blanket and the core subassemblies, the flow is parallel to the axis of the rods. The observed losses were generally greater than values predicted with conventional methods. Tests with water were generalized so that the results provide design criteria for liquid sodium which will be used in the reactor. (auth)

11950

POWER REACTOR TECHNOLOGY. Technical Progress Review, Vol. 1, No. 3. Dunedin, General Nuclear Engineering Corp., 1958. 63p. Available from U. S. Government Printing Office, Washington for \$0.55.

Buckling measurements are presented for D_2O moderated graphite lattices. The effects of changes in temperature on the reactivity of graphite lattices are presented. The conversion ratio of the PWR blanket is discussed. The results of capture-to-fission ratio measurements in the EBR-1 are reported. Burnout measurements and heat transfer under supercritical conditions are presented. Transient boiling experiments and dynamics of boiling water reactors are reported. Reactor fuel elements consisting of U and U alloys of high U content, plutonium-aluminum, non-metallic elements of high fuel content, and elements of low fuel content are discussed. Control, moderator, shielding, and structural materials for applications in various reactors are presented. Progress is reported on liquid metal fuel reactors and fast reactors. (W.L.H.)

12040 AECU-3701

[Atomic Power Development Associates, Inc., Detroit.] **SHIELDING OF A FAST BREEDER REACTOR—I. THE PRIMARY SHIELD.** H. E. Hungerford. [1957]. 19p. \$3.30(ph OTS); \$2.40(mf OTS).

The differing shielding requirements of a fast breeder reactor and a thermal reactor are discussed. The functions of a stainless steel layer inside the reactor vessel of a fast breeder are outlined: neutron reflector, gamma-ray absorbing thermal shield, and an inelastic neutron scatterer to protect the vessel walls against radiation damage. The use of borated graphite outside the vessel is discussed. The rotating plug shield is described. Shielding calculations and a primary shield design drawing are given for the Fermi Fast Breeder Reactor. (M.H.R.)

12685 AECU-3757 (Pts. I and II)

Michigan. Univ., Ann Arbor. Engineering Research Inst.

TESTS ON MODELS OF NUCLEAR REACTOR ELEMENTS. I. HEAD LOSSES IN BLANKET SUB-ASSEMBLY. J. S. McNown, R. A. Yagle, and W. W. O'Dell. Mar. 1956. 20p. **II. STUDIES OF DIFFUSION.** J. S. McNown, C. S. Yih, R. A. Yagle, and W. W. O'Dell. Mar. 1957. 15p. For Atomic Power Development Associates, Inc. (ERI-2031-1-P and ERI-2031-2-P). \$3.30(ph OTS); \$2.40(mf OTS) ea.

These two parts were issued separately, but are cataloged as a unit.

For a model of a blanket sub-assembly of the proposed Atomic Power Development Associates reactor, laboratory measurements have been made of the head losses for the blanket element and for the various distributing orifices. Tests with water have been generalized so that the results provide design criteria for liquid sodium which is to be used in the reactor. (auth)

12716 COO-240

Naval Research Lab., Washington, D. C. **PROGRESS REPORT ON FAST BREEDER REACTOR SAFETY STUDIES.** L. A. Beach, A. G. Pieper, and M. P. Young. [1957]. 26p. Contract [AT(11-1)-607]. (NRL-Memo-803). \$4.80(ph OTS); \$2.70(mf OTS).

A proposal was made to ANL for NRL to construct a gamma ray pin-hole camera to follow the flow of molten fuel elements by observing the emitted gamma rays with a TV chain viewing an image of the fuel elements on a NaI screen. However, an alternative system has been proposed that will give the same space resolution, improved time resolution and dynamic range with greater flexibility and reduced cost. This system consists of a rotating cylinder with a large number of small collimating holes scanning across the test section. Signals from scintillation detectors placed behind the drum will intensity modulate the Z axis of multi-gun oscilloscopes. A continuously moving film camera will record these sweeps for study and analysis. (auth)

12723 KAPL-1611(Del.)

Knolls Atomic Power Lab., Schenectady, N. Y. **REPORT OF THE PHYSICS SECTION FOR JUNE, JULY, AUGUST 1956.** Decl. with deletions Nov. 21, 1957. 42p. Contract W-31-109-Eng-52. \$7.80(ph OTS); \$3.30(mf OTS).

Resonance integrals were calculated for U^{235} fission. Experimental results on the measurement of the capture-to-fission ratio of U^{235} are given. Features of the magnetic alpha analyzer are briefly described. Experiments with the 3-stage mass spectrometer are described. A mass spectrometric analysis of millimicrogram samples of Pu is shown. Lattice parameter measurements on irradiated and unirradiated cubic crystals are summarized. A theoretical study of x-ray scattering by irradiated crystals is included. A study was also made of the fast neutron irradiation of semiconductors. Changes in carrier concentrations of Ge materials exposed to TTR neutrons are tabulated. A further study was made of the pulses produced by 5-Mev α particles in gold-doped Si. The breeding possibilities calculated for the KAPL Fast Oxide Breeder were shown to be consistent with reactor measurements of the neutron economy of Pu^{239} made in the EBR. (B.J.H.)

12724 KLX-1782

Vitro Engineering Co., New York. **INERT GAS SYSTEM HAZARDS ANALYSIS FOR ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.** Vitro Job No. 1221. R. Epstein and L. Geller. Apr. 25, 1957. 29p. \$3.30(ph OTS); \$2.40(mf OTS).

The radioactivity of the inert gas system of the Enrico Fermi Fast Breeder Reactor Plant was analyzed under normal operating conditions, after an instantaneous fission product release from 10% of a subassembly, and during a continuous fission product release at 10% of the production rate. (auth)

12729 NDA-14-153

Nuclear Development Associates, Inc., White Plains, N. Y. **SPHERICAL AND CYLINDRICAL ADJOINT CALCULATIONS FOR THE APDA FAST BREEDER REACTOR.** Leon Davidson and Carl Klahr. Nov. 14, 1956. 43p. For Atomic Power Development Associates, Inc. \$7.80(ph OTS); \$3.30(mf OTS).

The fluxes and adjoints for the Fermi Fast Breeder Reactor as functions of position and energy are presented in graphical form. The results were obtained from one-dimensional multigroup calculations on

UNIVAC in spherical and cylindrical representations of the reactor. The graphs give results for groups 1,3,5, 7,9 at all space points (W D M.)

12730 NDA-14-170

Nuclear Development Corp. of America, White Plains, N. Y.

THE DISTRIBUTION OF MELTING IN A SLOW MELT-DOWN ACCIDENT. H G Elrod, May 1, 1957. 28p. For Atomic Power Development Associates, Inc. \$4.80(ph OTS), \$2.70(mf OTS).

Revised edition of NDA-Memo-14-146.

Melting distributions are given for an hypothesized slow melt-down of the APDA reactor in the absence of sodium. Anticipated distributions of melting at various times after the start of the accident are computed on the assumption that the core retains its integrity. According to this hypothesis, melting starts at the core center 3.5 sec after initiation of the accident, and at the lower edges of the core after 22 sec. A maximum rate of melting of 0.46 ft³/sec is achieved over the central period of the melt-down. If the fuel distribution is altered because of liquid run-off, the melting configurations of this report will be approximately reached at comparable values of integrated core power (auth)

12733 PRDC-TR-10

Power Reactor Development Co., Detroit

MONTHLY TECHNICAL REPORT [FOR] APRIL 1958 11p Contract AT(11-1)-476 \$1.80(ph OTS), \$1.80(mf OTS)

Thermal and biological shielding requirements are reviewed. A reevaluation of the delayed neutron fraction was made to include those neutrons produced in the blanket. Calculations of the decay heating due to heavy isotopes have been completed. The maximum permissible concentration in air of a mixture of fission gases was calculated to be 6×10^{-7} $\mu\text{c/cc}$. Ten specimens of U-10 wt % Mo alloy were studied for cracking and splitting after irradiation to 2.5 at % burnup. Two cermet specimens (27 wt % UO_2 dispersed in a matrix of U-10 Mo) exhibited no visual indication of cracking or swelling after irradiation to 3.5 at % burnup. Latest heat generation data for core performance of 300 Mw indicate a central metal temperature of 1133°F and 1281°F in the hottest subassembly, and for blanket performance at 300 Mw total power and 0.5 at % blanket burnup central metal temperatures in excess of 1000°F for part of the subassemblies located in the high-pressure plenum. Thermocouples were tested for sodium leaks and grounding. The sealing properties of sodium and "Cerrobend" are being tested. (For preceding period see PRDC-TR-9) (W D M.)

12734 PRDC-TR-11

Power Reactor Development Co., Detroit

MONTHLY TECHNICAL REPORT [FOR] MAY 1958 10p Contract AT-(11-1)-476 \$1.80(ph OTS), \$1.80(mf OTS)

The status of the design and fabrication of major components being procured for the Reactor Components Test is shown. Progress in nuclear engineering materials, fuel engineering, reactor vessel engineering, mechanical handling, liquid metal and steam system, and test operations are summarized. (For preceding period see PRDC-TR-10) (M H R.)

12767

NUCLEAR POWER PLANTS IN U.S.S.R. E. P. Anan'ev. *Vestnik Akad. Nauk S.S.S.R.* 28, No. 3, 1-14(1958) Mar. (In Russian)

The design and performance of a water moderated reactor is described, and diagrams of power plants

powered by graphite moderated, boiling water, and fast reactors are presented. Analysis is made of the performance and economic factors of the various reactor types (R V.J.)

12780

POWER REACTORS. Washington, Atomic Energy Commission, 1958. 96p. Available from U. S. Government Printing Office, Washington for \$0.65.

A compact summary of technical information on 10 full-scale power plants built, under construction, or being undertaken in the U. S. is presented. These represent six different types of reactors: pressurized water, boiling water, sodium graphite, organic moderated, fast breeder, and sodium heavy-water. Sketches and pertinent data are given for each of the 10 power plants. The Consolidated Edison Thorium Reactor (Indian Point Power Reactor) is a pressurized-water uranium-thorium 275 Mwe plant. The Yankee Power Reactor is a 134 Mwe, pressurized-water, UO_2 -fueled plant. The Army Package Power Reactor is a 2 Mwe pressurized-water reactor. The Vallecitos Boiling Water Reactor is a 6 Mwe, UO_2 -fueled plant. The Dresden Power Reactor is a 180 Mwe, UO_2 -fueled, boiling water reactor. The Northern States Nuclear Power Plant (Sioux Falls Power Reactor) is a 66 Mwe UO_2 -fueled, boiling water reactor. The Hallam Power Reactor is a 75 Mwe, sodium-cooled, graphite-moderated, enriched uranium reactor. The Organic Moderated Reactor Power Plant (Piqua Power Reactor) is a 12.5 Mwe, organic moderated and cooled, enriched uranium reactor. The Fermi Fast Breeder Reactor is a 90 Mwe, sodium cooled, enriched uranium reactor. The Chugach Power Reactor is a 10 Mwe, sodium cooled, heavy water moderated, enriched uranium reactor. (M H R.)

12781

REACTOR CORE MATERIALS. Technical Progress Review, Vol. 1, No. 2. Columbus, Battelle Memorial Institute, 1958. 47p. Available from U. S. Government Printing Office, Washington for \$0.55.

The second of a series of quarterly publications covering significant developments in the field of solid reactor core materials is presented. It is a compilation of reviews of specialized topics. The purpose of the review series is to provide a convenient condensed source of information. Complete bibliographies are provided at the end of each section. Coverage is limited to those core materials which are solid at reactor operating temperatures. Fuel and fertile materials, moderator materials, control materials, cladding and structural materials, and special fabrication techniques are considered (W D.M.)

12863 BMI-APDA-622

Battelle Memorial Inst., Columbus, Ohio

A STUDY OF THE PRIMARY SHIELD FOR THE PRDC REACTOR. Harold M. Epstein, David A. Dingee, and Joel W. Chastain. Apr. 15, 1957. 34p. For Atomic Power Development Associates, Inc. \$6.30(ph OTS); \$3.00(mf OTS).

Temperature distributions, irradiation effects, stacking arrangements, voidage, and economics for the borated-graphite shield of the PRDC reactor were investigated. Of the shield systems considered, four are reported here. System 1 contains 30 in. of 1% borated graphite, with either ordinary graphite or a cement as a filler for the remainder of the volume. The maximum temperature at the flex plates in this system was calculated to be 500°F. Systems 2 and 3 consist of 2 in. of 5% borated graphite near the core vessel and 1/2 in. of Boral at the primary-shield tank. A filler material

of carbon blocks is used in System 2 and graphite in System 3. The calculated maximum temperatures were 700°F and 350°F, respectively. System 4 consists of a laminated structure of Boral and graphite near the primary-shield tank and carbon-block filler. It was calculated to have a maximum temperature of 600°F at the flex plates. The maximum temperature at the flex plates recommended by APDA is 500°F. Energy storage and radiation damage were found to be within permissible limits in all four systems. However, these conclusions are based on experimental data from the Hanford reactor in which the neutron-energy spectrum differs considerably from the PRDC spectrum. A porosity of less than 740 cu ft is required in order that a sodium leak from the core vessel does not expose the core. The voidages in any of the systems mentioned above is about 400 cu ft excluding absorption effects. These are believed to be small. The systems containing Boral were found to be less expensive than the ones using only borated graphite. Over-all material costs range between \$230,000 for Boral systems and \$350,000 for borated-graphite systems (auth).

12894 NDA-Memo-14-65

Nuclear Development Associates, Inc., White Plains, N. Y.

PROBLEM OF LOW MELTING POINT EUTECTICS AT FUEL ELEMENT INTERFACE. R. A. Daane. Jan 14, 1955. Dec 1 July 7, 1958 9p.

Work performed under contract with the Detroit Edison Co. and the Dow Chemical Co.

The effects of low-melting-point U-Fe or U-Ni eutectics being formed at the interface of U-Cr and U-Zr fuel alloys in the proposed APDA reactor are discussed. From the temperature distribution it appears that the low-melting-point eutectics at the fuel element interface will not cause any damage to the reactor fuel elements. (R.V.J.)

13474 NDA-014-167

Nuclear Development Corp. of America, White Plains, N. Y.

DANGER COEFFICIENTS FOR THE ENRICO FERMI FAST BREEDER REACTOR. Carl Klahr. Sept 3, 1957 23p. For Atomic Power Development Associates, Inc. \$3.30(ph OTS); \$2.40(mf OTS).

Danger coefficients are presented for eleven materials of interest in the Fermi Fast Breeder Reactor— U^{235} , U^{238} , Pu^{239} , B^{10} , stainless steel, Na^{23} , C^{12} , Zr, Mo, and fission products. The reactor was mocked-up by one-dimensional spherical and cylindrical representations. Danger coefficients were calculated by perturbation theory, using the reactor period as the eigenvalue corresponding to a neutron lifetime of 1.5×10^{-7} sec. Calculations were also made using reactivity as the eigenvalue. Danger coefficients calculated by the two methods agree. (A.C.)

14223 AECU-3803

Detroit Edison Co., Detroit.

THERMAL CYCLE TEST IN SODIUM OF NINE APDA FUEL PINS. Engineering Laboratory and Research Department Report 57C05-2. J. J. Bodzin and T. A. Alessi. Feb 12, 1958 45p. Contract [AT(11-1)-476] \$7.80(ph OTS), \$3.30(mf OTS).

The investigation undertaken to determine the effect of thermal cycling in sodium of various fuel pins has been concluded. Nine fuel pins, with various heat treatments, different end caps, and made by different manufacturers, were submitted for this test. The test consisted of the following thermal cycle: heating to 1375 ± 25°F, holding; cooling to 250 ± 25°F. The temperature

of the test specimens exceeded 1200°F for at least 1 hour for each cycle. At the conclusion of the tests, the pins were examined visually to determine whether any physical damage was evident, and what dimensional and metallurgical changes had occurred. The nine fuel pins that were thermally cycled all reacted similarly. The three variables that were thought to influence pin failure, namely fuel pin fabricator, type of heat treatment, and surface preparation were found to be inconsequential in this test. (W.L.H.)

14247 KAPL-1757

Knolls Atomic Power Lab., Schenectady, N. Y.

THE FAST OXIDE BREEDER—THE FUEL CYCLE. J. K. Davidson, W. O. Haas, Jr., J. L. Mewherter, R. S. Miller, and D. J. Smith. July 1, 1957 98p. Contract W-31-109-Eng-52 \$2.25(OTS).

The development of a fast breeder fuel cycle based on a core fuel consisting of mixed dioxides of uranium and plutonium is described. Process flowsheets, material balances, operating schedules, cost schedules, and a plant layout are presented, and the laboratory- and plant-scale process and equipment demonstrations are described. A cost estimate based on a demonstration of the process including the unique plant equipment gives a total fuel cycle cost of 2.04 mils/kw-hr and a net fuel cycle cost of 1.53 mils/kw-hr (0.3 breeding gain assumed). (auth.)

14252 NMI-4402

Nuclear Metals, Inc., Cambridge, Mass.

PRDC MELT-DOWN TESTS. PROGRESS REPORT ON SINGLE-PIN MELT-DOWN TESTS FOR PRDC DURING THE PERIOD FEBRUARY 1957 THROUGH JUNE 1957. J. J. Pickett, A. R. Kaufmann, and J. E. Roman, Jr. Jan 16, 1958. 42p. Contract [AT(11-1)-476] \$6.30(ph OTS), \$3.00(mf OTS).

This program has indicated that the time that a thin Zr cladding can contain the molten U-Mo core alloys can vary over a considerably wide range and is a function of the original cladding condition. Small surface imperfections which now are either overlooked or disregarded, as well as internal defects which are not now found appear to be principal causes of premature failure. Also demonstrated in this program is the fact that serious defects can be produced in what is believed to be sound cladding by touching the cladding with stainless steel at high temperature (>1100°C). (W.L.H.)

14270

ANALYSIS OF TEMPERATURES AND EXPANSIONS RESULTING FROM EXPONENTIAL POWER CHANGES IN A REACTOR. Thomas R. Bump and Ralph W. Seidenstucker (Argonne National Lab., Lemont, Ill.) Nuclear Sci. and Eng. 4, 44-64(1958) July.

Simple, analytic expressions for temperatures and axial fuel expansions in an exponentially heated, heterogeneous reactor are derived. Constant thermal resistances, equal to the steady-state values, are assumed and the effects of using this assumption are discussed. Multiregion fuel elements are treated, both with and without coolant flow. The effect of mechanical inertia upon axial expansion of fuel rods with one fixed end is taken into account. Numerical examples which pertain to EBR-II are presented. (auth.)

14816 A/CONF.15/P/622

Atomic Power Development Associates, Inc., Detroit and Battelle Memorial Inst., Columbus, Ohio.

RADIATION STABILITY OF FUEL ELEMENTS FOR THE ENRICO FERMI POWER REACTOR. D. O. Leeser, F. A. Rough and A. A. Bauer. 12p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

The irradiation test program on fuel elements for the Enrico Fermi Reactor has covered binary alloys of U-Cr, U-Zr, and U-Mo. The U-5 wt. % Cr eutectic alloy investigation showed insufficient radiation stability for use in a fast reactor being developed as a possible commercially practicable means of producing electric power. Preliminary studies on U-Zr alloys containing 2, 2.2, 3, 5, 10, and 15% by weight of Zr and U-Mo alloys containing 3½, 5, 7, and 10% by weight of Mo have been performed. Various heat treatments were used in preparing the alloyed specimens to determine which treatment, if any, would result in improved radiation stability. A screening program on these binary alloys at low burn-ups (about 0.1 total at. %) indicated that the U-10 wt. % Mo alloy possessed the best stability of the alloys studied, and provided a good reduction of the anisotropic behavior of fabricated U. Therefore, this material was selected as the reference fuel alloy for the Enrico Fermi Reactor. Test data reported include that performed on specimens that were bare, Zr clad with open ends, and Zr clad with ends capped with Zr and with stainless steel. In general, it can be stated that (a) the radiation stability of the gamma treated 10 wt. % material below 1100°F is good to 2 at. % burn-up, (b) large changes that occurred, such as swelling or ruptures, have occurred at burn-up less than 1.5 at. % in material irradiated above 1300°F, and (c) ruptures occurred in material that is restrained by heavy cladding (8 mil or greater) and/or material that is fabricated improperly. These latter criteria are detailed as to the effects of each and their relative importance to radiation stability. (auth)

14987 A/CONF.15/P/419

Los Alamos Scientific Lab., N. Mex.

APPLICATIONS OF GODIVA II NEUTRON PULSES

T. F. Wimet and J. D. Orndoff. 24p. \$0.50(OTS)

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

Photographs of Godiva I and II are shown. Some of the observed properties of fission bursts are presented showing their relation to theory together with examples of a number of different experiments utilizing Godiva I and II bursts, such as: effects of dosage rates on mammals, irradiation studies, instrument response studies, and transient radiation effects in elementary materials. (M.H.R.)

15007 A/CONF.15/P/458

Argonne National Lab., Lemont, Ill.

PREDICTED DYNAMIC BEHAVIOR OF EBR-II. T. R. Bump and H. O. Monson. 31p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

The Experimental Breeder Reactor-II is designed to operate with very high core power density (1.37 Mw/liter, max.) and high fuel temperature (1200°F, max.). Accordingly, even small perturbations in reactor power or coolant flow rate become potential sources of damage to the reactor. Therefore, extensive calculations were made to determine quantitative effects of disturbances considered likely to occur during normal reactor operation and those which would derive from certain postulated accidents. Three general types of perturbations are considered: (1) sudden cessation of reactor coolant pumping power, both before and after reactor scram; (2) small changes in reactor coolant pumping power, both with and without automatic regulation of reactor power; and (3) step changes in reactivity. The conclusions drawn from these calculations are presented and their effects upon EBR-II design and permissible operating conditions are discussed. (auth)

15014 A/CONF 15/P/541

Argonne National Lab., Lemont, Ill.

THE PYROMETALLURGICAL PROCESS AND PLANT

FOR EBR-II M. Levenson, G. Bernstein, J. Graae, L. F. Coleman, D. C. Hampson, and J. H. Schraidt. 13p. \$0.50(OTS)

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958

The Experimental Breeder Reactor-II plant will consist of a complete fuel cycle as well as a complete power cycle. The fuel cycle facility contains equipment for the determination of the technical feasibility of pyrometallurgical processing and remote refabrication of spent reactor fuel elements. The specific pyrometallurgical process selected for inclusion in this plant and the technical features of plant design are described. The main unit of the fuel cycle facilities is a sextodecagon hot cell. This shielded shop contains the equipment for the decanning step, the melt refining of the spent reactor fuel, the fabrication of new fuel pins and the assembly of new fuel elements. An ancillary shielded cell is used for the remote fabrication of complete subassemblies and for servicing of main cell equipment. (auth)

15023 A/CONF.15/P/598

Argonne National Lab., Lemont, Ill.

FAST NEUTRON POWER REACTOR STUDIES WITH

ZPR-III. J. K. Long, W. B. Loewenstein, C. E.

Branyan, G. S. Brunson, F. S. Kirn, D. Okrent, R. E. Rice, and F. W. Thalgott. 44p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

Argonne's critical facility, ZPR-III, is briefly described. More than 30 different assemblies of widely varying geometries and compositions were studied in this facility, most of which were to aid in the development of fast power breeder reactors. A description of the ZPR-III assemblies, tables of central reactivity coefficients, prompt neutron data, fission distributions, multigroup constants, critical masses, and other data from these experiments are presented. (M.H.R.)

15024 A/CONF.15/P/599

Argonne National Lab., Lemont, Ill.

EXPERIMENTAL AND THEORETICAL STUDIES OF THE COUPLED FAST-THERMAL SYSTEM, ZPR-V.

H. H. Hummel, C. E. Cohn, G. J. Fischer, W. Y. Kato, F. H. Martens, D. Meneghetti, and B. J. Toppel. 33p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958

The zero-power reactor ZPR-V consists of a central fast core region separated from an outer annular water-moderated region by a blanket of natural uranium. The fast core composition is typical of a dilute power breeder and has a U²³⁸ to U²³⁵ ratio of 5. System characteristics such as flux distributions, neutron energy spectra, effect of change in fuel loading, and effect of change in blanket thickness, have been studied experimentally. The energy spectrum in the fast core is similar to that in the core of a dilute fast breeder, therefore a study of the spectrum, of reactivity effects, and of activation rates provides useful checks of fast neutron parameters. A careful review has been made of the fast neutron cross sections of U²³⁵ and U²³⁸ to provide improved multigroup parameters. Multigroup diffusion theory calculations using these new parameters are employed to compute the statics of the over-all system for comparison with experiment. Selengut-Goertzel slowing-down is used in the water region. (auth)

15037 A/CONF.15/P/637

Argonne National Lab., Lemont, Ill.

THE PHYSICS OF FAST POWER REACTORS A STATUS REPORT. W. B. Loewenstein, and David Okrent. 41p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

The nuclear performance of various conceptual fast power breeder reactor systems is reviewed in terms of existing experimental critical assembly data. A single set of nuclear constants serves to bridge the gap between existing experimental data and conceptual systems. Current knowledge of pertinent neutron cross-sections and fission parameters is reviewed. (M.H.R.)

15046 A/CONF.15/P/791

Argonne National Lab., Lemont, Ill.

THE MANUFACTURE OF EBR-I, MARK III FUEL AND BLANKET RODS. Robert A. Noland, N. J. Carson, C. C. Stone, F. D. McCusig, and R. E. Macherey. 15p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

The fuel and blanket elements for the EBR-I, Mark III core loading are round rods nominally $0.403" \pm 0.001"$ diameter by 21.750" long. Both types are identical in external appearance and size. The uranium-bearing section is 19.812" long. This zone is a single section of normal uranium in the blanket rods. Fuel rods contain an 8.500" central section of uranium enriched to 92.5% U^{235} flanked at each end by normal uranium sections. Solid Zircaloy 2 end caps are at the terminal ends of both types. The rods were made of lengths of 2 wt. % uranium-zirconium alloy clad with a 20 mil jacket of Zircaloy 2 by coextruding alloy billets sealed in Zircaloy 2 cans. To circumvent the end effects normally encountered in coextrusion and to minimize the number of billets required, billets of a volume sufficient to provide a number of fuel blanket elements per extruded length were used. The long rods were next cut into shorter lengths appropriate for assembly into either fuel or blanket rods. These were then assembled with the Zircaloy 2 end caps and/or fuel sections and the sections were joined at the jackets by tungsten inert gas welding. To provide mechanical strength to the welded joints, the sections were next bonded throughout the cross section at each joint interface by fusing the uranium alloy to the Zircaloy 2 end fittings and disc separators. Each completed rod is provided with three longitudinal spacing ribs, fastened at 120° intervals around the periphery of the jacket. The ribs, in the form of 0.056" diameter zirconium wire, were automatically spot welded in place, all three ribs being fastened in a single set-up. (auth)

15047 A/CONF.15/P/792

Baltimore Gas and Electric Co., Md. and Atomic Power Development Associates, Inc., Detroit.

DEVELOPMENT OF CORE ELEMENTS FOR THE ENRICO FERMI POWER REACTOR. W. N. McDaniel, O. E. Homelster, and D. O. Leeser. 16p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

Various core element binary alloys and configurations have been covered in the core element fabrication development program for the Enrico Fermi Reactor. Alloys of U-Cr, U-Zr and U-Mo were considered. Those with good casting qualities were cast into flat plates, corrugated plates, and even large castings with integral coolant channels. The U-3.5 wt. % Mo alloy was reduced to powder by hydriding and fabricated into flat plates by a powder metallurgy technique. Still others were extruded into small diameter cylindrical rods. In all cases

it was difficult to develop the technique to suitably clad the fuel alloy, a necessary requirement in order to reduce the amount of fission gas release to the core coolant and to eliminate mass transport of the uranium alloy. A coextrusion process for fabricating cylindrical rods with the cladding metallurgically bonded to the uranium appeared to be a satisfactory method of obtaining the type of cladding desired. Therefore, this fabrication process was chosen for the core elements. The alloy chosen was U-10 wt. % Mo because of its irradiation stability. (auth)

15061 A/CONF.15/P/1075

Division of Reactor Development, AEC and Argonne National Lab., Lemont, Ill.

EXPERIENCE WITH U. S. NUCLEAR POWER REACTORS. Clark Goodman, Louis H. Roddis, Jr., and W. H. Zinn. 63p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

The time table and operating histories of all of the existing nuclear power plants and experimental power reactors in the United States are presented in tabular form. The long operating experience of the U. S. in a variety of power producing reactors, with special emphasis on the water systems and on the use of enriched fuels is summarized. A total of more than twenty reactors is covered ranging in size from the small Los Alamos Fast Breeder Reactor with no electrical power output, which started operating at temperatures useful for possible power production in 1946, to the first full-scale nuclear power plant designed as a commercial installation, the Shippingport Pressurized Water Reactor, which operated at 60 Mwe output in December 1957. Where appropriate, economic considerations are included. (auth)

15068 A/CONF.15/P/1547

Argonne National Lab., Lemont, Ill.

ZPR-IV - A VERSATILE SOURCE OF NEUTRONS.

F. C. Beyer, R. C. Doerner, and F. H. Martens. 7p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

The ZPR-IV is a low power reactor consisting of an MTR core placed against an aluminum leakage face in one side of a water tank (8 x 6 x 5 ft). Neutrons leaking from this face are used for experimentation. This reactor, its physical plant, instrumentation, controls, and program are described. (M.H.R.)

15069 A/CONF.15/P/1549

Argonne National Lab., Lemont, Ill.

THE RADIOCHEMICAL ANALYSIS ON THE EBR-I FUEL ASSEMBLY MELTDOWN. R. F. Buchanan. 10p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

This analysis was undertaken to determine the mechanism of the meltdown in the fuel element core of the Experimental Breeder Reactor-I (EBR-I). The conclusions are limited to those based on the results of these analyses. Selected samples from the meltdown area were analyzed for uranium, iron, potassium, cerium-144, plutonium-238, and for plutonium and uranium mass. The chemistry involved in these determinations is reported and a diagram of the sampling in the meltdown area is shown. The chemical and mass spectrographic analyses indicate that there was very little mixing in the core as a result of the meltdown and that the fuel alloy which penetrated the blanket sections originated from the outer part of the molten zone. (auth)

15072 A/CONF 15/P/1777

Knolls Atomic Power Lab., Schenectady N Y and
Argonne National Lab., Lemont, Ill.

MEASUREMENT OF DOPPLER TEMPERATURE COEFFICIENT IN INTERMEDIATE AND FAST ASSEMBLIES R T Frost, W Y Kato, and D K Butler
12p \$0.50(OTS)

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958

Measurements made of the Doppler temperature effect for material samples in intermediate and fast spectrum reactors and the experimental technique applied are described. The materials studied in the Be-moderated intermediate reactors were highly enriched U^{235} , U^{238} , and Hf. In the fast reactors, constructed alternatively of highly enriched U and Al or Pu and Al, experiments were performed with samples of enriched and natural U and Pu. Because of the extremely small magnitudes of the Doppler coefficient in the fast spectrum, only upper bounds could be measured for the effect in the latter experiments. (M H R)

15073 A/CONF.15/P/1782

Argonne National Lab., Lemont, Ill.

CONSTRUCTION DESIGN OF EBR-II AN INTEGRATED UNMODERATED NUCLEAR POWER PLANT. L. J. Koch, H. O. Monson, W. R. Simmons, M. Levenson, F. Verber, E. Hutter, R. A. Jaross, T. R. Spalding, J. R. Simanton, and A. Lovoff. 43p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

The Argonne Experimental Breeder Reactor-II is a completely integrated nuclear power plant which employs an unmoderated reactor as a heat source, and generates 20 000 kw of electricity. The integrated nuclear plant includes the fuel re-processing and fabrication cycle and thereby makes a significant contribution to power reactor technology by operating on recycled fuel. The preliminary design of this plant was described at the First International Conference on the Peaceful Uses of Atomic Energy in 1955. Although the basic design concept and objectives have remained unchanged many modifications have been made in the detailed design, including separation of the facility into several plants and providing containment for the reactor system. The plant capacity and basic power cycle have remained essentially unchanged. 62.5 Mw thermal 900F reactor sodium outlet temperature and 1250-psig steam at 850F at the turbine throttle. The feed-water temperature, however, has been increased to 550F to reduce thermal stresses in the steam generator. (auth)

15079 A/CONF.15/P/1845

Argonne National Lab., Lemont, Ill.

STABILITY STUDIES ON EBR-I. F. W. Thalgott, J. F. Boland, R. O. Brittan, J. C. Carter, F. D. McGinnis, M. Novick, D. Okrent, H. A. Sandmeier, R. R. Smith, and R. E. Rice. 25p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

During this period of operation as a small power reactor and irradiation-test facility, two dynamic characteristics were observed which required further study. One of these, the power response to sudden changes in coolant flow, indicated the presence of a prompt positive power reactivity coefficient, this coefficient being overcome by a slower negative coefficient to produce an over-all negative effect. The second phenomena was observed at power-to-coolant flow ratios greater than the design ratio. Oscillations in power occurred which became more severe as power was increased or coolant flow decreased. A complete description of the kinetic behavior of the EBR-I is given. (auth)

15081 A/CONF.15/P/1850

Atomic Power Development Associates, Inc., Detroit;
Power Reactor Development Co., Detroit; and
Detroit Edison Co., Detroit.

ENRICO FERMI ATOMIC POWER PLANT. Alton P. Donnell, Robert W. Hartwell, and Harvey A. Wagner.
19p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

The fast breeder reactor designed and developed during the past six years is being installed in the Enrico Fermi Atomic Power Plant. This plant is being constructed by the Power Reactor Development Company and the Detroit Edison Company under the Atomic Energy Commission power demonstration reactor program. On the basis of work done to date, the breeder reactor continues to be a promising type for the eventual development of electric power from nuclear fission on a commercially feasible basis. Construction is in progress pursuant to a permit issued by the AEC. The reactor, a full-scale developmental fast breeder, with a design rating of 300 Mw and 100 Mw, is being built on land leased from the Detroit Edison Company. The latter will construct, own, and operate adjacent turbine generating facilities and will purchase the steam required for those facilities from Power Reactor Development Company. Design work on the nuclear and non-nuclear portions of the plant is well advanced. (A.C.)

15090 A/CONF.15/P/1892

Argonne National Lab., Lemont, Ill., and Illinois Inst. of
Tech., Chicago Armour Research Foundation.

CONTAINMENT OF EBR-II. H. O. Monson and M. M. Sluyter. 33p. \$0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958

The Experimental Breeder Reactor-II features two systems for containment of fission products and/or plutonium released in the unlikely event of a major nuclear accident. The primary containment system, which surrounds the reactor tank, is designed to contain the effects of a nuclear energy release without breaching. The building containment system, which surrounds both the primary containment and the balance of the reactor plant, serves to localize within the building the effects of a possible sodium-air reaction. The systems are described and evaluated. (auth)

15098 A/CONF 15/P/2156

Argonne National Lab. Lemont Ill.

ANALYSIS OF THE EBR-I CORE MELTDOWN. R. O. Brittan. 13p. \$0.50(OTS)

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958

An analysis of the nuclear thermodynamic and physical behavior of the core was made which yields information on temperature and material disposition which is consistent with observation and measurements on the final state of the core. In particular it is shown that shutdown of the nuclear reaction occurred as a result of scram action and not as a result of change in reactivity due to physical changes in the core material. The latter could have been a self-limiting action but only with a resulting energy yield inconsistent with the final condition and disposition of the core materials. The data obtained during the experiment makes it possible to determine the power coefficient of reactivity and to estimate temperature coefficients. Estimates made during the time the power level recorder was off-scale yield most probable values of 9000 kw for the peak power level, and 14,000 kw for the total energy release. The corresponding temperature estimates indicate that the peak of about 1100°C occurred after peak power. (auth)

15822 AECU-3816

Power Reactor Development Co., Detroit and Detroit Edison Co., Detroit.

AN INTEGRATED DESIGN OF REACTOR AND TURBINE GENERATOR SECTIONS OF THE ENRICO FERMI ATOMIC POWER PLANT. R. C. Kirkby and C. M. Heidel. 36p. \$6.30(ph OTS); \$3.00(mf OTS).

The design of the steam-generating system and the turbine generator system of the Enrico Fermi power plant is given. The terms of the agreement between The Power Reactor Development Company and The Detroit Edison Company are discussed. (W.D.M.)

15851 PRDC-TR-12

Power Reactor Development Co., Detroit.

MONTHLY TECHNICAL REPORT [FOR] JUNE 1958 11p. Contract AT(11-1)-476. \$3 30(ph OTS), \$2.40 (mf OTS).

The status of the design and fabrication of major components being procured for the Reactor Components Test is shown. Progress in nuclear engineering materials, fuel engineering, reactor vessel engineering, mechanical handling, liquid metal and steam system, and test operations is summarized. (For preceding period see PRDC-TR-11.) (W.L.H.)

15889

FABRICATION PROBLEMS OF THE DOUNREAY FAST REACTOR P. K. Richards (John Thompson Water Tube Boilers, Ltd.) Nuclear Power 3, 359-62(1958) Aug

Requirements for the fabrication of the reactor vessel and cooling system are as follows: all internals must be degreased, free from dirt, and thoroughly dry; heat exchangers must be designed so that in the event of a leak occurring there is no possibility of water-sodium contact; oxygen-containing compounds within the vicinity of the sodium circuits must be kept to a minimum. (W.D.M.)

15897

EVALUATION OF HEAT RELEASING ELEMENTS OF VARIOUS FORMS M. A. Zimin pp 149-163 in "Fizika i Teploekhnika Reaktorov." Moscow, Publishing House on Atomic Power, 1958 213p. (In Russian)

Fuel rods and plates, tubular elements with unilateral cooling, multilayer tubular elements with bilateral cooling, Field's type tubes with unilateral cooling, and rods in channels and in packages distributed in a triangle lattice were evaluated. The rod and plate shaped fuel elements distributed in packages which permit a mean specific power extraction with sodium cooling of 1000 to 1500 kw/l ($w = 9\text{m/sec}$) are the most effective forms for fast reactors. All examined forms can be used for thermal reactors in the following order of efficiency: rods and plates, tubular elements with unilateral cooling, and elements of Field's type tubes. (R.V.J.)

15904

NUCLEAR REACTOR PLANT DATA VOLUME ONE POWER REACTORS New York, The American Society of Mechanical Engineers, 1958 66p \$3.00

Information about the types of reactor plants and their experimental prototypes that are designed to produce electric power is presented. Both general information and specific technical data are presented for each plant on identical data sheets. (W.D.M.)

15905

REACTORS OF THE WORLD. FIRST SERIES. New

York, Simmons-Boardman Publishing Corporation, 1958 29p \$2.50

Cutaway drawings and specifications are presented of the NRX, Bradwell-on-Sea, Dounreay, DIDO, Calder Hall, LIDO, ZEUS, Dimple, BEPO, the Russian 5 Mw power, the EBWR, and the CP5 reactors. (J.S.R.)

15991 WASH-174(Del.)

Technical Information Service, AEC.

REACTOR SHIELDING INFORMATION MEETING SCHENECTADY, MAY 13 AND 14, 1954. 193p \$30 30 (ph OTS), \$8.70(mf OTS).

Thirty-one papers covering practically all phases of reactor shielding and related studies are included. A chronology of reactor shielding and the present status of shielding theory and microscopic nuclear data as presented in Papers 1 and 2, respectively, lay the foundation for the more specific papers which follow. Specific related theoretical computations and experimental measurements discussed include: neutron and γ attenuation in Hanford shielding materials; neutron and γ transmission through air slots; penetration of 6-Mev γ rays in H_2O , Fe, and Pb; moments method calculation of neutron penetration through light materials; stochastic calculations of γ diffusion; use of Boral in shielding; dose measurements in air and H_2O ; slowing down, attenuation, and slant penetration of neutrons in H_2O ; γ doses due to irregular geometry in slab shields; flux characteristics of extended sources; secondary γ production; broad-beam γ attenuations; thermal neutron flux calculation for EBR, survey of the BNL shielding program, and Monte Carlo calculations of neutron penetration in H_2O . A book-type subject index has been prepared to facilitate the locating of the information. (L.M.T.)

16999 BNL-511

Brookhaven National Lab., Upton, N. Y.

REPROCESSING OF POWER REACTOR FUELS. THE ENRICO FERMI FAST BREEDER REACTOR FUEL PROGRESS REPORT NO. 1. C. W. Pierce and B. Manowitz. June 1958. 9p. \$0.50(OTS).

Studies on the recovery of U and Pu from the Enrico Fermi Fast Breeder Reactor fuel are presented. (W.L.H.)

17056 WASH-170(Del.)(p.63-7)

Idaho Operations Office, AEC.

AIR CLEANING PROBLEMS AT NRTS. A. L. Biladeau. p.63-7 [of] THIRD ATOMIC ENERGY COMMISSION AIR CLEANING CONFERENCE HELD AT LOS ALAMOS SCIENTIFIC LABORATORY, SEPTEMBER 21, 22, AND 23, 1953 (deleted). 4p.

Prefiltering of all air for the plants is required because of dust. The air cleaning facilities for the Chemical Processing Plant are described. The air cleaning facilities for the Materials Testing Reactor and Experimental Breeder Reactor are briefly discussed. (W.L.H.)

17302 TID-2501(Del.)(p.425-40)

Knolls Atomic Power Lab., Schenectady, N. Y.

FEASIBILITY OF PLUTONIUM BREEDING H. Brooks p.425-40 [of] NUCLEAR SCIENCE AND TECHNOLOGY. (EXTRACTS FROM JOURNAL OF METALLURGY AND CERAMICS. ISSUE NOS 1 TO 6, JULY 1948-JANUARY 1951) 16p. (KAPL-M-HB-32)

Following a discussion of the need for breeder development and a summary of the attributes of a power-breeder system, the results of investigations into such factors of breeder design as the capture-to-fission ratio, choice of neutron spectrum, structural materials, poisoning by fission products and Be nuclear reactions,

and the enhanced neutron production arising from fast fissions in natural U are reported. Tentative breeding ratios are estimated. (W.L.H.)

17424 TID-2506(DeI.) (p 39-55)
Argonne National Lab, Lemont, Ill.
RECOVERY OF FISSIONABLE MATERIAL FROM ENRICHED-URANIUM FUELS. PART II. EXPERIMENTAL BREEDER REACTOR FUEL ELEMENTS. Leslie Burris, Jr., S. Lawroski, M. Levenson, R. C. Vogel, and S. Vogler. p.39-55 [of] NUCLEAR SCIENCE AND TECHNOLOGY (EXTRACTS FROM NUCLEAR SCIENCE AND TECHNOLOGY. VOL. 1, ISSUES 1 TO 3, FEBRUARY-JUNE 1955) 17p

The first core loading of the Experimental Breeder Reactor was of unalloyed enriched uranium. A solvent-extraction recovery scheme for this core material using tri-n-butyl phosphate is described. The process consists of a single extraction-scrub-strip cycle. The extraction of fission-product zirconium is minimized by complexing with oxalate ion, and the extraction of the small amounts of plutonium is minimized by reducing to the trivalent state. The laboratory data leading to the process flow sheet are presented. The results of the semi-works investigation of the process at the West Stands and the operation of a second unit at Lemont are described (auth)

17753 AERE-R/M-176
United Kingdom Atomic Energy Authority. Research Group Atomic Energy Research Establishment, Harwell, Berks, England

AN ABSOLUTE MEASUREMENT OF THE EFFECTIVE DELAYED NEUTRON FRACTION IN THE FAST REACTOR ZEPHYR. D. B. McCulloch. July 1958. 10p.

The feasibility and technique of a method which may be used to measure the delayed neutron fraction in a reactor are investigated. The two experimental methods tried were satisfactory. The film method is economical in reactor time and less susceptible to changes in reactor power than is the scaler method; however, this method requires only standard equipment, and the counts are in a form ready for immediate calculation of results. (J R D)

17756 ANL-5809
Argonne National Lab, Lemont, Ill.
MEASUREMENT OF THE DOPPLER TEMPERATURE EFFECT IN AN EBR-I TYPE ASSEMBLY. W. Y. Kato and D. K. Butler July 1958. 32p. Contract W-31-109-eng-38 \$1.00(OTS).

A measurement of the Doppler temperature effect has been made in the fast spectrum of a mock-up of the EBR-I reactor in ZPR-III. The effect was measured by thermally cycling samples of enriched uranium, natural uranium, and plutonium, and detecting the small changes in reactivity. The pile oscillation technique using a resonant detector was employed to measure the small oscillating component of the neutron flux. An upper limit $0.5 \times 10^{-6} \Delta k / \Delta T$, was obtained for a 506-gram sample of U^{235} in the spectrum of an EBR-I mock-up, and for a 235-gram sample of Pu^{239} in a plutonium-fueled assembly of the same configuration. The results for natural uranium were inconclusive. (auth)

Volume 13, 1959

130

EFFECT OF ALLOYING ON THE CRITICAL MASS OF A PLUTONIUM SPHERICAL FAST REACTOR. James T. Waber, Mary Repar Kline, and Leah K. Johnson (Los Alamos Scientific Lab., N Mex) Nuclear Sci. and Eng. 4, 341-53(1958) Sept.

The effect of alloying on the amount of plutonium required in forming a critical mass of each alloy is expressed in terms of an inventory requirement ratio, R_I . This quantity was obtained for twenty-seven potential alloying elements at three compositional levels. The effectiveness of using Vegard's law to estimate the density of the alloys was appraised by comparing the estimated densities and R_I values of nine intermetallic compounds with their x-ray densities and the R_I values computed from them. The parametric variation of R_I with f_T , the number of excess neutrons per collision was also studied. (auth)

416

ANL-5855
Argonne National Lab., Lemont, Ill.
IRRADIATION OF PROTOTYPE CP-4 (EBR-I) REACTOR FUEL SLUGS IN THE CP-3' GOAT HOLE. Final Report of Metallurgy Division Program 6 1.1. S. H. Paine, P. J. Peisani, and W. F. Murphy. Aug. 1958. 24p. Contract W-31-109-eng-38. \$0 75(OTS).

Experiments are described which studied the effect of irradiation upon small U^{235} cylinders simulating in size those designed for the initial CP-4 (EBR-I) fuel loading. Qualitative similarities between thermal cycling and irradiation effects were found, but none of the three types of material tested were entirely dimensionally stable. (auth)

477

PRDC-TR-13
Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT FOR JULY 1958 16p. Contract AT(11-1)-476. \$3.30(ph OTS); \$2.40(mf OTS).
The design and fabrication status of major reactor components is shown in chart form. Short status statements or completion summaries are given on a number of developmental studies of fuel and reactor materials, nuclear characteristics, and reactor cooling and instrumentation. (For preceding period see PRDC-TR-12.) (D.E.B.)

481

SOLID FUEL REACTORS Joseph R. Dietrich and Walter H. Zinn, eds. Reading, Mass., Addison-Wesley Publishing Company, Inc., 1958. 856p. \$10.75

An attempt is made to consolidate the information that has been developed for several solid-fuel reactor types which have received serious consideration in the United States but which are not yet represented by operating prototype plants. Information presented is of three main types: general research and development results, design data for specific reactors, and analytical

studies. Basic types of reactors discussed are the fast, heavy-water, gas-cooled, organic moderated, and plutonium recycle. Specific reactors discussed are the Experimental Breeder Reactor-II and the Fermi Fast Breeder Reactor. (W D M)

983

POWER REACTOR. W. H. Zinn (to U. S. Atomic Energy Commission). U. S. Patent 2,841,545. July 1, 1958.

A fast nuclear reactor system is described for producing power and radioactive isotopes. The reactor core is of the heterogeneous, fluid sealed type comprised of vertically arranged elongated tubular fuel elements having vertical coolant passages. The active portion is surrounded by a neutron reflector and a shield. The system includes pumps and heat exchangers for the primary and secondary coolant circuits. The core, primary coolant pump and primary heat exchanger are disposed within an impermeate tank which is filled with the primary coolant, in this case a liquid metal such as Na or NaK, to completely submerge these elements. The tank is completely surrounded by a thick walled concrete shield. This reactor system utilizes enriched uranium or plutonium as the fissionable material, uranium or thorium as a diluent and thorium or uranium containing less than 0.7% of the U^{235} isotope as a fertile material.

1349

ANL-5790
Argonne National Lab., Lemont, Ill.
METALLURGY DIVISION QUARTERLY REPORT [FOR] APRIL, MAY, AND JUNE 1957. July 1958. 60p. Contract W-31-109-eng-38. \$1 75(OTS)

Advanced Water Reactor Program. Seven reaction couples were studied for 60 days at 600 and 800°C. ThO_2-UO_2 with Zircaloy-2, stainless steel, lead, and graphite; lead with Zircaloy-2 and stainless steel; and graphite with Zircaloy-2. No reactions were observed at 600°C and only lead-Zircaloy-2 appeared to react at 800°C. **Fast Power Breeder Reactor Program.** EBR-I Mark III core is being fabricated. Tests in Borax-IV on ThO_2-UO_2 pellets lead-bonded to Al-1 wt % Ni tube sheets showed fracturing dependent on UO_2 content. Non-destructive testing development continued. Alloying with Ru showed no improvement in corrosion properties of U. Corrosion of M388 Al alloy as affected by phosphate, pH, and flow rate were studied. **Production, Treatment, and Properties of Materials.** Plastic deformation and thermal conductivity of U single crystals was studied. Investigation of recrystallization of alpha-U without a texture change continued. Tests of French uranium are reported. Barrier layers of UO_2 were found in sputtering of Zr and U onto U, probably resulting from impurities in the argon used. Phase transformations in the U-C system were studied, and gamma U

lattice parameters for 900°C annealed and water quenched U-Mo-Ru are given. Studies are also reported on the Zr corrosion mechanism and sintering of ceramic materials. (For preceding period see ANL-5717.) (T.R.H.)

1451 BNL-506

Brookhaven National Lab., Upton, N. Y.

NUCLEAR ENGINEERING DEPARTMENT PROGRESS REPORT [FOR] JANUARY 1-MARCH 31, 1958.

F. Maslan, ed. 66p. \$2.00(OTS).

An improved method of drying MgCl before making up NaCl-KCl-MgCl₂ eutectic results in a more reproducible U partition coefficient in salt-Bi equilibration. The activity coefficient of Zr in very dilute Bi solutions was found to be 7×10^{-4} at 700°C. Annealing of irradiated quartz optical cells at 500°C is reported. The behavior of fission products I and Xe in LMFR fuel systems was studied. Corrosion tests of several materials in fused salt mixtures are reported. The operation of five in-pile loops is reviewed. Bench contactor experiments are described. Rates of transfer of Sm from Bi-Mg-Sm alloy to the ternary eutectic salt mixture were measured for the case of small drops falling through a column of salt. The pumping and corrosive effects of thorium bismuthide slurries were studied. Corrosion of steel immersed in Bi-Mg-U-Zr alloy was evaluated. A number of measurements of neutron slowing-down and diffusion parameters were made with graphite, bismuth, and graphite-bismuth mixtures. Satisfactory dissolution media were found for the Enrico Fermi Fast Breeder core. A 15-ft-long clay column experiment with a simulated waste solution spiked with ^{90}Sr gave a decontamination factor of 1.6×10^6 . The mechanism of ^{90}Y bonding to an anion exchange resin was determined. Reactor irradiation of LiOH produces no detectable O^{20} at currently available neutron fluxes. A satisfactory system was developed for milking As^{72} from its Se^{72} parent. A process for producing pure carrier-free Mo^{99} was developed. (For preceding period see BNL-491.) (W.D.M.)

1825

CAPTURE AND FISSION PATTERNS OF U^{238} IN EBR-I (MARK II). P. Kafalas and R. R. Heinrich (Argonne National Lab., Lemont, Ill.). Nuclear Sci. and Eng. 4, 698-702(1958) Nov.

The capture and fission probabilities of U^{238} were determined radiochemically as functions of horizontal position in EBR-I (Mark II). The ratios $\sigma_c(\text{U}^{238})/\sigma_f(\text{U}^{235})$ and $\sigma_c(\text{U}^{238})/\sigma_f(\text{Pu}^{239})$ are presented as functions of horizontal position in the reactor. (auth)

2575

FABRICATING THE TUBE NEST FOR THE DOUNREAY CORE. TOOLROOM TECHNIQUES REQUIRED TO MEET SPECIFICATIONS. Nuclear Eng. 3, 494-5 (1958) Nov.

The machining techniques used to meet the very close tolerances on the core plates, tubes, and side plates for the Dounreay Fast Reactor core are discussed. All components are of stainless steel which made the machining problems difficult. (W.D.M.)

2582 CF-58-10-37

Oak Ridge National Lab., Tenn.

U^{233} BREEDERS—COMMENTARY AND GENERAL REMARKS. W. K. Ergen. Oct. 8, 1958. 11p. Contract [W-7405-eng-26]. \$3.30(ph OTS); \$2.40(mf OTS).

Breeding is expected to become a necessity in about 40 years and in view of the 20 years development time and 30 years service life of breeder reactors, develop-

ment of such reactors at present is timely. In plutonium breeders, the specific power is inherently low and the doubling time long. This seems to prevent such breeders from furnishing a large fraction of the energy demands of the expanding economy from uranium recoverable at or about present cost. U^{233} breeders can be designed to the requirements of low inventory and short doubling time, but the aqueous homogeneous reactor seems to be the only type which can adequately meet these requirements. (auth)

2967 ANL-5797

Argonne National Lab., Lemont, Ill.

METALLURGY DIVISION QUARTERLY REPORT [FOR] JULY, AUGUST, AND SEPTEMBER 1957. Oct. 1958. 58p. Contract W-31-109-eng-38. \$1.75(OTS).

Advanced Water Reactor Program. Three firings were made of initial closed-porosity fuel pellet bodies. Each firing contained pellets of the composition 90 wt. % ThO_2 -10 wt. % UO_2 with various additives and firing variables. Fast Power Breeder Reactor Program. To determine the potential usefulness of a Zr-5 wt. % Pu alloy, the fabricability of the alloy was tested. The manufacture of rod stock from which fuel and blanket elements for the Mark III loading of the EBR-1 were produced has been completed. The effect of irradiation on extruded and heat-treated U-2 wt. % Zr alloy for the EBR-1 is reported. Fabrication procedures for making graphite- U_3O_8 test specimens for the TREAT Reactor were investigated. Advanced Engineering and Development. Ultrasonic bond tests were conducted on 590 EBR-1 Mark III blanket fuel elements. The blanket rods and part of the fuel rods for the EBR-1 Mark III loading are being checked for cladding thickness by an eddy current system. Investigations of corrosion-resistant Zr-Nb alloy were continued. Corrosion of Mg alloys is being studied in support of the Mighty Mouse reactor program. Dynamic corrosion tests were performed on aluminum alloys, and results are included. Production, Treatment, and Properties of Materials. The progress of the program of preparing high-purity Pu by fused salt electrolysis is summarized. Velocities of ultrasonic waves propagated in directions suitable for determining the room-temperature elastic moduli C_{12} , C_{13} , and C_{23} of alpha U were determined. The investigation of recrystallization in heavily cold-rolled alpha-uranium sheet without a texture change was essentially concluded during this quarter. Self-diffusion runs in polycrystalline uranium in the gamma phase, using the sputtering technique, have yielded a tentative value for the diffusion coefficient between 10^{-8} and 10^{-7} cm²/second. The preparation of high-purity U-Pu alloys is reported. The data for the alpha-to-beta transformation temperatures in high-purity U and U-C alloys were confirmed by experiments on new specimens. Microstructure, density, and thermal arrest data were obtained for several injection cast, nominal U-5 wt. % fissium and U-8 wt. % fissium alloys. Phase diagrams are presented for U-Mo and U-Ru alloys. Alloy Theory and The Nature of Solids. Four new isomorphs of Ti_2Ni have been discovered. Effects of Irradiation on Materials. The experimental and analytical work on the radial distribution of thermal neutrons within cylindrically shaped fuel specimens during irradiation was completed. (For preceding period see ANL-5790.) (W.L.H.)

3395

STUDIES IN THE PHYSICS OF FAST NEUTRON REACTORS. A. I. Leipunskii, A. I. Abramov, et al. Atomnaya Energ. 5, 277-93(1958). (In Russian)

A short review is given on the results of research

with three experimental fast reactors, BR-1 (built in 1955), BR-2 (built at the beginning of 1956), and BR-3 (built in the middle of 1957), for experimentally determining the possible magnitudes of breeding coefficient and for checking the methods of reactor efficiency calculations (R V J)

3403

SHIELDING THE ENRICO FERMI FAST BREEDER REACTOR H. E. Hungerford and R. F. Mantey (Atomic Power Development Associates, Inc., Detroit). *Nucleonics* 16, No. 11, 120-5 (1958) Nov.

The solution of the shielding problems encountered in the Enrico Fermi Reactor are given. The shield has been designed from common materials whose durability over the expected life of the plant is known and whose cost is low. These are graphite, steel, and concrete, all used in the primary, secondary, and biological shields. Special shields are necessary around the 30-in. primary-coolant outlet lines because of the large neutron leakage. Two materials were considered for this shield (a) borated, diatomaceous-earth-aggregate luminate concrete and (b) pressed calcium borate material. The latter was chosen as the shield. Temperatures to be expected in the primary-coolant pump and heat-exchanger plugs preclude the use of ordinary concrete as plug shields. The present design calls for a concrete made from serpentine rock aggregate in luminate cement within the plug containers. Costs of the shielding materials are given. Elevation and plane views of the reactor give the locations of the shields. The shielding properties of the shields are presented graphically. (T.B.A.)

3432

PLUTONIUM FAST POWER BREEDER WITH OXIDE FUEL AND BLANKET ELEMENTS John B. Sampson and E. A. Luebke (Knolls Atomic Power Lab., Schenectady, N. Y.). *Nuclear Sci. and Eng.* 4, 745-61 (1958) Dec.

A fuel element consisting of plutonium and uranium oxide in steel tubing and capable of a large fraction of fuel burnup is described. As this fuel element makes possible recovery and refabrication with fewer steps than are required for a metal fuel element, lower re-cycle costs result. Breeders with fuel and fertile material in both oxide and metallic form were analyzed by the multigroup method on the UNIVAC for the purpose of comparing characteristics. A summary of the calculations is presented. The decrease in the breeding ratio resulting from the replacement of the metal core by oxide is only 0.2, a small effect in a future nuclear power economy where plutonium will have a low value as fuel rather than a high value as weapon material. Use of an oxide blanket may further reduce the breeding ratio by 0.05. An illustrative design is presented which has five atoms of uranium per atom of plutonium in the core and 45% sodium, a breeding ratio of 1.4 and a critical mass of 400 kg. Incremental refueling is assumed to reduce the control range required for 50% burnup of the original fuel loaded. (auth)

4237

APFX-442
General Electric Co. Aircraft Nuclear Propulsion Dept., Cincinnati
LITERATURE SEARCH ON FAST REACTORS Task 7271. D. E. Wetzel. Apr. 3, 1958. Decl. Oct. 7, 1958. 16p. (XDC-58-6-83) \$3.30(ph OTS); \$2.40(mf OTS)

Descriptions are given of geometries and compositions of fast reactor designs and experiments reported in the literature. Each description is divided into these categories; location, rating, fuel description, reflector, control, coolant, and remarks. Some experimental data are included. The reactor descriptions presented are

those of the Clementine, FBR I, Topsy, Lady Godiva, Zephyr, ZPR III, Zeus, Dounreay, FBR II, and Enrico Fermi fast reactors. The data were collected as reference material for possible fast reactor neutron physics studies in support of advanced design work. (auth)

4274

PRDC-TR-14
Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] AUGUST 1958. 13p. Contract AT(11-1)-476. \$3.30(ph OTS), \$2.40(mf OTS).

Results from fuel pin irradiations and nondestructive testing are given. Fuel cycle costs were prepared for various burnups and interest rates on fuel. Conclusions are drawn from fuel pin corrosion tests. Work on computer programs and pressure vessel engineering is reported. A sodium-water reaction test is described. Preliminary tests were run on the sodium vapor trap. Ultrasonic cleaning of dummy nozzles is described. (For preceding period see PRDC-TR-13.) (W.D.M.)

4275

PRDC-TR-15
Power Reactor Development Co., Detroit
MONTHLY TECHNICAL REPORT [FOR] SEPTEMBER 1958 24p. Contract AT(11-1)-476. \$4.80(ph OTS), \$2.70(mf OTS).

Sample collection and analysis were started for the preoperational environmental survey. Reports on environmental radioactivity at various locations are attached. Samples of a fissium-type alloy were heated to 2000°F to determine the melting range. A chart showing the status of the design and fabrication of the major components being procured by APDA for the reactor components test is included. Radiation and temperature effects on fuel pins were investigated. Dry lubricants for bearing surfaces were tested. Test operations on reactor components are described. (For preceding period see PRDC-TR-14.) (W.D.M.)

4353

FUEL-BREEDER FUEL ELEMENT FOR NUCLEAR REACTOR. W. E. Abbott and R. Balent (to U. S. Atomic Energy Commission). U. S. Patent 2,852,460. Sept. 16, 1958.

A fuel element design to facilitate breeding reactor fuel is described. The fuel element is comprised of a container, a central core of fertile material in the container, a first bonding material surrounding the core, a sheet of fissionable material immediately surrounding the first bonding material, and a second bonding material surrounding the fissionable material and being in contact with said container.

5113

ANL-5940
Argonne National Lab., Lemont, Ill.
EBR-II FUEL IRRADIATION FACILITIES. R. J. Schiltz, J. H. Monaweck, and E. S. Sowa. Nov. 1958. 18p. Contract W-31-109-eng-38. \$0.75(OTS).

Two irradiation facilities, designed for installation within tubular fuel elements, and peripheral experimental thimbles in the CP-5 reactor core are described. Both units are cooled internally by natural convection of D₂O and feature an inherent variable temperature-control system which permits ± 75 -degree central metal temperature variation for a given heat-transfer rate. Heat-transfer rates ranging from 10 tkw (fuel) to 15 tkw (thimble) are possible. External cooling is provided by the reactor D₂O moderator. Although designed primarily to simulate an operational environment for the evaluation of EBR-II prototype fuel elements, both facilities can be utilized for the irradiation of other fuel

materials of interest, consistent with the internal geometry of each unit. Complementary information includes thermal analyses and mock-up tests pertinent to the ultimate design. (auth)

5183

FAST OXIDE BREEDER. FUEL CYCLE AND COST. J. K. Davidson, R. S. Miller, and D. J. Smith (Knolls Atomic Power Lab., Schenectady, N. Y.). *Nucleonics* 17, No. 1, 88-93(1959) Jan.

The fast breeder is of long-range interest because it is based on one of the two fuel systems (a thermal reactor with Th-U²³³ cycle is the other) that can produce more fissionable material than it uses. The oxide-fuel cycle (UO₂-PuO₂) in a fast breeder is of particular interest because it appears possible to arrive at a fuel-cycle cost low enough to make this system of short-range interest as well. The reactor, economic analysis, and fuel-cycle development are considered. (L.T.W.)

5945

BAC/RL-1052

Bendix Aviation Corp. Research Labs., Detroit. ANALYSIS OF THE POWER LIMITING SYSTEM FOR THE ENRICO FERMI ATOMIC POWER PLANT. Nov. 1, 1958. 199p. Project 2528. For Atomic Power Development Associates, Inc.

A study was made of the Enrico Fermi Atomic Power Plant to determine the salient features of its power-limiting system. The transients induced by reactivity insertions and coolant flow changes were studied with an analog computer. Using these data, the power-limiting system was designed. Besides the scram action, two more responses of the system were proposed. These are "setback" and "sequential scram." The setback acts to reduce the power when a preset limit is exceeded but does not shut the reactor down. The sequential scram is the response that minimizes thermal shock when the reactor must be shut down because of power transients or the multiple failure of coolant pumps. With the data obtained from the analog computer study, trip levels at which the various responses must be initiated were set. In addition, a recommendation of the flow and temperature interlock functions needed to protect the plant was made. An analysis of the statistical fluctuations of growth factors and power derivative signals is presented. A set of general specifications for the power-limiting equipment, including a block diagram of the logic circuits, was drawn up. (auth)

5953

NDA-14-179

Nuclear Development Corp. of America, White Plains, N. Y.

TRANSIENT FUEL AND COOLANT TEMPERATURES IN THE ENRICO FERMI REACTOR CORE DURING A PERIOD OF FLOW REVERSAL. Harold G. Elrod, Jr. July 30, 1957. Includes Appendixes A, B, and C. 35p. \$6.30(ph OTS); \$3.00(mf OTS).

An accident to be investigated as a part of the APDA safety studies is that which would involve meltdown at full power of the APDA reactor with gravity loss of sodium from both the top tank and core. An analysis is presented, permitting a logical prediction of the transient temperatures occurring prior to the time when boiling begins, due to flow reversal and other factors. The calculation method is given as well as the results of application of the method to a numerical example. (J.R.D.)

5958

PRDC-TR-16

Power Reactor Development Co., Detroit. MONTHLY TECHNICAL REPORT FOR OCTOBER 1958

52p. Contract AT(11-1)-476. \$9:30(ph OTS); \$3.60(mf OTS).

Meltdown studies were initiated on specimens of interest as reactor fuel alloys. Reports are included from an environmental radioactivity survey. The design and fabrication status of major reactor components is shown in chart form. Nuclear engineering studies and test procedures connected with installation of the various components are outlined. (For preceding period see PRDC-TR-15.) (C.H.)

6929

A/CONF.15/P/2038

STUDIES IN THE PHYSICS OF FAST NEUTRON REACTORS. A. I. Leipunskii (Leipunsky), A. I. Abramov, V. N. Andreev, et al. (Atomic Energy Utilization Central Board, U.S.S.R.). 46p.

Results of research carried out on fast reactor physics at the Institute of Physics of the Atomic Energy Utilization Central Board are presented. Description is given of studies on fission neutron propagation through thick spheres of U²³⁸. The experiments helped to determine a number of U²³⁸ properties defining the role of U²³⁸ fission in fast neutron reactors. Experimental results achieved on a series of critical assemblies are given for reactor power not in excess of 30 w. The reactor core consisted of plutonium rods. The critical facility design made it possible to study the reactor cores with relatively high plutonium content. The reactor control was accomplished by moving the inner parts of screens. Three reactor design versions are described in detail. In one of them the screen is made of solid uranium slugs. In the second reactor design version it consists of a close uranium water lattice separated from the core by a layer of solid uranium. In the third design version a nonfissionable screen made of copper is used. Neutron detectors of different sensitivity to neutron energy allow determination of the neutron energy spectrum in different parts of the reactor. Direct measurements of the neutron spectrum were made. The reactivity perturbations caused by foreign particles in the reactor vessel were measured, and measurements of space dependence of the neutron value for different neutron energies were made. Experimental results are compared with the data calculated by the multigroup kinetic theory. Basic results of nuclear research of the 100-kw experimental fast neutron reactor with mercury cooling are described. Space distribution of different processes, neutron energy spectrum, efficiency of regulating instruments, temperature effects, and other parameters of the reactor have been measured. Comparison is made of the experimental results with the results of theoretical calculations. (auth)

7082

A/CONF.15/P/39

EXPERIMENTAL WORK WITH ZERO ENERGY FAST REACTORS. R. D. Smith and J. E. Sanders (Atomic Energy Research Establishment, Harwell, Berks, Eng.). 83p.

In the small plutonium fuelled reactor, Zephyr, the original natural uranium reflector was replaced by graphite as a first stage in the study of mixed thermal-fast systems. Zeus uses enriched uranium fuel with a natural uranium reflector, in order to simulate as closely as possible the nuclear characteristics of the Dounreay fast breeder reactor. Fission and activation rate distributions, kinetic studies, and perturbation measurements were made for both reactors and the results for several different core configurations in Zeus are compared. Measurements of the mean neutron life-time were obtained by both static and kinetic techniques. The temperature coefficient due to the

Doppler effect in Zeus was estimated by measuring the perturbation effect caused by heating a U^{235} sample at the center of the core. The results of some of these experiments are compared with theoretical predictions using one-group and multi-group theories. (auth)

7151 A/CONF.15/P/1462

STABILITY CONSIDERATIONS IN THE DESIGN OF FAST REACTORS. R. T. Ackroyd, J. E. Mann, J. D. McCullen (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.), and G. H. Kinchin (Atomic Energy Research Establishment, Harwell, Berks, Eng.). 36p.

A theoretical study of stability considerations in the design of fast reactors is given and some experiments suggested. Reactor response to impressed sinusoidal, step and ramp additions of reactivity, and to pump failures is discussed. The influence of the magnitude and sign of individual reactivity coefficients on response is considered. The importance of the delay times of these coefficients is stressed and the possibility of self-excited oscillations and resonance instability is examined. Suggestions are made as to the possible sources of unacceptable types of coefficients and hence an indication is given of the design factors which should be kept in mind to ensure that such coefficients are eliminated or reduced to acceptable limits. It is shown that a negative power coefficient of reactivity does not, in itself, ensure stability and that oscillatory stability does not guarantee satisfactory over-all behavior. (W.D.M.)

7175 A/CONF.15/P/2129

EXPERIMENTAL FAST REACTORS IN THE SOVIET UNION. A. I. Lel'pinski (Lel'pinsky), V. G. Grabin, N. N. Aristarkhov, et al (U.S.S.R.) 32p.

In accordance with the general program aimed at fast reactor development, experimental reactors BR-1, BR-2, and BR-5 of 30 w, 150 kw, and 5 Mw, respectively, have been constructed and put into operation. The reactor and experimental programs are described. Preliminary design parameters are given for the proposed sodium-cooled 200 and 250 Mw power reactors BN-50 and BN-250. (W.D.M.)

7189 A/CONF.15/P/2411

THERMAL FLUX IN THE CENTRAL MODERATOR-ZONE OF THERMAL AND FAST REACTORS. M. Osredkar (J. Stefan Inst., Ljubljana, Yugoslavia) and R. Stephenson (New York Univ.). 26p.

To see the feasibility of the proposed assembly, the behavior of the thermal flux in the central moderator-zone of thermal reactors was investigated. The system having a pure-fuel fast core with the absorber at the interface was compared to systems without the absorber. A blocking layer (light water and the absorbing shell) was studied separately. The system with diluted fuel was investigated and the influence of the central moderator-zone on kinetics of the proposed system estimated. The two group diffusion theory in spherical geometry was used in calculations. (W.D.M.)

7200 A/CONF.15/P/44

THE MANUFACTURE OF FUEL ELEMENTS FOR THE DOUNREAY FAST REACTOR. C. J. Turner and L. R. Williams (United Kingdom Atomic Energy Authority, Springfields, Lancs, Eng.). 32p.

The initial fuel element design was based on the use of an outer niobium tube and a vanadium inner tube, the highly enriched uranium being contained between these two in the form of annular sections. This design has remained substantially unaltered throughout the devel-

opment period. An assessment of the complex problems associated with the production of such a fuel element is given. These included full metallurgical development of the two metals niobium and vanadium. A description of the development programs associated with the production of the fuel, canning materials, and assembly techniques is given, together with an account of the development of inspection and testing techniques for both the individual components and the completed fuel element. (auth)

7225 A/CONF.15/P/274

THE DOUNREAY FAST REACTOR—BASIC PROBLEMS IN DESIGN. H. Cartwright, J. Tatlock, and R. R. Matthews (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.). 21p.

The engineering design of the Dounreay Fast Reactor is reviewed. The basic problems inherent in a small highly-rated system are considered in the light of fundamental principles adopted at the outset. The reactor features are discussed in relation to the core, control system, primary coolant system, liquid metal to water heat exchangers, and multi-circuits for heat rejection. (auth)

7235 A/CONF.15/P/961

BREEDING AND CONVERSION PROCESSES IN HIGH-TEMPERATURE REACTORS. H. Wundt (Brown, Boveri & Cie. AG., Mannheim, Ger. and Fried. Krupp, Mannheim, Ger.). 39p.

The reactor design considered is a one-region reactor containing only fissile and fertile materials besides the graphite moderator and reflector. Since the maximum enrichment available to most countries is 20%, the conversion process involves both U^{233} and Pu^{239} . The two initial moderating ratios of the fertile materials are taken as independent parameters, and for 60 combinations of these the time behavior of k_{∞} is worked out which shows the most desirable charging schedule for new fuel and possible burnup times. (W.D.M.)

7270 A/CONF.15/P/2028

THE ECONOMICS OF FAST POWER REACTOR NUCLEAR FUEL. O. D. Kazachkovskii (Kazachkovsky), (U.S.S.R.) 20p.

The methods for calculating the fuel component in the cost structure of power produced by fast power reactors are defined. Account is taken of the many factors which determine the peculiarities of fast reactor operation. All considerations are made in relation to $Pu-U^{238}$ closed cycles. (W.D.M.)

8227 AECU-4007

[Argonne National Lab., Idaho Falls, Idaho].
RESULTS OF PARALLEL FLOW OSCILLATOR STUDIES. Progress Report No. 2. R. R. Smith, J. F. Boland, and F. W. Thalgott. Apr. 17, 1958. 41p.
\$7.80(ph), \$3.30(mf) OTS.

Parallel coolant flow studies on EBR-I Mark III are reported, and data from previous series flow studies are refined. It is indicated that changing from series to parallel causes little or no change in the reactivity feedback. The characteristics of the negative power coefficient of reactivity are apparently identical in both cases, and the conclusions reached as a result of the series flow studies are still valid. (T.R.H.)

8228 AECU-4008

[Argonne National Lab., Idaho Falls, Idaho]
OSCILLATOR RESULTS IN EBR-I, MARK III WITH

FUEL RODS LOOSE STEADY STATE TEMPERATURES FOR LOW POWER PARALLEL FLOW RUNS Progress Report No. 3 R. R. Smith, J. F. Boland, and F. W. Thalgott May 6 1958 13p \$3.30(ph) \$2.40 (mf) OTS

The results are summarized of oscillator studies on EBR-I Mark III at 515 kw under series flow with fuel elements in two of the seven core assemblies loosened. Steady-state temperatures for the low-power tests under parallel flow conditions and for 700 kw under series flow are also summarized. A significant positive temperature coefficient of reactivity was not found. (For preceding period see AECU-4007) (T. R. H.)

8285 NDA-14-178

Nuclear Development Corp. of America, White Plains, N. Y.
FLAT-SLAB FREEZING PROCESSES WITH RADIOACTIVE MATERIALS HAVING INTERNAL HEAT GENERATION H. G. Elrod, Jr. July 1, 1957 32p \$6.30(ph), \$3.00(mf) OTS

An analysis is presented for the case of a flat slab of material suddenly exposed on both sides to a pool of liquid metal. Results were obtained when radioactivity was present and when it was absent from the liquid metal. The analysis is applied to the axial blanket design of APDA reactor, and the times required after first solid-liquid contact for bridging between adjacent pins and for complete freezing are shown. (J. R. D.)

8286 NDA-14-181

Nuclear Development Corp. of America, White Plains, N. Y.
FREEZING AND REMELTING OF RADIOACTIVE MATERIAL ON COLD, VERTICAL PINS. H. G. Elrod, Jr. July 1, 1957. 24p. \$4.80(ph), \$2.70(mf) OTS.

If, in some unforeseeable manner, the APDA reactor should instantaneously lose all its coolant, a slow melt-down would occur. Molten fuel from the core would freeze upon encountering the colder pins of the lower axial blanket. A description of the kinetics of such a freezing process is presented. Generalized curves are presented, permitting rapid estimates of the time and extent of freezing for various pin configurations and rates of decay heat generation. For the blanket design described in APDA-115, it is estimated that bridging between adjacent pins would occur in no more than 0.17 sec and complete plugging in no more than 0.57 sec. (auth)

8340

SOME NOTES ON THE STRENGTH OF THE ENRICO FERMI REACTOR VESSEL STRUCTURE. F. R. Beyer (Atomic Power Development Associates, Inc., Detroit). Trans. Am. Soc. Mech. Engrs. Ser. A 81, 66-72(1959) Jan.

The place and function of the reactor vessel in the Enrico Fermi Atomic Power Plant are described. Strain gaging during hydrostatic testing gave limited results. High-temperature strain gages will be used during simulated-service tests prior to start-up. A successful high-temperature strain-gaging technique for long-time static applications up to at least 900°F is developed and described. (W. L. H.)

8863

EVALUATING HIGH-TEMPERATURE LIQUID PUMPS. A. F. Erwin. Allis-Chalmers Elec. Rev. 23, 3rd Qtr., 7-11(1958).

The liquid metal pump requirements for four major nuclear power reactor installations and two high-temperature liquid reactor plants are discussed. Details are given of the two types of pumps for extensive

testing as prototype primary and secondary coolant circulating units for the 75 Mw sodium-graphite reactor, the hermetically sealed and electromagnetic pumps selected for EBR-II, the mechanically sealed pumps selected for the 100 Mw Enrico Fermi breeder reactor plant, and the pumps considered for the LMFRE project, the 10 Mw SDR project for Alaska, and the 240 Mw molten-salt homogeneous-type reactor. (auth)

8982 TID-7562(p.78-135)

Nuclear Products-Erco Div., ACF Industries, Inc., Washington, D. C.; Argonne National Lab., Lemont, Ill.; Battelle Memorial Inst., Columbus, Ohio; General Electric Co. Hanford Atomic Products Operation, Richland Wash.; Oak Ridge National Lab., Tenn.; and Sandia Corp., Albuquerque, N. Mex.

WELDING PROBLEMS AND DEVELOPMENT PROGRAMS. Willis Groth, Cecil Stone, David C. Martin, E. B. Lavelle, P. Patriarca, W. J. Leonard, and C. F. Bild. p.78-135 [of] MINUTES OF THE SEVENTH ANNUAL ATOMIC ENERGY COMMISSION WELDING CONFERENCE HELD IN CHICAGO, ILLINOIS, NOVEMBER 6-8, 1957. 57p.

Brief reports are given of current welding techniques at various sites: ACF Industries. Information is given on welding Be-Cu, ultra-high strength steels, and 7075 Al. ANL. Welding problems in reactor structures are discussed. The motor arc process for 0.020-in. thick Al cans, the automatic arc spot welding gun for EBWR fuel assemblies, and the techniques used for the EBR-1, Mk III core are described. BMI. Experience in welding end caps on various fuel elements, welding Ta, and ultrasonic welding of Nb and Mo to themselves and to each other is given. A program for devising a large specimen for evaluating various types of stress relieving is discussed. Hanford Works (General Electric). Objectives of welding programs in the various departments at Hanford are outlined. Current research in inert-gas-shielded tungsten arc welding of carbon steel pipe and aluminum pipe is briefly reported. Other welding projects in progress are: substitution of N₂ for Ar purge gas in welding stainless steels, welding in high vacuum, and electron beam welding. ORNL. Results of studies of oxidation resistance of high-temperature brazing alloys are given. Thermal expansion data for several brazing alloys and structural alloys are also given. Work on joining TiC and WC to metals is summarized. The brazing characteristics of several types of experimental Ni-Mo alloys for high-temperature service are given. Data are presented in comparison of the tensile properties of Inconel, Hastelloy, and INOR-8 weldments. Welding of type 347 stainless steel, Ti, Zircaloy-2, and Ti 110AT, in connection with the homogeneous reactor project is discussed. Sandia Corporation. The welding equipment and welding standards and qualifications at Sandia are described. Information is given on welding of Boral sheet, and on joining stainless steel wire to U. (T. R. H.)

9040

PLUTONIUM. Bernhard Blumenthal (Argonne National Lab., Lemont, Ill.) J. Metals 11, 125-9(1959) Feb.

Information is presented on the use of plutonium as a reactor fuel. Discussions are included on the following: breeding cycles; recovery, toxicity, and properties of plutonium; fuel element assemblies for Experimental Breeder Reactor-II; recycling of plutonium; and fuel pin refabrication. Crystal structure data and transformation temperatures for plutonium phases are included. (J. H. M.)

9055

METHODS FOR END CAPPING ZIRCONIUM-CLAD

REACTOR FUEL PINS J J Vagi, R L Koppenhofer, and D C Martin (Battelle Memorial Inst, Columbus, Ohio) Welding J (N Y) 38, 78s-84s(1959) Feb

As part of a development program conducted for Atomic Power Development Associates, several methods for attaching end caps to fuel pins were studied. The method providing the best seal involved the use of a fuel pin in which the core was recessed to receive an end-cap insert that was attached and sealed by arc welding. The strength of the resulting end cap-to-fuel pin joint was low because all loading was supported only by the 4-mil thick cladding. Resistance-upset butt welding using sleeves to limit outward upset gave the strongest end cap-to-fuel pin joints. However, contamination of the fuel-pin surface or exposure of the core alloy at the weld joint was observed in nearly every joint made by resistance welding. The joint design giving most consistent results was one in which the fuel pin was beveled slightly and welded to a flat end cap. Very little end preparation of the fuel pins was required for resistance welding. Of the other methods explored, the mechanical attachment of end caps by a recessed-joint design and a grooved end-cap insert, offered some promise. With this type of joint design only a mechanical seal is formed and only about half the strength of end caps attached by arc welding was obtained. (auth)

9367 IGR-TN/CA-958

United Kingdom Atomic Energy Authority. Industrial Group. Capenhurst Works, Capenhurst, Ches, England

A STATIC TRANSDUCER FOR INDICATING THE POSITION OF CONTROL RODS IN THE FAST REACTOR.

A W Fenton 1959 17p

A static electrical transducer system is described for the remote indication of the position of a control rod in a sealed high-temperature reactor vessel over a range of 25 in. to an accuracy of ± 0.01 in. (J.E.D.)

9395

NEW METHOD FOR THE ACCOMPLISHMENT OF NUCLEAR REACTIONS IN A BREEDER REACTOR

(To Brown, Boveri & Cie AG, Mannheim) German Patent DAS 1032 432 Atompraxis 5, 42(1959) (In German)

It was proposed to load a breeder reactor after attainment of stationary operating conditions only with fission-free breeder elements and to use this breeder material, after a given saturation ratio was produced in them by neutron irradiation, in the same reactor as fuel elements without previous withdrawal or other preparation. The time consuming and costly chemical preparation of the fissionable materials would then be avoided.

9397

NUCLEAR REACTOR WITH CHARGE OF HOMOGENEOUSLY CAST BREEDER ELEMENTS (To Dr Rudolf Schulten, Mannheim, Ger.) German Patent DAS 1034 784 Atompraxis 5, 42(1959) (In German)

A reactor was proposed in which the breeder mantle would consist of a charge of homogeneous cast breeder elements so that the breeder element has the same shape as the fuel elements. By this method it would be possible to use the breeder element after its irradiation immediately for the charging of the fuel elements.

9479 PRDC-TR-17

Power Reactor Development Co., Detroit.

MONTHLY TECHNICAL REPORT [FOR] NOVEMBER 1958. 34p. Contract AT(11-1)-476. \$6.30(ph), \$3.00 (mf) OTS.

A summary of the first phase of the program directed toward determining the effects of rapid temperature changes on irradiated Zr clad Mo-U fuel pins is presented. It was found that cladding and core rupture occur when the Fermi Reactor fuel pins are heated to 1850°F for short periods. Fuel Engineering. Enriched U scrap is being reprocessed for the CP-5 irradiation program. Materials. Calculations of burnup and irradiation temperatures of cermet samples irradiated in the MTR are tabulated. Samples of irradiated specimens were heated to 1295°F for 100 hours. For any given time at exposure temperatures, fuel density decreases are directly proportional to the square of the burnup. An investigation of the properties of cast boron steel was completed.

Although the steel is not ductile and is difficult to weld, it is suitable for shielding. Nuclear Engineering. Phase I of the critical experiment was completed, and an effective delayed neutron fraction was calculated. The maximum and average concentrations of U^{237} that build up with irradiation were calculated for core, axial blankets, and radial blankets. The (n,2n) reaction accounts for the majority of the U^{237} formed. Shielding design for the spent fuel decay facility was completed. Electrical and Instrumentation. The system designed to ensure the proper sequence in fuel handling was manufactured. Liquid Metal and Steam Systems. The pressure, temperature, and thermal stress analyses for steam and water head were completed and set-up for computer analysis. Test Operations. Installation of the leak detector pan, tabular heaters, heater shielding, graphite blocks, insulation, and lagging was completed on the lower and upper reactor vessels. Steam generator tests were conducted, and results are being evaluated. An environmental radioactivity report for the Fermi Reactor site is included, as well as reports for 16 other locations around the reactor. (For preceding period see PRDC-TR-16.) (J.R.D.)

9480 PRDC-TR-18

Power Reactor Development Co., Detroit.

MONTHLY TECHNICAL REPORT [FOR] DECEMBER 1958. 24p. Contract AT(11-1)-476. \$4.80(ph), \$2.70 (mf) OTS.

The status of the PRDC meltdown studies and the system analysis programs is given. Design and fabrication progress of the major components being procured by APDA for the Reactor Components Test is shown. Progress made by various subcontractors is summarized on fuel engineering, materials, nuclear engineering, reactor vessel engineering, handling, instrumentation, liquid metal and steam systems, and test operation. Environmental radioactivity reports for the Enrico Fermi Reactor are included. (For preceding period see PRDC-TR-17.) (W.D.M.)

9879 AECU-4017

National Sanitation Foundation, Ann Arbor, Mich.

COLLECTION AND ANALYSIS OF RADIOACTIVE MATERIAL FROM SIBLEY QUARRY. Walter F. Snyder and John V. Nehemias. Dec. 29, 1958. 19p. For Detroit Edison Co. \$3.30(ph), \$2.40(mf) OTS.

Samples of various media were collected at several points in southwestern Michigan, Ohio, and Ontario during 1958 and analyzed for radioactivity as part of a general pre-operational radiological survey of the environment of the Enrico Fermi Nuclear Power Reactor. Media sampled included earth, dust, rain, surface water, underground water, surface sediment, benthos, fish, vegetation, and mammals. Data provide an indication of the existing levels of radioactivity present in environmental media in the area under study and their variations with time and location. It is apparent from the nature of the radioactivity found, as well as from the

distribution, that the material originated at nuclear test detonations in the Pacific and certain other remote areas. Samples were collected from Sibley Quarry after readings notably higher than background were reported. Sampling procedures and laboratory procedures used in an evaluation of these samples are outlined. Data are tabulated. Results are compared with radioactivity found in similar samples collected in the area. The separation of radionuclides by elements provides convincing evidence that the observed radioactivity is due to fission products from detonations occurring a few days to a few months prior to October 10, 1958. (C.H.)

10588 RDB(C)/TN-48

Gt. Brit. Culcheth Labs., Culcheth, Lancs, England. THE EFFECT OF FISSION PRODUCT ACCUMULATION ON THE PHYSICAL CONDITION OF FAST REACTOR FUEL ELEMENTS. K. Q. Bagley and D. S. Oliver. July 17, 1953. Decl. Nov. 20, 1957. 36p. (FRDC/P-41).

Data are presented on the chemical, physical, and alloying characteristics of the fission products likely to be formed in the fast reactor. The distribution of these fission products at the end of 40,000 Mwd irradiation is discussed, together with some of the consequences of their formation. Recommendations are made for the investigation of the expected phenomena and for the development of fuels which are less likely to undergo severe irradiation damage. (auth)

10633 IGR-R/R-221

United Kingdom Atomic Energy Authority. Industrial Group H. Q., Risley, Lancs, England. APPLICATIONS OF THE RAYLEIGH-RITZ APPROXIMATION. PART II. UPPER BOUNDS FOR THE CRITICAL LAPLACIANS OF SOME COMPLEX SOLIDS. R. T. Ackroyd and M. A. Perks. Dec. 1957. 29p. \$0.63(BIS).

The method of optimized parameters which was introduced in Part I is used to determine critical Laplacians and flux plots for a finite elliptical cylinder, a finite hexagonal prism, and a two-region core finite circular cylinder, all fully reflected. A typical fast reactor is used as a model for calculations in all three cases. The two-region circular reactor result is compared with the finite difference solution obtained using a computing machine. (auth)

10708

THE KINETICS AND STABILITY OF FAST REACTORS WITH SPECIAL CONSIDERATIONS OF NONLINEARITIES. Armin Henry Sandmeier. Thesis, Swiss Federal Institute of Technology, Zurich. 1959. 88p.

The linear kinetics of fast reactors is discussed in the first two chapters, chapter I dealing with zero power kinetics and chapter II with linear power kinetics. The third chapter treats the influence of nonlinearities on the dynamics. (T.R.H.)

10714 AECU-4058

Atomic Power Development Associates, Inc., Detroit. HEAT TRANSFER FROM SPENT FUEL AND BLANKET SUBASSEMBLIES DURING TRANSFER FROM REACTOR TO DECAY STORAGE FACILITY. Technical Memorandum No. 2. L. L. Kintner. May 15, 1957. 60p. Contract [AT(11-1)-476]. \$9.30(ph), \$3.60(mf) OTS.

During the transfer of spent fuel and blanket subassemblies from the reactor to the decay storage facility, the subassemblies must be moved through an inert gas. The heat generation due to the decay of fission products is sufficiently large that provisions must be made for

cooling the subassembly when in the gas. The transfer equipment was designed for conduction, radiation, and natural convection cooling only. Various steps in the transfer process are explained, and computations on heat generation and transfer are given. (W.D.M.)

10716 AECU-4061

Atomic Power Development Associates, Inc., Detroit. SOME NUCLEAR CONSIDERATIONS IN FUEL RE-LOADING AND CONTROL. Technical Memorandum No. 8. E. L. Alexanderson. Aug. 12, 1957. 15p. Contract [AT(11-1)-476]. \$3.30(ph), \$2.40(mf) OTS.

The APDA reactor will be operated initially with 92 cents of control reactivity. This will allow 20 cents for temperature over-ride, 30 cents for control margin, and 42 cents for fuel burn-out and element growth. There are 6 different reasons for requiring a control margin, and 30 cents may not be adequate. Thus, even less than 42 cents may be available for shimming, and careful scheduling of subassembly replacements is necessary. Several possible power and load factor conditions are assumed and charts presented to show cycling of core and part of the blanket fuel. Individual fuel subassembly removal times are chosen so as to obtain very close to maximum allowable burn-up and to minimize cycle to cycle variations in reactivity. (auth)

10717 AECU-4063

Atomic Power Development Associates, Inc., Detroit. SURVEY OF THE RADIATION LEVELS IN THE CONTAINMENT VESSEL OF THE ENRICO FERMI ATOMIC POWER PLANT. PART I. GAMMA RADIATION LEVELS IN THE EQUIPMENT COMPARTMENT DUE TO PRIMARY SODIUM ACTIVITY AND ASSOCIATED FISSION PRODUCT CONTAMINATION. Technical Memorandum No. 16. W. F. Chaltrton and H. E. Hungerford. Sept. 4, 1958. 51p. Contract [AT(11-1)-476]. \$9.30(ph), \$3.60(mf) OTS.

The results are presented of a gamma radiation survey of the equipment compartment. The following six radioactive sources within the sodium were used: Na^{24} , Na^{22} , and four cases of fission product contamination. The four cases of fission product contamination are divided as follows: Case I—fission products from uranium occurring naturally in the sodium, Case II—fission products leaking into the sodium at 6×10^{-7} of the rate of their production, Case III—fission products from dissolution of one complete pin in the primary coolant, and Case IV—fission products from the dissolution of 14 complete pins. Two conditions of the system were considered under various operating and shutdown procedures: (1) normal, and (2) drained. The calculations were made by first dividing the primary system into various discrete sources, and then computing the contributions of each source to each point on a coordinate system representing the equipment compartment. The various conditions of the system which were considered and results of the calculations are presented. Entry into the equipment compartment under the various assumed conditions is discussed. (auth)

10718 AECU-4064

Atomic Power Development Associates, Inc., Detroit. DESIGN BASIS FOR THE FUEL HANDLING AND DECAY STORAGE FACILITY. Technical Memorandum No. 18. W. W. Kendall, R. A. MacLeod, and D. D. Peden. Nov. 14, 1958. 37p. Contract [AT(11-1)-476]. \$6.30(ph), \$3.00(mf) OTS.

The design basis is given for the fuel handling and decay storage facility. The units to be handled include the core and blanket subassemblies, the control rods,

the neutron source, and the surveillance specimens. The handling operations include receiving of new units, inspection before irradiation, inspection after irradiation, radioactive decay in a pool of water, disposal of waste material, and the shipment of radioactive material from the plant site. (auth)

10721 APDA-128

Atomic Power Development Associates, Inc., Detroit.
THERMAL BOWING OF CORE SUBASSEMBLIES.
Feb. 20, 1959. 41p. \$6.30(ph), \$3.00(mf) OTS.

The potential bowing problem in a core of the Fermi type is associated with the core subassembly bowing which could result in relatively large core geometry changes if no steps are taken to prevent it. This problem in the design of the Enrico Fermi Reactor is dealt with, and the method which has been adopted to avoid a positive temperature coefficient of reactivity is discussed. (W.D.M.)

11371

TIME CONSTANT OF CYLINDRICAL FUEL PIN WITH AXIAL COOLANT FLOW. H. A. Sandmeier and D. M. O'Shea (Argonne National Lab., Lemont, Ill.) Nuclear Sci. and Eng. 5, 186-9(1959) Mar.

From the time dependent heat conduction and temperature distribution, an expression is derived for the time constant in a cylindrical fuel pin and cladding with axial coolant flow. The power production and the inlet temperature are functions of time. In the radial direction perfect mixing of the coolant is assumed. The average coolant temperature in a region is the average between inlet and outlet temperature assuming a linear rise in the axial direction. The set of partial differential equations can be solved by means of Laplace transform. The reciprocal of the roots of the characteristic equation for the temperature in the transform domain represents the time constants. The smallest root represents the dominant transient time constant. This dominant time constant is compared with a qualitative expression for the thermal relaxation time of a reactor after a power change given by Bethe. The numerical example used is a fuel pin in EBR-I Mark III in flowing NaK coolant at a core power generation of 1 Mw at various coolant flow conditions. (auth)

11819 ANL-5975

Argonne National Lab., Lemont, Ill.
METALLURGY DIVISION ANNUAL REPORT FOR 1958.
Mar. 1959. 125p. Contract W-31-109-eng-38. \$2.75 (OTS)

Engineering Metallurgy. The fuel matrix for the TREAT reactor consists of a uniform dispersion of U_3O_8 in graphite. Some of the developments that led to the fabrication procedure are described. The fabrication of the fuel elements for the ALPR has been completed. The elements are Al-clad thin plates arranged in parallel assemblies, each assembly consisting of 9 plates. The fuel core material is an Al-base alloy containing 17.5 wt. % highly enriched U-2.0 wt. % Ni-0.5 wt. % Fe. Fabrication procedures are summarized. In support of the development of fuel material for the EBR-II, an experimental survey was made of the engineering properties of the reference alloy: 70 wt. % U-20 wt. % Pu-10 wt. % fissium. The specimens used were machined from injection-cast rods, either 0.145 inch or 0.250 inch in diameter by 15 inches long. Development work was initiated for the manufacture of the highly enriched U core for the Fast Source Reactor. Irradiation evaluations of various experimental fuel and control rods consisting of U-Mo, U-Th, Al- U_3O_8 , Al-Pu, and ThO_2-UO_2 are presented. Preparation and properties of experimental

ceramic fuel materials UC, UO_2-ZrO_2 , CaO, UO_2-ZrO_2-MgO , $UO_2-Al_2O_3$, UO_2-MgO , ThO_2-UO_2 are presented. The development of water-corrosion-resistant U-Ti, Al and Mg alloys is presented. Developments in nondestructive testing are reported. The development of steam generator and superheater for EBR-II is presented. Research Metallurgy Self-diffusion in gamma U has been measured at four temperatures in the region of stability of the gamma phase. The studies on the operative deformation mechanism of single crystals upon compression have been extended to 600°C. Work on determining the principal elastic moduli of single crystals of alpha U at room temperature has been completed. Phase diagrams of U binary and ternary alloys with the following fission product elements have been investigated: U-Rh, U-Ru, U-Mo, and U-Ru-Mo. The preparation of high-purity Pu by fused salt electrolysis is presented. The sintering mechanisms of Al_2O_3 single crystals are reported. The occurrence of the CsCl-type structure in transition element alloys was investigated. Polarization measurements were made in distilled water to determine the relationship between the change in potential and current flowing at corroding metal surfaces. Low-temperature Al corrosion tests are reported. The mechanism of aqueous corrosion of Zr is being investigated. (W.L.H.)

12255 ANL-LAT-86

Argonne National Lab., Lemont, Ill.
MEASUREMENT OF NEUTRON FLUX IN THE EBR.
P. J. Persiani and G. R. Ringo. May 16, 1952. 6p.
\$3.30(ph), \$2.40(mf) OTS.

A study of the neutron flux distribution in the EBR was made utilizing the neutron activation of Au and P foils. The 270-min activity on the P foils coming from an (n,p) reaction with a 1-Mev threshold was taken as a measure of the fast neutrons and the Au activity as a measure of the slower ones. The foils were standardized by activation on a U^{235} converter plate. The positions in the reactor measured are shown, and the results are tabulated. The neutron leakage was calculated to be 3.5% of production from measurements of the thermal neutron current in the thermal column. (auth)

12265 NDA-014-189

Nuclear Development Corp. of America, White Plains, N. Y.
CALCULATION OF THE REACTIVITY EFFECT OF CONTROL ROD MOVEMENT IN THE ENRICO FERMI REACTOR. J. H. Ray. Oct. 1, 1957. 19p. \$3.30(ph), \$2.40(mf) OTS.

Two-dimensional multigroup problems were prepared and solved to determine the effect on reactivity of control rod movement in the Enrico Fermi Reactor. Part of the problems were designed to investigate the effect of safety rod movement, while others were designed to examine the effect of shim rod movement. In addition, the effect of rod movement was calculated by hand using a set of danger coefficients for this reactor. (auth)

12286 AECU-4094

Atomic Power Development Associates, Inc., Detroit.
BASIC CONCEPT OF THE INERT GAS SYSTEM FOR PRIMARY SODIUM. Technical Memorandum No. 9.
W. W. Kendall and W. L. Chase. Feb. 3, 1958. 33p.
Contract [AT(11-1)-476]. \$6.30(ph), \$3.00(mf) OTS.

The design values which were selected and the reasons for selecting them are given for the Fermi Reactor inert gas system. The concept of the system which is to be used to prepare the detailed flow sheet and to design the system components is described. The gas is to be either helium, argon, or nitrogen. (W.D.M.)

12287 AECU-4095

Atomic Power Development Associates, Inc., Detroit.
SHIELDING ON 30-INCH PIPE. Technical Memorandum
No. 10. John J. Morabito. Feb. 5, 1958. 9p. Con-
tract [AT(11-1)-476]. \$3.30(ph), \$2.40(mf) OTS.

The materials of construction and the design of the
shielding for a 30-inch pipe are presented. A calcium
borate and a borated diatomaceous earth-luminite con-
crete performed equally satisfactorily and both are con-
sidered for the shielding of the pipe. (W.L.H.)

12288 AECU-4096

Atomic Power Development Associates, Inc., Detroit.
DESCRIPTION OF THE INERT GAS SUPPLY SYSTEM. Technical
Memorandum No. 11. W. L. Chase. Mar. 7,
1958. 6p. Contract [AT(11-1)-476]. \$1.80(ph), \$1.80
(mf) OTS.

A description of the inert gas supply system for the
Fermi Fast Breeder Reactor is presented. Design con-
ditions are given and the supply and purification ar-
rangements are discussed. (W.D.M.)

12289 AECU-4097

Atomic Power Development Associates, Inc., Detroit.
LEAKAGE FROM INERT GAS SYSTEM. Technical
Memorandum No. 14. W. W. Kendall and W. L. Chase.
Apr. 4, 1958. 7p. Contract [AT(11-1)-476]. \$1.80(ph),
\$1.80(mf) OTS.

Leakage of inert gas into the containment building is
examined from the standpoint of sources and rates.
Methods of calculating maximum permissible leak rates
are presented as well as graphs showing permissible
total leak rate as a function of building ventilation. The
ventilation system described covers only radioactive
requirements. (J.R.D.)

12290 AECU-4098 and ADD. 1

Atomic Power Development Associates, Inc., Detroit.
REACTOR BUILDING VENTILATION. Technical
Memorandum No. 17. R. B. Shumaker. Oct. 3, 1958
14p. ADDENDUM 1. Dec. 8, 1958. 9p. Contract [AT
(11-1)-476.] \$4.80(ph), \$2.70(mf) OTS.

This report and addendum 1 were issued separately,
but are cataloged as a unit.

A review is presented of the Fermi Power Plant build-
ing heating, cooling, and ventilation. The heating and
cooling loads and operation pressures in the areas above
and below the operating floor are summarized. Design
features such as ventilation rate and operation of shut-
off valves are included. (W.L.H.)

13026 SCR-76

Sandia Corp., Albuquerque, N. Mex.
RADIATION EFFECTS TESTING AT THE LOS ALAMOS
GODIVA II A FACILITY. L. J. Zipprich, comp. Apr.
1959. 20p. \$0.75(OTS).

Revision of an Informal Memorandum "Godiva II—Its
Availability and Suitability for Radiation Effects Tests."
W. J. Buckalew, comp

The Los Alamos Scientific Lab., recognizing the
unique nuclear environment offered by Godiva II radia-
tion bursts, has agreed to make this facility available
without charge to DOD contractors on a general sched-
ule of 2 days each month for the performance of qual-
ified radiation effects tests. The procedure for reserv-
ing time and space at the facility is outlined. A brief
description of the facility is given. (W.D.M.)

13361 BNL-549

Brookhaven National Lab., Upton, N. Y.
REPROCESSING OF POWER REACTOR FUELS The

Enrico Fermi Fast Breeder Reactor Fuel Progress
Report No 2 C W Pierce Mar 1959. 8p. \$0.50
(OTS)

A summary of work on the problems of reprocessing
Enrico Fermi Fast Breeder Reactor between July 1958
and January 1959 is given. Dissolution experience with
the Zr-clad U—Mo fuel pins is related. Corrosion of
the stainless steel waste evaporator was studied. Nu-
clear hazards, flow sheet design and chemistry, and
waste handling studies are also reported. (For preced-
ing period see BNL-511) (T R H)

13514 ANL-5893

Argonne National Lab., Lemont, Ill
NONDESTRUCTIVE TESTING OF EBR-I MARK III
FUEL ELEMENTS AND COMPONENTS. W. N. Beck,
C. J. Renken, R. G. Myers, and W. J. McGonnagle,
May 1959. 35p. Contract W-31-109-eng-38. \$1.00
(OTS)

Ultrasonic and eddy current methods were used to
inspect EBR-I Mark III fuel elements and components.
Ultrasonic techniques were used to inspect for homoge-
neity of the casting, bonding of the core to the clad on
the extruded rod, bonding of the Zircaloy spacer disk
to the uranium, and cracks in the Zircaloy rod used for
end caps. Eddy current techniques were used to meas-
ure the cladding thickness on the extruded rods and to
inspect the zirconium wire used for spacers on the
completed fuel element. (auth)

13708 LA-2185

Los Alamos Scientific Lab., N. Mex.
HIGH ENERGY NEUTRONS IN A COLLIMATED BEAM
FROM THE LOS ALAMOS FAST PLUTONIUM REAC-
TOR. Bob E. Watt, Feb. 1951. 30p. Contract W-
7405-eng-36. \$4.80(ph), \$2.70(mf) OTS.

Estimates of the excitation energies of fission frag-
ments suggested that it would be useful to search for an
upper limit to the neutron energy spectrum. The work
reported was undertaken to extend the fission spectrum
as far as was permitted by the beam intensity available
from the Los Alamos Fast Plutonium Reactor. A sec-
ondary objective was to improve the accuracy of the
spectrum reported in Los Alamos Scientific Laboratory
Report LA-718. The spectrum reported there was de-
termined from fissions induced by thermal neutrons.
Neutrons were observed in the energy range of 24 to
25 Mev, implying fragment excitation energies greater
than 30 Mev. In the spectrum no change of slope is
evident, suggesting that fragments of still higher ex-
citation are reasonably abundant. (auth)

14037

METHOD AND APPARATUS FOR IMPROVING PER-
FORMANCE OF A FAST REACTOR. L. J. Koch (to
U. S. Atomic Energy Commission). U. S. Patent
2,870,076. Jan. 20, 1959.

A specific arrangement of the fertile material and
fissionable material in the active portion of a fast re-
actor to achieve improvement in performance and to
effectively lower the operating temperatures in the
center of the reactor is described. According to this
invention a group of fuel elements containing fissionable
material are assembled to form a hollow fuel core.
Elements containing a fertile material, such as depleted
uranium, are inserted into the interior of the fuel core
to form a central blanket. Additional elements of fertile
material are arranged about the fuel core to form outer
blankets which in turn are surrounded by a reflector.
This arrangement of fuel core and blankets results in
substantial flattening of the flux pattern.

14054 APDA-124

Atomic Power Development Associates, Inc., Detroit.
ENRICO FERMI ATOMIC POWER PLANT. Jan. 1959.
235p. \$3.50(OTS).

The Enrico Fermi Atomic Power Plant is described with emphasis on the fast breeder reactor and heat transport systems. In addition to description and illustration of the physical plant and its operation, the safety considerations and waste disposal arrangement are described. (T.R.H.)

14066 NP-7383

Central Research Inst. of Electric Power Industry.
Nuclear Power System Development Committee,
Tokyo.

CRITICAL BREEDING FAST REACTOR C.B.F.R.—ITS BASIC CONCEPT AND PRELIMINARY CALCULATIONS. Matsujiro Oyama, Minoru Takahashi, Hiroshi Ogawa, Ryohei Miki, and Katsuaki Ono. Oct. 1, 1958. 34p.

Critical Breeding Fast Reactor is proposed to burn out almost all U-238 atoms contained in natural uranium and generate electricity economically without any development of complicated and expensive technology such as chemical processing or plutonium recycling. Main features of C.B.F.R. are given. About 10% enriched uranium is submerged in Na coolant. Fuel metal will be protected by strong and thick pipes made of steel (or more preferably nickel). Conversion ratio (as a reactor whole) will be kept always equal or nearly equal to 1. Fissile materials in the reactor neither decrease nor increase excessively. Sufficient quantity of fissionable material to sustain the reactivity is kept throughout the long reactor operation time, and thus it can stand away from the necessity of chemical processing or fuel recycling. Damaged protecting pipes will be exchanged mechanically by new ones, on the other hand fuel metal itself will neither be refreshed nor chemically processed. Large burn up value will be obtained. Even in primary stage of development of this reactor (rather small or mediate sized reactor), 200,000 to 300,000 MWD/ton irradiation will be possible. In an advanced stage of development of this reactor (large sized reactor) it will be possible to burn out more than half of U-238 atoms in natural uranium. The make up process of fuel element is very simple, resulting in low fabrication cost. Almost all part of fuel cost will come from the initial charge enriched uranium, and this part of fuel cost coming from initial charge (in form of UF_6) is estimated to be 1.2 mill/kwh (at 80% plant factor and 6% annual charge rate). (auth)

14067 PRDC-TR-19

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] JANUARY
1959. 32p. Contract AT(11-1)-476. \$6.30(ph), \$3.00
(mf) OTS.

The core isothermal Na temperature coefficient at 675°F was calculated using a Na worth value of $-4.66 \times 10^{-6} \Delta k/k/^\circ C$ which was determined from the homogeneous critical experiment. This value is to be compared with the current value of $-7.1 \times 10^{-6} \Delta k/k/^\circ C$ which was obtained from past multigroup calculations. The lower value of sodium coefficient is compensated by a new value of the fuel temperature based on axial expansion of the fuel. Axial expansion coefficient data are included. The transfer rotor assembly was shop tested and met all specifications. Preliminary tests were completed on a transparent plastic model vapor trap. Observations were recorded with high-speed motion pictures. Exploratory runs were made on the test flow loop to check pump performance, and instrumentation core pins were found to impede flow because pump

packing materials were adhering to them. Changes were made in the bird cage by moving the pins to the top. A chart showing progress on major components is included as well as environs monitoring radioactivity tables. (For preceding period see PRDC-18.) (J.R.D.)

14068 RIC-2/6

United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England.

A TRANSFER FUNCTION ANALYSER FOR THE DOUNREAY FAST REACTOR. D. Harrison. Dec. 1957. 14p.

Equipment developed to obtain reactivity change information for the Dounreay Fast Reactor is described. The technique of steady-state analysis was adopted. A sinusoidal perturbation of reactivity is applied to the reactor by means of a rod, formed of one half cylinder of absorber material (B^{10}), which extends into the reactor core and is rotated about the cylinder axis at an angular velocity corresponding to that at which the measurement is to be made. The measured response will be that of the neutron flux, and the sensing device is an ion chamber. The basic circuit requirements are listed. The modulating device is a carrier system with a magstrip element. Chopper stabilized d-c computer type amplifiers are used for the ion chamber current, the recorder output, and the in-phase and quadrature channel amplifiers. The a-c amplifier is of conventional design and makes use of transformer coupling into the synchro-resolver. This equipment together with stabilized power supplies and certain relay and servo circuits will be housed in two standard 6 ft P.O. type racks within the Dounreay sphere. (M.H.R.)

14359 TID-7556(p.160-7)

Argonne National Lab., Lemont, Ill.
SHIELDING WINDOW DESIGN FOR THE EBR-II PROCESS BUILDING. K. R. Ferguson and L. M. Safranski. p.160-7 [of] SIXTH HOT LABORATORIES AND EQUIPMENT CONFERENCE, MARCH 19-21, 1958, INTERNATIONAL AMPHITHEATRE, CHICAGO, ILLINOIS. 8p.

The shielding window design for the facility for processing and refabrication of fuel elements for the Experimental Breeder Reactor No. II is described. The window is designed for a gamma radiation intensity up to 10^6 r/hr for eight hours per day. Steel shutters will shield the window when it is not in use. The design reasons for the window configuration and the choice of materials are discussed. Operation of the plant in an inert atmosphere requires a gas tight seal at the window aperture. The design details of the seal and test data on its performance are included. Also, data are included on the computed light transmittance of the window before and after a 10^{10} r exposure. (auth)

14915

ATOMIC ENERGY PILE. E. D. Courant (Canada)
British Patent 801,651. Nuclear Eng. 4, 238(1959) May

The pile has a core dimensioned slightly below critical size. The control members contain an amount of fissile material equivalent to at least 1.5% of their weight and larger than the amount of fissile material in the core substance. Initially, the control rods need only be inserted a short distance to bring the multiplication factor to a value exceeding 1. But, when the pile activity drops, the rods are inserted farther. The design is of particular interest in connection with breeder piles.

14924 AECU-4161

Atomic Power Development Associates, Inc., Detroit.
SODIUM-AIR ACCIDENT STUDY FOR A SODIUM

COOLED REACTOR. Technical Memorandum No. 20. Edward Garelis. Apr. 6, 1959. 22p. \$4.80(ph), \$2.70 (mf) OTS.

The design requirements of a containment building for a sodium-cooled reactor have been investigated from the point of view of a hypothetical sodium-air reaction. In particular, this study was directed toward the containment building of the Enrico Fermi Atomic Power Plant. The hypothetical accident assumes a major leak in the sodium system with a concurrent or earlier failure of the ventilation and cooling system, which allows air to enter the areas around the sodium system which are normally maintained in an atmosphere inert insofar as sodium burning is concerned. The resulting sodium pool, equivalent in burning surface to the fraction of the cross sectional area of the containment building, which is flooded, burns at a rate depending upon oxygen concentration and gas temperature in the containment building. A series of experiments were carried out to determine this burning rate and its mechanism. The resulting gas temperature and pressure due to the sodium-air incident were determined by solving a system of equations based on time dependent heat balances on the containment building and its contents. The assumed incident gives a maximum gas pressure of about 26 psig and a corresponding gas temperature of 1115°F. This pressure is well below the design pressure of 32 psig for the containment vessel. In addition to the study of the sodium-air reaction in the assumed incident, the theoretical maximum pressure and temperature for the constant volume adiabatic sodium-air reaction is also discussed. (auth)

14931 NP-7539

United Kingdom Atomic Energy Authority. Industrial Group. Dounreay Experimental Reactor Establishment, Caithness, Scotland. FAST REACTOR NEWSLETTER. J. L. Phillips, comp. May 1959. 6p.

The testing of all flasks and handling equipment is almost complete. Modifications necessary to ensure satisfactory operation have been made. The natural uranium elements have been loaded into the core. A total of 74 tons of liquid metal has been melted and is held in the dump tanks. Filling the primary circuit with liquid metal is due to start soon. Criticality during July is programmed. (A.C.)

15811

MEASUREMENT OF THE DOPPLER TEMPERATURE EFFECT IN AN EBR-1 TYPE ASSEMBLY. W. Y. Kato and D. K. Butler (Argonne National Lab., Lemont, Ill.). Nuclear Sci. and Eng. 5, 320-30(1959) May.

A measurement of the Doppler temperature effect was made in the fast spectrum of a mockup of the EBR-I reactor in ZPR-III. The effect was measured by thermal cycling samples of enriched uranium, natural uranium, and plutonium and detecting the small changes in reactivity. The pile oscillation technique using a resonant detector was employed to measure the small oscillating component of the neutron flux. An upper limit of $0.5 \times 10^{-8} \Delta k / \Delta T$ was obtained for a 506-gm sample of U^{235} in the spectrum of an EBR-I mockup, and for a 235-gm sample of Pu^{239} in a plutonium-fueled assembly of the same configuration. The results for natural uranium were inconclusive. (auth)

16140

THE PRDC CASE. James L. Morrisson (Ginsburg, Leventhal, Brown and Morrisson, Washington, D. C.) and B. John Garrick (Holmes and Narver, Inc., Los Angeles). Nucleonics 17, No. 7, 60-3(1959) July.

Litigation of the PRDC case is reviewed. A construction permit for the PRDC Reactor was requested by the Power Reactor Development Company. A provisional construction permit was issued by AEC. Three labor unions filed petitions to intervene in the licensing proceeding and requested a hearing. The hearing demonstrated that reactor safety issues can be handled by traditional administrative processes. (A.C.)

16224 RDB(C)/TN-66

Gt. Brit. Culcheth Labs., Culcheth, Lancs, England. STRESSES IN THE FAST FISSION REACTOR FUEL ELEMENTS. M. J. Makin. Mar. 8, 1954. 27p.

Stresses likely to arise in the fuel elements of the fast fission reactor due to its design and operation have been calculated using simple elastic theory. The effects of some of these stresses are briefly considered, and recommendations are made as to ways in which some of the more dangerous stresses may be eliminated or reduced. (auth)

16565

INTERNAL INSPECTION OF FAST REACTOR FUEL ELEMENTS. C. D. Reid (United Kingdom Atomic Energy Authority, Capenhurst, Ches., Eng.). Nuclear Power 4, No. 37, 109-10(1959) May.

A proposed inspection program for fuel elements in the Dounreay Fast Reactor is discussed. Prototype apparatus designed to provide optical inspection of the bore of the inner tube is described. (A.C.)

16576

PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957. G. W. Wensch, ed. Washington, D. C., U. S. Atomic Energy Commission. 286p. \$2.00 (GPO).

The purpose of the meeting was to describe the status of the Fast Breeder Reactor Programs and Fast Reactor Safety Programs for the benefit of access permit holders from industry who have expressed interest in these programs. Thirty-one papers were presented by members of the staff of ANL, LASL, PRDC, and USAEC. One paper on the UK Fast Reactor Program was given. (W.D.M.)

16577

ANL FAST REACTOR PROGRAM. L. J. Koch (Argonne National Lab., Lemont, Ill.). p.7-8 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

ANL is conducting a very broad program in fast reactor development. It includes all of the major scientific and engineering disciplines in the Laboratory, and it encompasses all of the phases of power reactor technology as well as fuel processing and fabrication. The major effort is directed toward solid metal fuel, heterogeneous, fast power reactors. The immediate emphasis is on the design, development, and construction of EBR-II. (W.D.M.)

16578

LOS ALAMOS SCIENTIFIC LABORATORY FAST REACTOR PROGRAM. D. B. Hall (Los Alamos Scientific Lab., N. Mex.). p.9-11 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

LASL is concerned with the development of molten plutonium fueled reactors cooled with liquid metal and surrounded by a fertile blanket of uranium. The fundamental properties of Pu alloys are under investigation

with particular emphasis on those with melting points below 800°F and with Pu content in the range 20 to 90 at. %. Associated with the fuel development is a vigorous attack on the important problem of fuel handling and reprocessing. The problems involved in fast reactor safety are being considered. (W.D.M.)

16579

THE POWER REACTOR DEVELOPMENT COMPANY FAST REACTOR PROGRAM. J. F. Anderson (Power Reactor Development Co., Detroit). p.12-14 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

PRDC is a nonprofit membership organization dedicated to an early demonstration of the practical and economical utilization of nuclear energy for the production of electric power. To this end PRDC is engaged in the construction of a full-scale developmental fast neutron breeder reactor, the Enrico Fermi Atomic Power Plant. The over-all responsibility for the design, financing, construction, and operation of the reactor portion of the plant lies with PRDC; however, the cooperative assistance of Atomic Power Development Associates, Inc., The Detroit Edison Company, and the AEC are vital parts of the over-all program. The project responsibilities of these organizations and the highlights of the PRDC project to date are outlined. (W.D.M.)

16580

ENGINEERING DESIGN OF EBR-II. (Reactor and Power Cycle). H. O. Monson (Argonne National Lab., Lemont, Ill.). p.15-22 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

The EBR-II is an integral nuclear power plant, i.e., it includes a complete fuel processing and fuel element fabrication facility in addition to the reactor, the heat transfer systems, and a steam-electric plant. Engineering design and component development are presently nearing completion. Construction of the plant has started, and operation is expected to begin in 1960. The plant as a whole is described very briefly, with particular emphasis on the power cycle. (W.D.M.)

16581

ENGINEERING DESIGN OF EBR-II. (Fuel Process and Fabrication Cycle). M. Levenson (Argonne National Lab., Lemont, Ill.). p.23-8 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

The EBR-II, in addition to demonstrating advances in reactor technology, will demonstrate the use of a closed fuel cycle. The fuel cycle selected for inclusion in the EBR-II plant is designed for remote operation. Fuel processing will be done by melt refining and fuel fabrication by injection casting. The facilities are being designed to achieve maximum flexibility while the equipment is being designed as specific and simple as possible. The 16-sided shielded shop is described. (W.D.M.)

16582

DESIGN OF THE ENRICO FERMI ATOMIC POWER PLANT. A. Amorosi (Atomic Power Development Associates, Inc., Detroit). p.29-62 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

The arrangements of the reactor and power plant are shown. The core is about 2½ feet in diameter and 2½ feet high. The radial blanket is 2 feet thick, and the

axial blanket is 1½ feet thick. The coolant flow is up through the core. The primary system sodium flows by gravity from the reactor through the intermediate heat exchanger into the pump tank. The heat exchangers are arranged to take care of the decay heat by natural circulation. The fuel and blanket subassemblies are shown. Construction progress is discussed. (W.D.M.)

16583

FAST REACTOR SAFETY; EBR-I. F. W. Thalgott (Argonne National Lab., Lemont, Ill.). p.68-76 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

Experience with EBR-I as it applies to fast reactor safety especially with regard to the stability of fast reactors is discussed. The history of EBR-I and its role in the fast reactor program are given. Design features of EBR-I, Marks I, II, and III, are illustrated. A phenomenon observed in EBR-I was an oscillation in power which occurred when operating conditions were made severe. This phenomenon, which led to core meltdown is discussed. (W.D.M.)

16584

MELTDOWN AND ANALYSIS. D. Okrent (Argonne National Lab., Lemont, Ill.). p.77-81 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

The various ways in which core meltdown could occur and the possible courses for a meltdown to follow are discussed. The question of present danger areas in design is considered. Experiments for studying meltdown in the Transient Reactor Test Facility are discussed. (W.D.M.)

16585

ANALYSIS OF PROMPT EXCURSIONS IN SIMPLE SYSTEMS. W. R. Stratton and T. H. Colvin (Los Alamos Scientific Lab., N. Mex.). p.82-92 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

Godiva is a bare, 94% enriched uranium metal assembly which is fabricated in three sections that go together to form a sphere. The total mass of the system is about 55 kg. The system is normally assembled by remote control, and any experiments are monitored by counters, detectors, and closed circuit television. A calculation, coded for a fast computer, that can fairly accurately match the experimental results obtained from the LASL Lady Godiva prompt burst program is described. The code can predict the reactivity, burst width, and maximum rates with reasonable accuracy. (W.D.M.)

16586

NUCLEAR ANALYSIS (CRITICAL EXPERIMENTS WITH ZPR-III). R. R. Smith (Argonne National Lab., Lemont, Ill.). p.96-100 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

ZPR-III was designed specifically for the simulation and study of fast reactor cores fueled either with U²³⁵ or Pu²³⁹. Details of its construction and operation are presented. The types of experiments conducted with this facility and the reasons why these experiments are conducted are discussed. Experiments conducted thus far may be grouped under three general categories according to whether they provide fundamental, engineering, or pure physics data. (W.D.M.)

16587

NUCLEAR ANALYSIS (STATICS ANALYSIS) W B Loewenstein (Argonne National Lab, Lemont, Ill) p 101-7 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL, NOVEMBER 20-21, 1957

Two principal topics, demonstrating the primary functions of a program to predict the static behavior of a system, are discussed. First, where does the critical assembly information fall into this pattern of analysis? Second, what sort of agreement can be obtained between theory and experiment after a few critical experiments have been done? (W D M)

16589

REACTOR DESIGN AND ANALYSIS, EBR-II W R Simmons (Argonne National Lab, Lemont, Ill) p 118-30 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL, NOVEMBER 20-21, 1957

The EBR-II is a sodium-cooled unreflected fast breeder type reactor. The initial core loading will be U-5 wt % fission, and subsequent loadings will be U-20 wt % Pu-10 wt % fission. The expected breeding ratios with these fuels are 1.2 and 1.6 respectively. Initial operation will be limited to 45 Mw. Design is discussed in terms of the reactor, primary coolant system, control and safety rod system, and power density and temperature distribution. (W D M)

16590

REACTOR DESIGN AND ANALYSIS ENRICO FERMI REACTOR J B Nims (Atomic Power Development Associates, Inc., Detroit) p 131-43 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL, NOVEMBER 20-21, 1957

The Fermi Reactor is a sodium-cooled fast breeder fueled with a solid uranium alloy. The core is approximately cylindrical, 2.5 ft in height and diameter. Surrounding the core on all sides is a breeding blanket of depleted uranium of slightly less than 50% normal density. The fuel element design and cooling are discussed. Nuclear characteristics, control system parameters, control reactivity requirements, and the effects of temperature on reactivity are given. (W D M)

16591

FUEL CYCLES, FUEL ELEMENT DESIGN W N McDaniel (Atomic Power Development Associates, Inc., Detroit) p 149-54 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL, NOVEMBER 20-21, 1957

Considerations which enter into the design of fixed fast reactor fuel elements and subassemblies are discussed. The likenesses and differences in the ANL and APDA design approaches are compared. (W D M)

16592

EBR-II FUEL CYCLE (FUEL CYCLE PLANT AND PROCESS) J H Schradtt (Argonne National Lab, Lemont, Ill) p 155-63 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL, NOVEMBER 20-21, 1957

The EBR-II system provides for a closed fuel cycle. Remote fabrication appears to be necessary because of the high decontamination factors required for direct re-fabrication. The fuel cycle steps discussed include cooling for 15 days, transfer to the process building disassembly, decladding, and melt refining and casting of the ingot. (W D M)

16593

REMOTE FUEL ELEMENT FABRICATION AND ASSEMBLY A B Shuck and E S Sowa (Argonne National Lab, Lemont, Ill) p 164-71 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL, NOVEMBER 20-21, 1957

The pyrometallurgically refined billets produced by the melt refining of irradiated EBR-II fuel elements must be remanufactured into fuel elements suitable for use in the reactor. The fuel for the EBR-II is in the form of cylindrical rods 0.144 ± 0.001 inch in diameter by 14.22 ± 0.032 inches long. Casing, cutting and canning the fuel pins are discussed. Reassembly into fuel subassemblies and return to the reactor are reviewed. (W D M)

16595

FUEL HANDLING AND PROCESSING W R Martini (Los Alamos Scientific Lab, N. Mex) p 179-81 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL, NOVEMBER 20-21, 1957

Present research at Los Alamos on handling liquid fuels in a reactor and means of reprocessing molten Pu alloys is intended to result in better realization of the high burnup potentialities of fast reactors. All of the work in this field at LASL is in a very preliminary or exploratory stage, so only an indication of the type of experiments underway is given. (W D M)

16596

FUEL CYCLES, IRRADIATION AND THERMAL STABILITY J H Kittel (Argonne National Lab, Lemont, Ill) p 182-88 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL, NOVEMBER 20-21, 1957

In general, fast reactors operate at higher heat fluxes, higher fuel temperatures, and higher burnups than do most thermal reactors. For economical operation and efficient breeding, the expected lifetime must be relatively long, on the order of at least 2 at % burnup, or 20,000 Mwd/t of fuel. The present status of information obtained in the U.S. on irradiation and thermal effects on fast reactor fuels is reviewed. The data which are presently available have been obtained principally in connection with the development of EBR-I, EBR-II, Fermi Reactor, and the Fast Oxide Breeder (KAPL). (W D M)

16597

COMPONENTS PUMPS R A Jaross (Argonne National Lab, Lemont, Ill) p 199-204 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL, NOVEMBER 20-21, 1957

Several notable advancements in the field of sodium pump technology have been made in the past year. The successful testing of two 5,000 gpm, 40 psi head pumps at ANL has afforded much valuable information and added greatly to the store of experience needed in sodium coolant system design. Extensive tests on these two pumps, one a centrifugal sump type and the other an AC linear induction pump, were made to assist in the selection of pumps for EBR-II. The pumps are described and tests are reviewed. (W D M)

16598

COMPONENTS PIPING SYSTEM COMPONENTS F A Smith (Argonne National Lab, Lemont, Ill) p 205-14 of PROCEEDINGS OF THE 1957 FAST REACTOR IN-

FORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

The EBR-I indicates that reliability in a liquid metals system has been achieved. The components of a sodium piping system are comparable to conventional components. Nearly all the sodium system component items are commercially available and are competitively priced. The present state of development of the piping components other than nuclear cores or steam generators does indicate that there is no serious technical or economical limitation for the reliable operation of sodium piping components at temperatures up to 1,000°F. (W.D.M.)

16599

HEAT EXCHANGERS AND STEAM GENERATORS.

R. H. Anderson (Atomic Power Development Associates, Inc., Detroit) p.215-28 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

The philosophy for the use of, and the safety considerations for, the two coolant systems are presented. For the intermediate heat exchangers the basic over-all design considerations for the Fermi Reactor, EBR-II, and LAMPRE-II are discussed. The general design considerations, operating conditions, design features, and descriptions of the liquid metal steam generators are given. Test units operated by APDA and LASL are described. (W.D.M.)

16600

COMPONENTS—HIGH PURITY SODIUM SYSTEMS.

Joseph E. Kemme (Los Alamos Scientific Lab., N. Mex.) p.229-33 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

A series of small, pumped, sodium loops was used to study the problems peculiar to a LAMPRE-type reactor. The ultimate purpose of these experiments is to discover a means of purifying sodium which also will protect the plutonium fuel. A description and results of the tests are given. (W.D.M.)

16601

ADVANCED PLUTONIUM REACTOR CONCEPTS.

R. P. Hammond (Los Alamos Scientific Lab., N. Mex.) p.238-47 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

Some of the characteristics which economic fast reactors must approach and which are beginning to be achieved in some advanced concepts are listed and discussed. Specific plutonium reactor systems considered are LAMPRE-II, Paste Fuel Reactor (settled slurry), and Equilibrium Reactor (multi-zoned fluid fuel). (W.D.M.)

16602

FLUID FUELS. B. M. Abraham (Argonne National Lab., Lemont, Ill.) p.248-53 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

Slurries in general are discussed, and their apparent advantages for fast reactor application are reviewed. Experiments on UO_2 -NaK slurries are described. A slurry of NaK-5 vol. % UO_2 has been circulated successfully for long periods of time at 500°C. Erosion and corrosion do not appear to be a problem. The UO_2 is readily resuspendable after it has been permitted to settle either hot or cold. One preliminary experiment with a 15 vol. % slurry, which begins to approach the

concentration of interest for a fast reactor, showed that the suspension was still fluid and it appeared uniform as it was circulated around the loop. (W.D.M.)

16603

ADVANCED FAST REACTOR DEVELOPMENT, PLUTONIUM-URANIUM ALLOY METAL SYSTEMS. R. J. Dunworth and L. R. Kelman (Argonne National Lab., Lemont, Ill.) p.254-6 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

The data obtained so far on the U-Pu and 70 wt. % U-20 wt. % Pu-10 wt. % fission alloys are presented. The binary composition is one of the worst alloys tested, but the ternary fission alloy is about the best alloy examined to date. Properties of the EBR-II fuel alloy reported on include casting, extrusion, rolling, hardness, thermal expansion, thermal conductivity, and corrosion resistance. (W.D.M.)

16604

FUEL ELEMENT CONCEPTS. C. M. Ladd (Atomic Power Development Associates, Inc., Detroit) p.257-62 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

The fuel element concepts that have evolved from the programs at ANL, APDA, and KAPL are: (1) variations of the shell and tube radiator design using a sodium thermal bond; (2) the sodium-bonded flat plate radiator design; (3) the sodium-bonded pin element design used by ANL; (4) the metallurgical bonded plate subassembly proposed by BMI for the APDA Core II program; and (5) the gas-bonded pin element design studied by KAPL. The advanced fuel element designs developed by these three projects are discussed. (W.D.M.)

16605

FUEL AND BLANKET PROCESS DEVELOPMENT.

L. Burris, Jr. (Argonne National Lab., Lemont, Ill.) p.263-71 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

With the emphasis on the importance of the over-all fuel cycle in fast reactors, attention is now being focused on one of the steps within the fuel cycle—fuel reprocessing. Major types of fuel processing are discussed, with emphasis being given to the more recent development of pyrometallurgical processes which have many potential advantages in processing fast reactor fuels. Types discussed include solvent extraction, fluoride volatility processes, pyrometallurgical process development, fractional distillation or sublimation, fractional crystallization, liquid-liquid extraction, cyclic oxidation and reduction, and electrochemical. (W.D.M.)

16606

COUPLED FAST-THERMAL REACTOR. R. Avery (Argonne National Lab., Lemont, Ill.) p.272-5 of PROCEEDINGS OF THE 1957 FAST REACTOR INFORMATION MEETING HELD AT CHICAGO, ILL., NOVEMBER 20-21, 1957.

The primary interest in coupling a thermal reactor to a fast reactor is to obtain a system combining high breeding and long neutron lifetime. It is hoped that this can be achieved without sacrificing too much breeding gain. A conceptual design of a coupled system operating as a power breeder is presented. The design consists of a fast core surrounded by an inner blanket of natural uranium. Outside this inner blanket is a moderator region surrounded by a depleted uranium blanket.

Below and above the fast core is a normal blanket containing no moderator. (W.D.M.)

16641 PRDC-TR-20

Power Reactor Development Co., Detroit.

MONTHLY TECHNICAL REPORT [FOR] FEBRUARY 1959. 11p. Contract AT(11-1)-476. \$3.30(ph), \$2.40(mf) OTS.

The status of the design and fabrication of the major components being procured by APDA for the Reactor Components Test is shown. Work is summarized on various projects including the CP-5 Irradiation Program, fuel irradiations, metallographic examination, corrosion studies, oscillator and control rods, instrumentation, and test operations. (For preceding period see PRDC-TR-19.) (W.D.M.)

16998 RDB(C)/TN-74

Gt. Brit. Culcheth Labs., Culcheth, Lances, England. THE PRECISION CASTING OF URANIUM. A REVIEW OF PROGRESS TO FEBRUARY 12, 1954. H. N. Jones. 19p.

A survey was made of the possibilities of precision casting enriched uranium shapes for the fast reactor. Theoretically, the refractories most able to withstand molten uranium are stabilized zirconia, beryllia, and thoria. Simple melting experiments show, however, that with zircon, fused magnesia, and fused alumina, the rate of attack is very small, and may be negligible. A furnace was constructed, in which precision casting of uranium can be carried out. It is pointed out that precision casting does not eliminate machining operations, and that a large proportion of circulating scrap will be inevitable. The fuel element shapes now proposed are unsuitable for the precision casting technique, and it is therefore recommended that the present work be terminated. (auth)

17004 TID-8505

Civilian Reactor Fuel Element Review Group, AEC. REPORT ON CIVILIAN REACTOR FUEL ELEMENTS. June 1959. 191p. \$1.75(OTS).

The development of fuel elements for pressurized and boiling water reactors, organic-cooled reactors, sodium-cooled thermal reactors, fast breeder reactors, and gas-cooled reactors is discussed. The development of metallic, ceramic, and cladding materials for reactor fuel elements is reported. Fabrication, chemical processing, irradiation testing, and fuel cycle economics for reactor fuel elements are discussed. (W.L.H.)

17346

A COMPARISON OF EXPERIMENTAL FAST AND THERMAL FLUXES WITH ANALYTICAL PREDICTIONS THROUGH THE RADIAL SHIELD OF EBR-1. Larry M. Rentschler (Argonne National Lab., Lemont, Ill.). Nuclear Sci. and Eng. 2, No. 1, Suppl., 40-1 (1959) June.

An investigation performed to appraise the accuracy of the thermal neutron fluxes predicted by the use of removal theory is described. Analytical results are compared with experimental data measured previously in the radial shield of EBR-1. The thermal neutron flux distribution was calculated by two approaches. The first used removal cross sections to predict the fast flux which in turn determined the source term for the thermal diffusion equation. Second, an epithermal energy group was inserted which had as its source the fast neutron flux multiplied by the removal cross section. This epithermal flux described the thermal neutron source. (W.D.M.)

17404

FAST REACTOR SAFETY PROGRAM—A STATUS REPORT. D. Okrent (Argonne National Lab., Lemont, Ill.). Nuclear Sci. and Eng. 2, No. 1, Suppl., 108-9(1959) June.

The past several years have seen major advances made in the field of fast reactor safety, as well as a considerable change of emphasis in the experimental and theoretical work underway. The major items under discussion at a 1955 APDA-sponsored meeting on fast reactor safety and control are listed, and the present status of each of the problems is discussed. (W.D.M.)

17405

KINETIC STUDIES ON EBR-1—A PROGRESS REPORT. J. K. Long, R. R. Smith, and F. W. Thalgott (Argonne National Lab., Lemont, Ill.). Nuclear Sci. and Eng. 2, No. 1, Suppl., 109-10(1959) June.

One of the main features incorporated into the design of the Mark III loading of EBR-I concerns a mechanism for varying the clearance between neighboring fuel and blanket rods. The basic fuel subassembly is described, and experiments which are made possible by the unique design are discussed. (W.D.M.)

17409

AX-1, A COMPUTING PROGRAM FOR COUPLED NEUTRONICS-HYDRODYNAMICS CALCULATIONS ON THE IBM-704. J. M. Cook, D. Okrent, and D. Satkus (Argonne National Lab., Lemont, Ill.) and R. B. Lazarus and M. B. Wells (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 2, No. 1, Suppl., 113-15(1959) June.

In connection with studies in the safety of fast reactors it has been necessary to calculate the energy yield and explosive force of a variety of hypothetical nuclear accidents. A numerical solution for the coupled neutronics-hydrodynamics equations was developed which calculates the variation in time and space of the specific energy, temperature, pressure, density, and velocity, given a spherically symmetric system which is initially superprompt critical, or whose initially specified velocities will lead to a super critical system later in time. As a function of time it computes the reactivity in the form of alpha (the inverse period), the power, the total energy, and the position of the boundaries of the various shells into which the system has been divided. (W.D.M.)

17410

A PARAMETER STUDY USING AX-1. V. Z. Jankus (Argonne National Lab., Lemont, Ill.). Nuclear Sci. and Eng. 2, No. 1, Suppl., 115(1959) June.

The yield of an excursion in a fast reactor depends upon a number of parameters: reactivity inserted, prompt lifetime, the flux shape, equations of state in the core and blanket, etc. For a step input of excess reactivity, k_{max} , and prompt lifetime, l , the yield is a function of k_{max}^3/l , and the functional dependence is determined from the equation of state in the core and the flux shape in the reactor. Several sets of numerical computation using Ax-1 were performed in order to compare the results with predictions of Bethe and Tait. (W.D.M.)

17524

NEUTRON-ENERGY SPECTRUM STUDIES IN UNMODERATED ASSEMBLIES. George Fischer (Argonne National Lab., Lemont, Ill.). Nuclear Sci. and Eng. 2, No. 1, Suppl., 244-6(1959) June.

A procedure for flux determination is described which was applied to a study of the neutron spectrum in a dilute fast core of the ZPR-V and of the equilibrium neutron spectrum of natural uranium in a column of the LASL HYDRO Reactor. Atomic fission ratios

determined by use of the four uranium isotopes U^{234} , U^{235} , U^{236} , and U^{238} quantitatively electrodeposited onto electrodes of small fission chambers provided the means of determining a four-group neutron-energy spectrum analysis for each assembly (W.D.M.)

17566 PRDC-TR-21

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] MARCH 1959.
27p. Contract AT(11-1)-476. \$4.80 (ph), \$2.70 (mf) OTS.

The status of the design and fabrication of the major components being procured by APDA for the Reactor Components Test is shown. Difficulty was experienced during the final heat treatment of the U-2.75 wt. % Mo alloy blanket specimens. The burnup and operating temperatures of the irradiated fuel specimens are reported. Further studies were made on the design of the graphite shielding within the primary shield tank. Environments monitoring radioactivity tables are presented. (For preceding period see PRDC-TR-20.) (W.L.H.)

17971 CF-59-2-78

Oak Ridge National Lab., Tenn.
ANALYSIS OF NEUTRON PULSES IN A GODIVA-TYPE REACTOR. C. W. Nestor and Melvin Tobias. Feb. 25, 1959. 16p. Contract [W-7405-eng-26]. \$3.30 (ph), \$2.40 (mf) OTS.

Some calculations have been made to estimate the characteristics of a neutron-burst type fast reactor similar to Godiva but made up of relatively small component parts—the so-called "layered assembly." One spherical and three cylindrical assemblies have been considered. Critical masses, assuming 5% voids, range from 58 to 65 kg of 93.4% enriched U^{235} . For a reactivity addition of 0.33 dollars above prompt critical, bursts between 2×10^{17} and 6.7×10^{17} fissions were computed with accompanying temperature rises varying from 514 to 1600°C. The burst width at half-maximum was about 12 microseconds. To obtain an idea of the possibilities of stress reduction which might be achieved by layering, an assembly made of small rings was considered. While the critical masses obtained here are believed to be fairly accurate, the predictions concerning mechanical energy generated, total fissions, and burst width may be subject to sizeable error due to the many simplifications required to allow hand computations. Nevertheless, considerable improvement in safety and burst-size is indicated by the use of a "layered assembly" instead of an assembly composed of relatively thick parts. (auth)

18140

WELDING AND THE NUCLEAR POWER PROGRAMME.
Atom No. 32, 22-35(1959) June.

The welding processes of metals, both ferrous and non-ferrous, are discussed in relation to the Calder Hall type reactor, the advanced gas-cooled reactor, and the fast fission breeder reactor. Various inspection methods are given. (C.J.G.)

18222 CWR-900-13-4

Curtiss-Wright Corp. Research Div., Quehanna, Penna.

MONTE CARLO CALCULATIONS FOR NEUTRON AND GAMMA PARTICLE INTERACTIONS AND TRANSPORT IN ONE, TWO, AND THREE DIMENSIONAL NUCLEAR ASSEMBLIES—MAGNET IV, PROGRAM FOR ONE AND TWO DIMENSIONAL GEOMETRY, NEUTRON INTERACTIONS, PLAIN SAMPLING. J. F. Logue, W. J. Roberts, Kevin L. Rooney, and J. A. DeVries. May 1959. 179p.

MAGNET IV is designed to attack two basic problems

in fast reactor analysis. First, fast reactor criticality, neutron lifetime, isotopic absorption ratios, inelastic scattering by isotopes, leakage spectra, etc., in many region spherical and two dimensional cylindrical geometries. Second, for arbitrary source type, transmission through spherical and cylindrical annuli and absorption and energy loss in such annuli are determined. (W.D.M.)

18725 BMI-APDA-636

Battelle Memorial Inst., Columbus, Ohio.
A STUDY OF CORE FUEL SYSTEMS FOR A FAST BREEDER POWER REACTOR. Sherwood L. Fawcett, ed. Nov. 5, 1957. 152p. For Atomic Power Development Associates, Inc.

The first phase of a program aimed toward the development of materials and a core-subassembly design for the second core of the Fermi Reactor is outlined. The ground rules established by APDA for the study were based upon the performance requirements of the reactor plant with some modification to permit higher power generation and upon a value of Pu produced of about \$44 per g. Considerations of various factors led to the selection of the dispersion or cermet fuel as having the most potential for the Core II application. The fuel selected was a 30 vol. % dispersion of UO_2 in U-10 wt. % Mo or gamma-phase-type alloy, zirconium clad, in a flat-plate-subassembly design. The plant economics for subassembly designs of the fuel systems were evaluated at a core power level of 616 Mw. (W.D.M.)

18726 BMI-APDA-641

Battelle Memorial Inst., Columbus, Ohio.
A STUDY OF CORE FUEL SYSTEMS FOR A FAST BREEDER POWER REACTOR—ADDENDUM I. Charles K. Smith, David A. Dingee, and Joel W. Chastain. May 2, 1958. 20p. For Atomic Power Development Associates, Inc.

Report BMI-APDA-636 contains information on a study of core fuel systems for the Fermi Fast Breeder Reactor. In view of certain changes in philosophy and design, additional studies were performed which are based on a revised set of ground rules. The revised ground rules are: reactor power output shall be 830 Mwh with sodium outlet at 1000°F; plutonium revenue shall be calculated at the value of \$30 per g; and blanket cycle time shall be based on 0.5 at. % burnup or 5 yr of residence time. Two new materials uranium carbide and uranium carbide dispersed in U-10 wt. % Mo alloy, were considered in addition to the materials previously considered. (W.D.M.)

18727 BMI-APDA-643

Battelle Memorial Inst., Columbus, Ohio.
A STUDY OF CORE FUEL SYSTEMS FOR A FAST BREEDER POWER REACTOR—ADDENDUM II. Ronald F. Dickerson, ed. July 7, 1958. 76p. For Atomic Power Development Associates, Inc.

Because none of the systems evaluated in BMI-APDA-641 showed any economic promise within the revised ground rules, planning was initiated on a Fermi II reactor as opposed to the modified Fermi I dictated by the original ground rules. A brief summary of the major ground rule changes is presented. The basic method of evaluation and the criteria for choosing a design are the same as previously discussed in BMI-APDA-636. The five basic fuel systems chosen for economic evaluation in this phase of the program were: U-10 wt. % Mo alloy; UO_2 ceramic; UC ceramic; UC dispersed in U-20 wt. % Mo alloy; and UC dispersed in Inconel X. (W.D.M.)

19215 ANL-5837

Argonne National Lab., Lemont, Ill.
 QUARTERLY REPORT [FOR] OCTOBER, NOVEMBER,
 AND DECEMBER 1957. 71p. Contract W-31-109-eng-
 38. \$2.00(OTS).

Advanced Water Reactor Program. Investigations have been started on possible uranium-ceramic fuels with magnesia, alumina, zirconia, magnesia-zirconia and calcia-zirconia. Mixtures of 45 magnesia-45 zirconia and 9 magnesia-81 zirconia by weight appeared to give good preliminary promise. Work on ThO_2 -10 wt.% UO_2 has centered about improvement in density of pellets and elimination of cracking and open porosity. Efforts to increase the percentage of uranium in thorium-uranium have met with considerable success up to 70 wt.% uranium in air-firing tests. Some attempts have been made to prepare UC by reaction of UO_2 with carbon, but so far an incomplete reaction has resulted. As to metal fuels, a sample of U-3 wt.% Ti-1.5 wt.% Nb has shown good corrosion resistance in 245°C water, slightly better than the EBWR fuel alloy. **Fast Power Breeder Program.** An improvement of the average yield of EBR-II-type fuel pin injection castings has been attained, reaching over 90% as compared with yields of 50 to 60% in the recent past. The uranium-plutonium-fissium alloys proposed for the fuel of EBR-II are being given further examination. The thermal conductivity of U-20 wt.% Pu-10 wt.% Fe alloy was determined at 0.053 (cal/sec-cm-°C) at 600°C and thermal expansion at about $16.7 \cdot 10^{-6}$ per °C in the range 20 to 510°C. **Reactor Systems Research and Development.** A calibration test of the ultrasonic inspection technique was made on known defect Croloy tubing to check its effectiveness in the examination of fuel jacketing material. Development is continuing on a beta ray method for measuring clad thickness and on the scintillation counting method of measuring uranium content of fuel plates. An improved design of probe coil for eddy current measurement of cladding thickness has made possible the use of pulse signals instead of the sinusoidal type. Dynamic tests in a new loop are being employed to check the hypothesis that cathodic ions from the loop construction material have been responsible for unusual effects of velocity in the corrosion of aluminum. Corrosion inhibition of magnesium by fluorides is being studied. Specimens of graphite-1 wt.% U_3O_8 fired at 950°C have been made for density and porosity measurements. The bulk density was determined at 1.53 g/cc for a sample with porosity of 21.2 vol.%. **Argonne Low Power Reactor.** An ultrasonic evaluation of the bond quality of 197 fuel elements for ALPR showed only 8 free of defects. A common defect was edge bond separation. **Production, Treatment, and Properties of Materials.** Dilatation curves have been obtained for single crystals of uranium in two crystallographic directions, [001] and [010]. Compressive tests have been made at 600°C on an uranium single crystal. An analysis made from x-ray back-reflection diagrams shows the slip planes to be {021} and the slip direction to be quite close to the $[\bar{1}12]$. Since the {021} planes have denser packing directions, the reason for the $[\bar{1}12]$ is not clear. The densest direction [100], being perpendicular to the direction of compression, was inoperative in this test. Results of the investigation of recrystallization of heavily cold rolled alpha uranium sheet have shown that three recrystallized preferred orientations can be obtained, depending on the annealing temperature, whether it is (1) below 360°C, (2) 400°-600°C, or (3) near 650°C. The beta-gamma transformation in high purity uranium has been determined at $774.8 \pm 1.6^\circ\text{C}$. A higher than expected transformation temperature for an uranium-carbon alloy containing 325 ppm C has been attributed to the very low rate of precipitation of small amounts of

UC from gamma uranium. Phase diagram work continues on uranium systems with individual fissium elements: Ruthenium, rhodium, ruthenium-molybdenum, and others. Density, thermal expansion and thermal arrest data of 3 to 10 wt.% fissium in uranium have been obtained, with densities ranging from 18.4 down to 17.5 grams/cc, and thermal expansions of the order of 14 microinches per inch per °C. Neutron and x-ray diffraction intensities for UN have been measured and compared with calculated values, which confirm the NaCl-type structure for this material. The fused salt electrolysis cell for the production of high purity plutonium has been completed and a trial run using uranium has been made. A study of a Zr-24 wt.% Pu alloy has shown the existence of a two-phase structure, one of the phases being alpha zirconium. Corrosion work on the titanium-zirconium alloy system is directed at present on explaining the differences in the behavior of these alloys in water and in dry oxygen. The electrical resistance of corrosion films of alloys with increasing ratios of titanium to zirconium has shown a continually decreasing resistivity both in oxygen at 500°C and in water at 350°C. In the determination of solution potentials of aluminum at 100°C, effort has continued on obtaining reproducibility of potential as a function of polarizing current. **Alloy Theory and the Nature of Solids.** The first phase of the study of the Ti_2Ni -type phase in zirconium and hafnium has been completed. (For preceding period see ANL-5797.) (auth)

19740 TID-3534

Technical Information Service Extension, AEC.
 FUEL BREEDING. A Literature Search. James M.
 Jacobs, comp. Aug. 1959. 72p. \$2.00(OTS).

A compilation is presented of selected references to data on breeding theory, breeder element and blanket design and fabrication, breeder reactors, and breeder reactor fuel elements. There are 696 references to unclassified reports. (T.R.H.)

19748

NONLINEAR TREATMENT OF LARGE PERTURBATION IN POWER REACTOR STABILITY. H. A. Sandmeler (Argonne National Lab., Lemont, Ill.). *Nuclear Sci. and Eng.* 6, 85-92(1959) Aug.

For large perturbations the usual assumption in power reactor stability that the flux variations δn are small in comparison with the steady-state flux level n_0 is not valid. Due to the product term of excess reactivity and neutron flux, harmonics are generated in the flux. The stability under such conditions can be discussed by extracting the fundamental flux component and obtaining a quasi frequency response as the ratio of fundamental incremental flux component to excess reactivity as a function of frequency and amplitude of perturbation. This approach is applied to two Argonne power reactors, the Experimental Boiling Water Reactor and the Experimental Breeder Reactor, where experimental frequency responses have been obtained and a peak has been observed. It is found that for both reactors the stability decreases as the amplitude of perturbation increases. (auth)

19983 DP-393

Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.
 REPROCESSING OF POWER REACTOR FUELS. Quarterly Progress Report No. 6 [for] January 1 to April 1, 1959. Edward S. Ochipinti, comp. June 1959. 11p. Contract AT(07-2)-1. \$0.50(OTS).

Welds of 309SCb corrode at a rate much faster than that of the wrought metal in boiling HF-HNO₃ systems, although the weld and wrought metal corrode at about equal rates in HF alone. When a 309SCb dissolver is used for processing Detroit-Edison core elements, a satisfactory alternative to dissolving in HNO₃ containing a low concentration of HF is to remove the cladding of zirconium with dilute HF, and then to dissolve the core alloy (10% Mo-90% U) by addition of HNO₃ containing a small amount of Al(NO₃)₃ to complex free fluoride. (For preceding period see DP-338.) (auth)

20138 ANL-5973

Argonne National Lab., Lemont, Ill
TESTING OF POTENTIAL BEARING MATERIALS IN AN ARGON ATMOSPHERE OF LOW HUMIDITY. N. J. Carson, Jr. and W. H. Morris. Aug. 1959. 27p. Contract W-31-109-eng-38 \$0.75(OTS)

Bearing materials for use in the Metallurgy Division Sector of the EBR-II Fuel Cycle Plant must operate without lubrication in an argon atmosphere of low humidity (dew-point of -50°C or less) and high gamma activity. To select suitable materials, a test program, in which samples of materials known to be radiation resistant were tested for wear resistance in an argon atmosphere containing less than 30 ppm water by volume, was initiated. The criterion by which wear resistance was judged is loss in sample weight over a 24-hour abrasion period during which the samples rub along the periphery of a 2½-in. OD wear ring made of SAE 1045 steel. The majority of the samples were tested using a load of 2755 gm and a rubbing velocity of 109 ft per sec. Contact between the rectangular samples and the wear ring was initially linear. Of the materials tested, Reactor Grade Graphite and some graphite derivatives showed the best bearing qualities. A group of steel samples coated with impregnated plastic also showed desirable properties. While they are not as wear resistant as the best graphitic samples, their superior physical properties give them a wider range of application. (auth)

20712 PRDC-TR-23

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] MAY 1959.
11p. Contract AT(11-1)-476. \$3.30(ph), \$2.40(mf)OTS.

The status of various research and development projects for the Fermi Fast Breeder Reactor is briefly summarized. (For preceding period see PRDC-TR-22.) (W.L.H.)

20713 PRDC-TR-24

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] JUNE 1959.
9p. Contract AT(11-1)-476. \$1.80(ph), \$1.80(mf)OTS.

The status of various research and development projects for the Fermi Fast Breeder Reactor is briefly summarized. (For preceding period see PRDC-TR-23.) (W.L.H.)

21783 PRDC-TR-22

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] REPORT PERIOD APRIL 1959. 47p. Contract AT(11-1)-476.
\$7.80(ph), \$3.30(mf)OTS.

The status of various research and development projects for the Fermi Fast Breeder Reactor is briefly summarized. (For preceding period see PRDC-TR-21.) (W.L.H.)

22144 ANL-6029

Argonne National Lab., Lemont, Ill.

CHEMICAL ENGINEERING DIVISION SUMMARY REPORT [FOR] APRIL, MAY, JUNE 1959. Sept. 1959. 144p. Contract W-31-109-eng-38. \$2.75(OTS).

Chemical-Metallurgical Processing. A direct-cycle Pyrometallurgical fuel-reprocessing plant is being designed in conjunction with the EBR-II reactor. Studies were continued on processing of melt-refining residues (crucible skull material). Skull oxide reductions in Mg-Cd solutions are in progress. Methods of isolation of Pu from EBR-II blanket material are being examined. Cadmium containing ~2% U and 0.2% Mg is being circulated at 550°C by an electromagnetic pump. Solubilities of Ru, Ce, and La in Cd were measured at temperatures from 325 to 625°C; the solubility of U in Zn was measured at temperatures from 425 to 800°C. The high-Cd covers of the ternary systems Cd-U-Zn and Cd-U-Mg were investigated. Sixteen colorimetric combustions in oxygen were completed with natural MoS₃. Eight successful colorimetric combustions of Mo in F₂ were completed. Fuel Cycle Application of Volatility and Fluidization Techniques. Studies on the rates of fluorination of UO₂, as it would appear in irradiated fuels, are in progress. Equilibrium constants are being obtained at various temperatures for the dissociation reaction of PuF₆ to F₂ and PuF₄. A systematic study of the reaction of Ni with F₂ is in progress as a part of the study of metal reactions with fluorine. A fluid-bed volatility process for the recovery of U from low-U-Zircaloy-2 fuel alloys is under development. A graphite dissolver-hydrofluorinator was given a shake-down test with a dummy fuel element. Feasibility studies on the calcination of simulated Purex process wastes were successful in providing a free-flowing, granular solid with a volume reduction factor of 4.7. Preliminary tests of a one-step, continuous fluid-bed process for the conversion of UF₆ to UO₂ were made in a 1½-in.-diam. Monel column. Reactor Safety. The pyrophoric nature of the reactor metals U, Zr, Th, and Pu are being investigated. The experimental program to determine rates of reaction of molten reactor fuel and cladding metals with water is continuing. A series of runs with U was completed. Reactor Chemistry. The decay characteristics of K⁴² and Te¹³² are being studied. Work has begun on a study of the reactions of superheated steam with metals, such as Zr, Zircalloys, and stainless steel. (For preceding period see ANL-5996.) (W.L.H.)

22213

CALCULATION ON THE DIMENSION CHANGE EFFECT ON THE FAST REACTOR CRITICAL MASS BY MEANS OF THE PERTURBATION THEORY. S. B. Shikhov. Atomnaya Energ. 6, 162-8(1959) Feb. (In Russian)

A method is suggested for calculating critical reactor mass with respect to active zone composition and dimensions. An operating formula is developed by the ordinary first approximation scheme in the theory of perturbations. In order to calculate the formula coefficients it is necessary first to find the neutron spatial-energy spectra for certain fixed active zones with sufficiently thick reflectors. The critical mass in a wide range of variations was predicted on the basis of the obtained formula coefficients. For cases when the dimensions of the active zone exceed the studied range, a numerical calculation of spectra becomes necessary for calculating coefficients. A formula is derived with coefficients estimated for typical fast reactors. The formula was corrected by nine-group numerical calculations for active zone volumes from 200 to 1000 dm³. (tr-auth)

23030 ANL-5547

Argonne National Lab., Lemont, Ill.

SUMMARY REPORT OF THE HAZARDS OF THE INTERNAL EXPONENTIAL EXPERIMENT (ZPR-V). H. H. Hummel, F. H. Martens, and D. Meneghetti. Mar. 1956. Decl. Apr. 28, 1959. 100p. Contract W-31-109-eng-38. \$15.30(ph), \$5.40(mf) OTS.

The Internal Exponential Experiment (ZPR-V) is to be constructed by loading up to 49 of the fuel cans, containing up to 155 kg of U^{235} , of the Fast Exponential Experiment in a 22-in square iron tank, surrounded by an annular thermal region of fully enriched light water lattice 10 to 15 cm thick. Maximum power is about 1 kw. The fast region, thermal annulus, safety evaluation, and hazards to surrounding area are discussed. (W.D.M.)

23034 LA-2310

Los Alamos Scientific Lab., N. Mex.
DOSIMETRY FOR THE GODIVA II CRITICAL ASSEMBLY NEUTRON FLUX AND TISSUE DOSE MEASUREMENTS. J. A. Sayeg, E. R. Ballinger, and P. S. Harris. Sept. 28, 1959. 23p. Contract W-7405-eng-36 \$1.25(OTS).

Estimates of neutron flux and tissue dose as a function of distance were obtained for the Los Alamos Godiva II critical assembly. Three independent types of measurements were performed: (1) neutron flux by means of the threshold detectors Pu^{239} , Np^{237} , U^{238} , and S^{32} , and bare and cadmium-covered Au^{197} foils; (2) neutron tissue dose by means of the Hurst polyethylene-ethylene proportional counter; and (3) neutron and gamma tissue dose by means of beryllium-shelled tissue-equivalent and graphite- CO_2 ionization chambers. The results for tissue dose as obtained by the above methods showed good agreement. (auth)

23035 LA-2332

Los Alamos Scientific Lab., N. Mex.
A PRELIMINARY STUDY OF A FAST REACTOR CORE TEST FACILITY. D. B. Hall, R. P. Hammond, and R. E. Peterson, eds. Aug. 1959. 129p. Contract W-7405-eng-36 \$2.75(OTS).

The construction of a 20 Mw fast reactor facility is proposed for operational testing of a series of experimental cores using plutonium alloy fuel in molten form. The facility will permit tests to be carried out at specific powers of 500 watts/g for studies of fission gas purging from the liquid fuel, container resistance to corrosion and radiation damage, and continuous processing methods. Supporting programs have provided adequate technological data to permit construction of the facility and have shown promise for the general feasibility of the reactor concept. (auth)

23053 TID-7559(Pt.1)(Paper 15)
Argonne National Lab., Lemont, Ill.

MANUFACTURING METHODS FOR THE EXPERIMENTAL BREEDER REACTOR MARK I AND MARK II FUEL LOADINGS. A. B. Shuck. Paper 15 of FUEL ELEMENTS CONFERENCE HELD AT GATLINBURG, TENNESSEE, MAY 14-16, 1958. 22p.

The EBR-II fuel cycle, the reactor core subassembly, and fuel element are described. The proposed methods for manufacturing the U^{235} fueled Mark I loading and the U-Pu fueled Mark II loadings are described in some detail. The initial loadings will be made using a simulated equilibrium fissium alloy. These will be manufactured by non-remote methods but using the same operational sequences as the recycle material. The recycle loadings will be pyrometallurgically processed, refabricated, and reassembled by remote control in a special facility under construction adjacent to the reactor. A brief description of this plant and the remote control process is given. (auth)

23056 TID-7559(Pt.1)(Paper 18)

Argonne National Lab., Lemont, Ill.
IRRADIATION BEHAVIOR OF URANIUM-FISSIUM AND URANIUM-PLUTONIUM FISSIUM FAST REACTOR FUELS. J. H. Kittel and K. F. Smith. Paper 18 of FUEL ELEMENTS CONFERENCE HELD AT GATLINBURG, TENNESSEE, MAY 14-16, 1958. 10p.

The fuel alloys to be used in EBR-II will contain U, Pu, and a group of fission-product elements known as "fissium." Most of the irradiation studies to date have been made on U-fissium alloys, without Pu additions. These alloys are similar in irradiation behavior to U-Mo alloys, in that if sufficient fissium is present, the desirable gamma phase can be retained so that excellent stability can be achieved. Uranium-plutonium-fissium alloys are also highly stable under irradiation, although there has been some indication that the Pu alloys may be subject to a lower swelling temperature. (auth)

23101 APDA-131

Atomic Power Development Associates, Inc., Detroit.
FUEL ENRICHMENT FOR THE ENRICO FERMI REACTOR. Edward Garelis. May 12, 1959. 31p. \$6.30(ph), \$3.00(mf) OTS.

The enrichment of the Enrico Fermi core was determined by calculations based on critical experiments and was established at 25.6% contained U^{235} for a total uranium content of 1734 kg. This enrichment should ensure criticality for a minimum of 91 subassemblies assuming an equilibrium core for a burn-up of one percent. The uncertainty is estimated to be +20 subassemblies, i.e., a range from 91 to 111 subassemblies. The most probable number of subassemblies on initial loading to criticality is 96. (auth)

Volume 14, 1960

1112 NRL-Memo-819

Naval Research Lab., Washington, D. C.
PROGRESS REPORT ON FAST BREEDER REACTOR SAFETY STUDIES. L. A. Beach, A. G. Pieper, and M. P. Young. July 31, 1958. 25p. (COO-241). Contract AT(11-1)-607. OTS.

The design of the mechanical parts of the radiation scanning system to view melting of fuel elements in the TREAT Reactor is nearly completed. Fabrication work on all these parts will be started by early in the next quarter. Problems in fabrication and in particular, the casting of the rotating drum are being studied. Detailed calculations of the signal to background ratio indicated that the drum must be cast with a material approaching the density of lead. Casting the drum with White's Metal (55% bismuth and 45% lead) will provide a signal to background ratio of greater than 85 which is sufficient for a reasonable dynamic range. (For preceding period see NRL-Memo-803.) (auth)

1113 NRL-Memo-975

Naval Research Lab., Washington, D. C.
PROGRESS REPORT ON FAST BREEDER REACTOR SAFETY STUDIES. L. A. Beach, A. G. Pieper, and M. P. Young. Sept. 1959. 59p. Contract AT(11-1)-607. (COO-243). OTS.

A Radiation Scanning Device was constructed to observe meltdown of fast breeder reactor fuel elements in the TREAT reactor by means of self gamma-emission. Measurements with the rotating drum in front of a beam port of the NRL reactor determined that the gamma transmission of the collimating holes averages approximately 40%, and that the space resolution obtainable with the system is better than the desired 0.1 in. The recording and control parts of the system were completed and given a preliminary check-out. The entire system was shipped to the National Reactor Test Site, Idaho, during July for installation into the North access hole of the TREAT reactor. Final check-out of the system will be made early this fall when it will view fuel pins during transient bursts of the TREAT reactor. (For preceding period see NRL-Memo-936.) (auth)

1125

SOVIET FAST REACTOR—BR 5. R. R. Matthews (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.). Nuclear Eng. 4, 359-60(1959) Oct.

The BR 5 has a maximum power of 5 mw(t), is fueled with plutonium oxide stainless-steel clad pins, and cooled by sodium. Control is by a nickel reflector external to the central sodium tube. Shielding consists of a 20 in. thick water tank followed by a cast iron ring 16 in. thick, and the whole enclosed in a concrete cell of 43 in. average thickness. The reactor has an experi-

mental sodium loop which can operate up to 600°C and is cooled independently of the main circuit. The inlet and outlet circuits to the reactor are split into circuits provided with isolating and non-return valves, a sodium to NaK heat-exchange, a mechanical pump, and cold trap. One secondary circuit is air-cooled and the other generates steam, mercury being used as the heat transfer medium in the interspace of the double wall shell-and-tube type heat exchanger. (C.J.G.)

1135 KAPL-1701

Knolls Atomic Power Lab., Schenectady, N. Y.
THE FAST OXIDE BREEDER—A SUMMARY. J. K. Davidson. July 1, 1957. Changed from OFFICIAL USE ONLY Dec. 6, 1957. 25p. Contract W-31-109-Eng-52. OTS.

The current development of a fast breeder fuel cycle based on a core fuel consisting of mixed dioxides of uranium and plutonium is described. A fast reactor system with this oxide mixture as the fuel is potentially capable of producing economical electrical power with relatively small dependence on the market value of the plutonium produced. The potential of the system results from the advantageous properties of the mixed oxide core fuel. The two major advantages of this fuel are: the fuel element should be capable of extremely high burnup since radiation damage is not a problem and void space is available for the accumulation of fission gases; this results in a low processing rate and small reprocessing plant; and the recovery process has fewer steps than the usual metal fuel element processes, and fabrication is relatively simple. As a result of these advantages, the cost of the fuel cycle, which must be less than the cost of coal for a conventional plant, is potentially low. The initial work on this system was an investigation of the reactor physics by multigroup calculations. The emphasis of the more recent work is on the development of a satisfactory fuel element and on the development and demonstration of the essential features of the core reprocessing cycle, which will then enable a firm estimate of the fuel cycle cost to be made. A preliminary estimate places this cost at about 1.6 mills/kw-hr. The reactor system in general, the fuel element, and the core fuel cycle are described. (auth)

2191 AECU-4384

Power Reactor Development Co., Detroit.
FIRST QUARTERLY TECHNICAL REPORT OF POWER REACTOR DEVELOPMENT COMPANY. Mar. 6, 1959. 60p. OTS.

The major technical developments relevant to the subject matter which have taken place with respect to the Enrico Fermi Atomic Power Plant since Aug. 1957 are summarized. The containment building has been erected and the reactor vessel and a considerable amount of equip-

ment have been installed. The ZPR-III critical experiments concerning core composition and geometry and operational and control data have been substantially completed with highly satisfactory results, and the remainder of the experiments are to be completed by the end of 1959. A great variety of data has and is being collected with respect to environmental phenomena. A new technical description of the power plant was prepared (APDA-124) and is included as an appendix. Research areas covered in some detail include stability of fast reactors, temperature and power coefficients, oscillator tests, meltdown, containment, site and environment, and analysis of maximum credible accident. (W.D.M.)

2222 PRDC-TR-25

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] PERIOD JULY 1959. 27p. Contract AT(11-1)-476. OTS.

Progress in core design, materials, nuclear engineering, reactor vessel engineering, mechanical handling, electrical and instrumentation, liquid metal and steam systems, and test operations are summarized. Environmental radioactivity reports for the Enrico Fermi Reactor are included. (For preceding period see PRDC-TR-24.) (W.L.H.)

2223 PRDC-TR-26

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] PERIOD AUGUST 1959. 28p. Contract AT(11-1)-476. OTS.

Progress in core design, materials, nuclear engineering, reactor vessel engineering, mechanical handling, electrical and instrumentation, liquid metal and steam systems, and test operations are summarized. Environmental radioactivity reports for the Enrico Fermi Reactor are included. (For preceding period see PRDC-TR-25.) (W.L.H.)

2239

IMPROVEMENTS IN OR RELATING TO TWO ZONE BREEDER NUCLEAR REACTORS. Ronald George Tongue (to United Kingdom Atomic Energy Authority). British Patent 822,356. Oct. 21, 1959.

A reactor arrangement is described in which a two-zone breeder is characterized by having a blanket liquid at a lower pressure than the core liquid at the same temperature. Thus, if the core is self-supporting with regard to internal pressure, then the blanket vessel may be built to withstand the reduced blanket pressure. This simplifies construction and cost. (T.R.H.)

3140 IGR-209(RD/R)

United Kingdom Atomic Energy Authority. Industrial Group H. Q., Risley, Lancs, England.
A METHOD FOR ESTIMATING FAST-REACTOR ONE-GROUP CROSS-SECTIONS. B. M. Segal. July 30, 1959. 37p. BIS

A method for estimating one-group cross sections with some degree of spectrum-dependence for application to fast reactor surveys is presented. Cross sections are given in the form of curves which represent a condensation of multigroup diffusion calculations. The accuracy of the method for critical mass calculations is discussed and the errors involved are listed. (C.J.G.)

3213

NEUTRON LIFETIME MEASUREMENTS IN THE FAST REACTORS ZEUS AND ZEPHYR. G. Ingram, D. B. McCulloch, and J. E. Sanders (Atomic Energy Research Establishment, Harwell, Berks, Eng.). Reactor Sci. 10, 22-31(1959) July.

Measurements of mean prompt neutron lifetime were made by a perturbation technique, by the use of a reac-

tivity oscillator, and by the Rossi- α method in which the time correlation of fission chamber pulses is studied. The lifetime in Zeus was found to be 7×10^{-8} sec and in the graphite reflected version of Zephyr, 6×10^{-8} sec. (auth)

3235 NP-8052

Atomic Power Development Associates, Inc., Detroit.
SPECIFICATION FOR NON-NUCLEAR INSTRUMENTATION AND CONTROL FOR THE ENRICO FERMI REACTOR PLANT AND ASSOCIATED FULL-SCALE TEST FACILITY. Specification No. 70-4. Revision No. 1. Paul D. Finger and M. R. Lane. Nov. 23, 1956. Revised Apr. 26, 1957. 102p. OTS.

Specifications for non-radioactive tests on non-nuclear instrumentation and control devices for the Fermi reactor plant are presented. Included are requirements for furnishing and delivery, ready for erection, detectors, indicating and recording instruments, control systems, and panels. (J.R.D.)

3236 PRDC-TR-27

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT: REPORT PERIOD-SEPTEMBER 1959. 27p. Contract AT(11-1)-476. OTS.

Progress in nuclear engineering materials, fuel engineering, reactor vessel engineering, mechanical handling, liquid metal and steam system, and test operations is summarized. Environmental radioactivity reports for the Enrico Fermi Reactor are included. (For preceding period see PRDC-TR-26.) (W.L.H.)

3851 LA-2355

Los Alamos Scientific Lab., N. Mex.
DOSE AND FLUX MEASUREMENTS ON GODIVA RADIATION EFFECTS EXPERIMENTS. Payne S. Harris, E. Frank Montoya, and William H. Schweitzer. Sept. 21, 1959. 19p. Contract W-7405-eng-36. OTS.

Flux and dose measurements of fast neutrons, thermal neutrons, and gamma rays under perturbing conditions using Godiva as a source are described and discussed. The limitations of single monitoring devices are noted. It is pointed out that in order to improve the accuracy of flux and dose measurements in radiation effects studies, specific detectors must be used at the position of the experimental object. (auth)

4138

DOUNREAY FAST REACTOR. COMMISSIONING AND START-UP. J. L. Phillips (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.). Nuclear Power 4, No. 44, 94-100(1959) Dec.

A summary is presented on construction, dry, metal, wet, and final approach tests employed in commissioning and start-up of the Dounreay Fast Reactor. (C.J.G.)

4139

DOUNREAY FAST REACTOR. INSTRUMENTATION AND CONTROL. K. R. Sandiford (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.). Nuclear Power 4, No. 44, 100a-2a(1959) Dec.

The instrumentation for flux measurements in the Dounreay Fast Reactor is described. Design features of the coolant circuit and reactor power control system are described. A discussion is presented on special problems such as the use of liquid metal coolant, the relatively high temperatures in which a number of the instruments have to operate, and the necessity of connecting the instrument cables through the containment sphere. (C.J.G.)

4397

THE MELT REFINING OF IRRADIATED URANIUM:

APPLICATION TO EBR-II FAST REACTOR FUEL. I. INTRODUCTION. L. Burris, H. M. Feder, S. Lawroski, W. A. Rodger, and R. C. Vogel (Argonne National Lab., Lemont, Ill.). Nuclear Sci. and Eng. 6, 493-5(1959) Dec.

A process for recycling plutonium and uranium alloy fuels is described; it involves melting and liquating the alloys in ceramic crucibles and casting the partially purified metals. Its particular merit is the rapid turnover of a fuel suitable for a fast reactor. A pilot plant, now under construction, will operate in a closed cycle with the EBR-II, and will permit technical and economic evaluation. (auth)

4499 ANL-6048

Argonne National Lab., Lemont, Ill.

EXAMINATION OF JACKETED URANIUM SLUGS FROM THE LOS ALAMOS FAST REACTOR "CLEMENTINE." S. H. Paine, W. F. Murphy, and F. L. Brown. Oct. 1959. 17p. Contract W-31-109-eng-38. OTS.

Work done during period 1950-1953. Final Report of Metallurgy Division Program 6.1.5.

Examination of clad uranium reflector slugs which failed by rupture and swelling during the first year of operation of Clementine revealed no anisotropic damage to the uranium because of burnup or thermal cycling. The cladding was found to be of high-sulfur-free-machining steel, and it is postulated that thermal stresses caused welding flaws to open enough to permit seepage and corrosive attack upon the uranium cores by the reactor coolant (mercury). (auth)

5262 TID-3529(1st Rev.)

Technical Information Service Extension, AEC

REPROCESSING OF IRRADIATED FISSION REACTOR FUEL AND BREEDING MATERIAL. A Literature Search James M Jacobs, comp Jan 1960. 41p OTS

A listing is presented of selected unclassified reports received by Technical Information Service Extension since TID-3312, Reprocessing of Irradiated Fission Reactor Fuel and Breeding Materials, was prepared. The listing contains 363 references (J R D)

5450

THE CONCEPTUAL DESIGN OF A POWER-LIMITING SYSTEM FOR A SODIUM COOLED ATOMIC POWER PLANT. James H. Talbot, Jr. and Earl H. Lemon (Bendix Aviation Corp., Detroit) and Richard G. Olson (Atomic Power Development Associates, Inc., Detroit). p.9-18 of "Nuclear Electronics. II." (In English)

An analytical study was made of the conceptual design of the power-limiting system of the Enrico Fermi Reactor. A block diagram of the computer system used is given. The philosophy of the power-limiting system is reviewed, and the considerations which led to the choice of the setback and sequential scram responses are presented. The signals, derived from flux levels, that are available for triggering the responses of the equipment are described. (J.S.R.)

5453

NON-LINEAR KINETICS AND STABILITY STUDIES ON ANALOG COMPUTERS. H. A. Sandmeier (Argonne National Lab., Lemont, Ill.). p.39-55 of "Nuclear Electronics. II." (In English)

Analog studies of non-linear fast reactor kinetics were made, and a description and diagram of the simulation is given. The results obtained are discussed. It is shown that the EBR-I Mark II may be stable for small disturbances at a certain power level, but will not be stable for sufficiently large disturbances because of the product term K_{exn} in the kinetic equations. The stability of a linear system is independent of the disturbance but a non-linear system is not.

Self-sustained oscillations can be visualized in a fast reactor operating on a high power level when the bowing of the fuel pins is limited. These conditions are simulated, and the mechanism for self-sustained oscillations is discussed. (auth)

5637 ANL-5918

Argonne National Lab., Lemont, Ill.

IRRADIATION OF EXTRUSION-CLAD URANIUM-2 w/o ZIRCONIUM ALLOY FOR EBR-I, MARK III. J. H. Kittel. Oct. 1959. 18p. Contract W-31-109-eng-38. OTS.

The fuel material specified for the Mark III core of EBR-I was uranium-2 wt.% zirconium alloy coextruded with Zircaloy-2 cladding. From previous work on swaged or rolled uranium-2 wt.% zirconium alloy, it was anticipated that the extruded alloy would be dimensionally unstable under irradiation unless stabilized by suitable heat treatment. In order to determine an effective heat treatment, irradiation studies were made on both clad and unclad extruded uranium-2 wt.% zirconium alloy specimens at irradiation temperature estimated at 200 to 750°C. The irradiation specimens included material with three different heat treatments, selected on the basis of previous studies, and material transiently melted in its cladding. For unclad specimens, it was found that the irradiation temperature strongly influenced the various irradiation growth rates resulting from different heat treatments. Growth rates of the clad specimens were relatively insensitive to either irradiation temperature or prior heat treatment. An exception was the transiently-melted material, which shortened under irradiation. The cladding had only limited ability to restrain the swelling rates of specimens irradiated at the more elevated temperatures. Clad transiently-melted material was found to be most resistant to high-temperature swelling under irradiation. The results of the present study combined with observations in earlier investigations resulted in a recommendation that the reference heat treatment for the core consist of gamma solution at 800°C followed by isothermal transformation at 690°C. (auth)

6053

ELECTRICAL ENGINEERING ASPECTS OF THE ENRICO FERMI ATOMIC POWER PLANT. Robert H. Logue (Power Reactor Development Co., Detroit) Power App. and Systems No. 41, 71-6(1959) Apr

The electrical engineering problems associated with the design of the Fermi Fast Breeder Reactor are discussed. Design considerations include the main and emergency electric system, reactor control and instrumentation, electrical penetrations in the containment building, induction heating, and other equipment (C J G)

6334 TID-7583(p.174-86)

Du Pont de Nemours (E. I.) & Co., Wilmington, Del.

MECHANICAL HANDLING OF CHALK RIVER AND PRDC FUEL. J. W. Langhaar. 13p.

NRU and NRX fuel from the reactors at Chalk River, Ontario, is shipped to the Savannah River Plant in a finned, externally air-cooled cask provided for this purpose. The cask has an internal length of 11 feet to accommodate full-length elements. At SRP the elements are placed in a dis-solvable frame or "bundle," about 25 per bundle, for storage and subsequent handling. Two bundles at a time are shipped from storage to the separations plant in an available 70-ton cask which has been equipped with a suitable internal rack. It is planned to ship at least the first two charges of PRDC fuel to the Savannah River Plant. The blanket element end sections of the fuel assembly will have been removed at the reactor site. Criticality hazards will probably limit handling operations to four fuel assemblies

in any basket or bundle. The shipping cask design has not yet been specified. The possible need for removing the stainless steel housing tube before processing is to be determined by studies of dissolver operation. (auth)

6336 TID-7583(p.195-205)

Du Pont de Nemours (E. I.) & Co., Wilmington, Del.
CHEMICAL PROCESSING OF PRDC CORE FUEL. H. L. Hull. 11p.

It is planned to process the PRDC core fuel elements in an existing stainless steel dissolver at the Savannah River Plant. The zirconium cladding will be dissolved in dilute hydrofluoric acid. The uranium-molybdenum alloy will be dissolved in nitric acid. Stainless steel portions of the fuel assemblies will be retrieved and buried. Chain nuclear fission will be avoided by keeping the uranium at low concentrations in solution and by handling the fuel in thin slabs. (auth)

7139

COMMISSIONING DFR. J. Allen (Dounreay Experimental Reactor Establishment, Caithness, Scotland). Nuclear Eng. 5, 5-9(1960) Jan.

The designed power output from the DFR is 60 Mw(t) using uranium or plutonium as fuel and a sodium-potassium liquid-metal alloy as coolant. Testing of components in the reactor flasks and removal equipment, fuel element charging mechanism, coolant circuits, and critical assembly was carried out over a period of 50 weeks. A stringent testing program was laid down for all equipment used to transfer components in and out of the primary coolant circuit. Prior to the critical assembly 2,100 of the 2,239 channels, 1,872 for breeder elements and 367 for core elements, were charged "in air"; about 150 were charged with the element charge machine after the coolant was charged. The coolant consisted of four different compositions of NaK weighing a total of 119 tons. Enriched fuel elements containing 45.5% U^{235} were gradually inserted in place of the natural uranium elements in the approach to criticality. The elements in the outer edge of the core and in control rods were replaced first, then the interchange proceeded from the center of the core outward. The interchange of elements was begun on October 28, 1959, and criticality was reached on November 14, 1959. (B.O.G.)

8275 APDA-135

Atomic Power Development Associates, Inc., Detroit.
PRELIMINARY REPORT ON SODIUM TEMPERATURE COEFFICIENTS IN LARGE FAST REACTORS. J. B. Nims and P. F. Zweifel. Nov. 1959. 24p. OTS.

Present practice in fast reactor design is to eliminate any known positive temperature coefficients of reactivity. This approach has been feasible for the fast power reactors now being constructed. In recent considerations of advanced fast reactor designs at APDA, some cases were encountered in which the sodium temperature coefficient of reactivity was positive. Because of the serious implications of this result, a study of the sodium temperature coefficient of a design in which the problem appears to be most severe was made. The principle objective of the study was to obtain a better understanding of the problem. Simply stated, reducing the sodium density causes spectral reactivity effects which can, in some situations, be positive and larger in magnitude than the negative leakage effect, thereby causing a positive temperature coefficient of reactivity. In addition, some attention was also given to possible methods of influencing the sign of the sodium coefficient. Although the results indicate that these methods could be successfully applied, there are penalties involved with their use. The results given are believed to be qualitatively correct, but the uncertainties in the cross section

data on which they are based cast some doubt on the quantitative accuracy. (auth)

8297

BREEDING—HOW SOON A NECESSITY? W. K. Ergen (Oak Ridge National Lab., Tenn.) and E. L. Zebroski (General Electric Co., San Jose, Calif.). Nucleonics 18, No. 2, 60-6; 87(1960) Feb.

Two points of view are presented concerning the necessity of breeder reactors coupled with the presently available fossil and fissionable fuel supplies. One sees the need at 40 years and the other at a century, but both concepts are linked with the economic feasibility of the production of nuclear power at costs competitive with present-day prices. Both concepts envision the necessity for breeder reactors before fuel reserves are depleted. (B.O.G.)

8511

ATOMIC FUEL PROCESSING. PART IV. SAVANNAH RIVER DOES IT CHEAPER. Chem. Eng. 67, No. 4, 135-8 (1960) Feb. 22.

Three fuels are to be processed at Savannah River: the Enrico Fermi breeder power reactor (PRDC) and the NRX and NRU reactors at Chalk River. Methods which are used in the processing of the fuels are presented in order of shipment from the reactor to the solvent extraction of plutonium from the uranium solution. The processing of two PRDC cores yield about 100,000 gal of waste, which will be stored until fission products decay sufficiently for concentration to about 35,000 gal and final storage. (B.O.G.)

9195

IMPROVEMENTS IN OR RELATING TO NUCLEAR REACTORS. Oscar Bunemann (to United Kingdom Atomic Energy Authority). British Patent 819,468. Sept. 2, 1959.

A reactor which uses breeder material coated with fissionable material in the fuel elements is described. In one example ThC powder pressed into rods 1-in. diam by 3- to 12-in. length is wrapped with Pu steel weighing 0.5% of the ThC. The wrapped rod is canned in Al and used as a U substitute in a natural-U reactor. After discharge from the reactor the can is dissolved and the Pu removed and processed, while the ThC is put into another fuel element for irradiation to the desired U^{233} production level. Alternatively, the ThC may be replaced by U which can be re-used as seed elements in the reactor. These seed elements are rods of natural U, depleted U, or Th wrapped in Pu sheet and canned in Al. (T.R.H.)

9228

NUCLEAR REACTOR PLANT DATA. VOLUME ONE. POWER REACTORS, 1959. New York, American Society of Mechanical Engineers, 1959. 132p.

A description of various nuclear power reactors including a schematic flow diagram, power and nuclear data, fuel element and blanket data, reactor core and control data, primary and secondary coolant data, and steam system data is presented. (C.J.G.)

9578 ANL-6106

Argonne National Lab., Lemont, Ill.
A STUDY OF TECHNIQUES AND THE DEVELOPMENT OF EQUIPMENT FOR DECANNING EBR-II FUEL ELEMENTS. J. P. Simon. Jan. 1960. 21p. Contract W-31-109-eng-38. OTS.

Remotely operated, semi-automatic machines, designed to effect the mechanical disassembly (i.e., decanning) of spent fuel elements from the Experimental Breeder Reactor-II, were developed. The machines described are laboratory models which successfully decanned simulated fuel elements. They are electrically or pneumatically op-

erated and are capable of decanning three fuel elements per minute without the routine use of manipulators. Each machine is made up of functional units which are sequentially placed to eliminate handling between steps of the decanning operation and which are easily removable for replacement or maintenance. In the first of the two machines described, pneumatically operated devices prepare the fuel element for a rotary shearing operation which cuts the tubing jacket along a helical circumferential path as it is being removed. The alternate machine removes the jacket by means of a series of shearing rolls. Both machines are served by auxiliary mechanisms, such as inspection jigs, fuel and scrap choppers, and transfer magazines. (auth)

10235

IMPROVEMENTS IN OR RELATING TO HEAT TRANSFER SYSTEMS. George Clifford James Dalton and George Edward Lockett (to United Kingdom Atomic Energy Authority). British Patent 826,931. Jan. 27, 1960.

A fast reactor cooling system is described. Coolant tubes pass through a container of the metal fuel which is liquid during reactor operation. The liquid-metal coolant flowing through the shell-and-tube core is satisfactory mechanically and provides a heat transfer comparable with a multiple flat-plate structure. (T.R.H.)

10257 KAPL-1696

Knolls Atomic Power Lab., Schenectady, N. Y.
REPORT OF THE CHEMISTRY AND CHEMICAL ENGINEERING SECTION FOR NOVEMBER, DECEMBER, 1956, JANUARY 1957. Decl. Feb. 24, 1960. 65p. Contract W-31-109-Eng-52. OTS.

A test facility for the continuous electrolytic destruction of HNO_3 wastes is described and its operation reported. The chemical processing of Fast Oxide Breeder mixed oxide fuel is discussed. The nuclear properties of the Cm daughters of Cf were confirmed. Data on the preparation and properties of PuF_6 are reviewed, and studies of the rate of decomposition of UF_6 on stressed Al are reported. The corrosive effects of UO_2 -NaK slurries on stainless steel were investigated. Development of an electrorefining process for U continued and container materials for the process were sought. The data from the completed second phase of the subcooled burnout study are given. (For preceding period see KAPL-1620.) (T.R.H.)

10545 APDA-134

Atomic Power Development Associates, Inc., Detroit.
SODIUM COMPONENTS TEST FACILITY. Nov. 1959. 208p. Contract AT(11-1)-772. OTS.

The design and construction of a test facility that will be capable of subjecting sodium components to both steady state and transient conditions expected in full size nuclear power electric generating stations are presented. This study encompasses design criteria, an estimate of time required to construct, and a survey of existing facilities to be selected by mutual agreement of the commission and APDA which would be suitable to accommodate the desired tests. (W.L.H.)

10873 KAPL-1210

Knolls Atomic Power Lab., Schenectady, N. Y.
REPORT OF THE NUCLEAR PHYSICS SECTION FOR JUNE, JULY, AUGUST 1954. Decl. Feb. 24, 1960. 55p. Contract W-31-109-Eng-52. OTS.

A group of tests was made on a ZnS-lucite scintillator for detecting fast fission neutrons from U^{235} . It is possible that such a detector will allow the presence of larger quantities of fissionable material than in conventional fission chambers because there is no limitation due to α pile-up.

The measurement of η and ν as a function of neutron energy is also indicated. Counting rates at low (20 ev) and high (1000 ev) energies were low, but improvement is anticipated by construction of a larger detector. Analytical work with the 2-stage magnetic analyzer was used to measure isotope abundances in samples as low as 2×10^{-9} g. Single-stage analyzers were used for B^{10} burnup measurements in irradiated B_4C and the $\text{U}^{235}/\text{Pu}^{239}$ ratio of special fission detector plates for the Hanford crystal spectrometer. Several Pu-fueled power breeder reactor designs were analyzed by multigroup theory using a UNIVAC coding. Fuel and fertile materials in the metallic and oxide form were evaluated. Numerical results of critical mass and breeding ratio as a function of fuel and fertile material composition are presented. (For preceding period see KAPL-1155.) (W.D.M.)

10952 ANL-5321(Rev.)

Argonne National Lab., Lemont, Ill.

ON THE APPLICATION OF MULTIGROUP DIFFUSION THEORY TO FAST CRITICAL ASSEMBLIES. David Okrent. Sept. 1954. Decl. Feb. 24, 1960. 34p. Contract W-31-109-eng-38. OTS.

Multigroup diffusion theory predicts too high a critical mass for small, highly enriched, critical assemblies. An attempt to improve the accuracy of diffusion theory predictions for fast reactors is described. Transport cross sections were adjusted on the basis of multigroup comparison of the Lady Godiva Reactor by diffusion theory and by the asymptotic solution to the Boltzmann equation. In general, the predictions made by the adjusted cross sections were in very good agreement with experiment. The procedure described is not recommended as a panacea but indicates that modifications in elastic transport cross sections may make multigroup diffusion theory more applicable to a wide range of fast reactors. (T.R.H.)

11328 TID-5686

Atomic Power Development Associates, Inc., Detroit.
SPECIFICATION FOR FIXED IN-PLANT RADIATION MONITORING FOR THE ENRICO FERMI ATOMIC POWER PLANT. Specification 708. G. C. Burkitt and J. J. Owens. Aug. 26, 1959. 27p. (APDA-Spec-70-8). OTS.

Requirements are given for furnishing, designing, erection, field adjustment, and preliminary operation of the fixed in-plant radiation monitoring systems for the Enrico Fermi Atomic Power Plant. (W.D.M.)

11351

A GASEOUS-FUEL FAST BREEDER REACTOR. Frederick G. Hammit (Univ. of Michigan, Ann Arbor). *Nuclear Power* 5, No. 47, 125-6(1960) Mar.

The feasibility of a gaseous uranium hexafluoride fueled fast breeder reactor using sodium cooling is discussed. The major difficulty appears to be the mechanical complexity involved in a very large number of small-bore tubes for containing the UF_6 , which must be either individually valved-off or connected into a high-pressure header. (C.J.G.)

12330

FAST REACTORS. Leonard J. Koch (Argonne National Lab., Lemont, Ill.) and Hugh C. Paxton (Los Alamos Scientific Lab., Albuquerque, N. Mex.). *Ann. Rev. Nuclear Sci.* 9, 437-72(1959).

A review of the United States, Russia, and United Kingdom fast reactor and fast critical assembly programs is given. Critical size, perturbation, and spectral data obtained from fast critical assemblies are treated. Coolant and heat transfer problems, control and kinetics transient accidents, and fuel cycles of fast reactors are discussed. (C.J.G.)

12356 PRDC-TR-28

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] PERIOD OCTOBER 1959. 23p. Contract AT(11-1)-476. OTS.

Developmental investigations concerning core design, corrosion, fast reactor fuel elements, nuclear engineering, reactor vessel engineering, mechanical handling, reactor components instrumentation, liquid metal and steam systems, and reactor component test operations are described. Data on environmental radioactivity levels at several locations around the Fermi Reactor are included (For preceding period see PRDC-TR-27.) (J.R.D.)

12357 PRDC-TR-29

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] PERIOD NOVEMBER 1959. 28p. Contract AT(11-1)-476. OTS

Developmental investigations concerning core design, nuclear engineering, mechanical handling, liquid metal and steam systems, and reactor component test operations are described. Data on environmental radioactivity levels at several locations around the Fermi Reactor are included (For preceding period see PRDC-TR-28.) (J.R.D.)

12806

THE PRE-OPERATIONAL ENVIRONMENTAL SURVEY FOR THE ENRICO FERMI REACTOR—1958 John V. Nehemias and G. Hoyt Whipple (Univ. of Michigan, Ann Arbor). *Am. Ind. Hyg. Assoc. J.* 21, 144-50(1960) Apr.

Methods and results are summarized from a pre-operational environmental survey for the Enrico Fermi Reactor. The data will provide a basis for comparison with levels of environmental radioactivity observed after the reactor is in operation. (C.H.)

13462

NEUTRON ENERGY SPECTRUM MEASUREMENTS IN UNMODERATED ASSEMBLIES. G. J. Fischer (Argonne National Lab., Lemont, Ill.). *Nuclear Sci. and Eng.* 7, 355-62(1960) Apr.

The neutron energy spectrum at the center of the dilute fast core of the coupled fast-thermal reactor ZPR-V was studied by use of fission chambers having electrodes quantitatively electrodeposited with U^{234} , U^{235} , U^{238} and U^{238} . Atomic fission ratios found with these four isotopes determine a four-group neutron energy spectrum which can readily be measured as a function of position in the core by use of suitable drive units. The same fission chamber procedure was used to study the equilibrium neutron energy spectrum in a natural uranium exponential column at Los Alamos. The results of measurements in these two spectra are shown and compared with theoretical predictions. The ZPR-V results are also compared to an analysis of this spectrum made by use of nuclear emulsions for the range 0.2 to 2.2 Mev. (auth)

13463

DIFFUSION OF INTERMEDIATE ENERGY NEUTRONS INTO A THICK RESONANCE ABSORBER. H. H. Hummel and D. Meneghetti (Argonne National Lab., Lemont, Ill.). *Nuclear Sci. and Eng.* 7, 363-8(1960) Apr.

Diffusion of intermediate energy neutrons from a large moderated region into a thick resonance absorber is calculated with special reference to a coupled fast-thermal reactor. Boundary conditions at the interface between the two regions appropriate for a $P-1$ calculation in the moderator are investigated. The transport approximation in the moderator is found to give good results for a hydrogenous moderator. (auth)

13504 CEA-BIB-5

France. Commissariat à l'Énergie Atomique, Paris.
DESCRIPTION DES PILES ÉTRANGÈRES, SURGENÉRATRICES, A NEUTRONS RAPIDES PROBLÈMES PARTICULIERS LIÉS AU CONTRÔLE. (Description of Foreign Fast-Neutron Breeder Piles Special Problems Connected with Control). J. C. Arlaud. 1959. 51p.

The main characteristics of the fast piles (DFR, EBR-2, and EFFBR) are grouped in comparative tables. The text describes the solutions adopted for each pile to meet the common problems of construction and operation. 34 references. (auth).

13791

CONCEPTUAL DESIGN OF PYROMETALLURGICAL REPROCESSING PLANT. Louis Basel (Atomic Power Development Associates, Inc., Detroit) and Joseph Koslov (Vitro Corp. of America, New York). *Chem. Eng. Progr.* 55, Symposium Ser. No. 27, 97-106(1959).

A pyrometallurgical reprocessing plant for handling fuel from a 500-Mw fast-breeder reactor is described. Also included are capital and operating costs and a discussion of problems requiring further investigation. (auth)

13793

A TECHNICAL AND ECONOMIC ANALYSIS OF THE SEPARATION OF PLUTONIUM AND FISSION PRODUCTS FROM IRRADIATED NUCLEAR REACTOR FUELS. H. A. Ohlgren and J. G. Lewis (American Metal Products, Detroit); M. E. Weech (Phillips Petroleum Co., Idaho Falls, Idaho), and G. M. Wensch (U. S. Atomic Energy Commission, Washington, D. C.). *Chem. Eng. Progr.* 55, Symposium Ser. No. 27, 125-41(1959).

The technical background for the chemical processing of irradiated nuclear reactor fuels is discussed, and aqueous solvent extraction methods for the separation of uranium from plutonium and fission products are briefly described. An example is presented of the application of aqueous solvent extraction methods to the processing of the fuel and blanket elements of a fast-breeder power reactor operating on uranium and plutonium. Estimated capital and operating costs are presented for an aqueous separation of plutonium and fission products from irradiated fuel elements. A possible plant for the chemical processing of the irradiated reactor elements, integrated in location and function with the remaining parts of a nuclear reactor system, is also described. (auth)

14009 NDA-2116-4

Nuclear Development Corp. of America, White Plains, N. Y.
CONCEPTUAL DESIGN AND ECONOMIC STUDY, URANIUM SPONGE FUEL DEVELOPMENT. G. A. Sofer and M. Raber. Jan 31, 1960. 57p. Project III. Contract AT(30-1)-2303. OTS.

The economic feasibility of substituting uranium sponge fuel for the U-10 wt.% Mo and UC fuels being considered for the Hallam Power Reactor and the Fermi Fast Breeder Reactor was evaluated. The effect of the thermal characteristics of sponge fuel on the number and size of the fuel rods required was studied. The nuclear characteristics of these reactors when fueled with sponge elements were calculated. (C.J.G.)

14544

A METHOD OF AVERAGING THE CONSTANTS WITH ALLOWANCE FOR NEUTRON VALUES FOR CALCULATIONS ON A FAST REACTOR. A. I. Novozhilov and S. B. Shikhov. *Atomnaya Energ.* 8, 209-13(1960) Mar. (In Russian)

A method of averaging multigroup constants for one-

group calculations of the critical dimension or critical mass in two-region fast reactors is suggested. The method, used for the solution of a nine-group spatial problem, proved highly accurate and permitted calculations of the critical mass in a two-region reactor without preliminary calculations of the spatial multigroup problem. (R.V.J.)

14555 APAE-62

Alco Products, Inc., Schenectady, N. Y.
CONTROL AND DYNAMICS PERFORMANCE OF A SODIUM COOLED REACTOR POWER SYSTEM. [1960]. 29p. Contract AT(11-1)-666. OTS.

In an attempt to prove the feasibility of liquid sodium as a reactor coolant, a design study of three vital system components was undertaken: the intermediate exchanger, the boiler, and the superheater. Investigation shows that, during transient changes in load, the steam and sodium systems' temperature changes will not significantly exceed the small steady-state temperature changes, provided that the control system maintains the design steady-state ratio between the flow rates and power level during the transient. An analog computer study of the controlled system behavior for normal and casualty control is discussed. (W.D.M.)

14556 ATL-A-102

Advanced Technology Labs. Div. of American-Standard, Mountain View, Calif.

AN EVALUATION OF MERCURY COOLED BREEDER REACTORS. John O. Bradfute, Donald W. Battles, George S. Clark, Robert E. Corridan, Edward T. Gellenbeck, Devereux L. Kavanagh, Donald R. Mash, Fred E. Romie, and Roger H. Whitlock. Oct. 31, 1959. 289p. Contract AT(04-3)-109, Project Agreement No. 4. OTS.

Under the New Reactor Concepts Evaluation Program sponsored by the United States Atomic Energy Commission, Advanced Technology Laboratories (a Division of American Radiator & Standard Sanitary Corporation) has undertaken an investigation of the technical feasibility and economic potential of the use of boiling mercury as a coolant for fast breeder reactors. The investigation was performed between January 1, 1959, and October 31, 1959. This is the final report on that investigation and is submitted in compliance with the terms of the program authorization, Contract Number AT(04-3)-109, Project Agreement Number 4. (auth)

14593

General Nuclear Engineering Corp., Dunedin, Fla.
POWER REACTOR TECHNOLOGY. Technical Progress Review, Vol. 3, No. 2. Walter H. Zinn, ed. 1960. 76p. \$0.55(GPO)(domestic), \$0.70(GPO)(foreign).

General Research and Development. Section one of this volume is devoted to evaluation of large and small power reactors. The second section discusses applications to nuclear ship propulsion. In the third section the cost and supplies of enriched uranium are reported. The fourth section discusses the significance of breeding, fast breeders, and thermal breeders. The fifth section is devoted to reactor physics and discusses heavy-water lattices, measurement of Doppler coefficient, and thermal neutron flux distributions in fuel specimens. Heat transfer and fluid flow are discussed in the sixth section. The seventh section is devoted to shielding and discusses the neutron distribution in the reactor shield, interactions of fast neutrons in Fe, Pb, O₂, and N₂, and gamma-ray attenuation and heating codes. The last section on materials reports on the fabrication of UO₂ fuel elements, thermal conductivity of UO₂ fuel elements, steam corrosion data for UO₂, and oxidation of UO₂ in CO₂. Progress on Specific Reactor Types. The four sections comprising this part of the volume are de-

voted to heavy-water reactors, Sodium Graphite Reactor, molten plutonium reactors, and organic-cooled reactors. (W.L.H.)

14863 ORNL-2695(p.67-85)

Argonne National Lab., Lemont, Ill.
INSTRUMENTS USED IN HIGH-TEMPERATURE SODIUM AT ARGONNE NATIONAL LABORATORY. F. A. Smith. p.67-85 of NUCLEAR PROCESS INSTRUMENTATION AND CONTROLS CONFERENCE HELD IN GATLINBURG, TENNESSEE, MAY 20-22, 1958.

The development of flow meters, thermocouples, level indicators, and pressure transmitters for use in the liquid metal systems of the EBR-1 and EBR-2 is described. (W.L.H.)

15091 CF-53-3-276(Pt.I)(Del.)(p.36-50)

Argonne National Lab., Lemont, Ill.
IRRADIATION DAMAGE TO THE PROPERTIES OF EBR FUEL AND STRUCTURAL MATERIALS. W. F. Murphy, A. C. Klank, J. H. Kittel, and S. H. Paine. p.36-50 of RADIATION DAMAGE CONFERENCE, MARCH 23-24, 1953.

The Experimental Breeder Reactor (EBR) is a liquid metal cooled, fast flux reactor. During early operation, it was found that the reactivity was dropping at a fairly rapid rate. The drop in reactivity was believed to be due to an unprecedented growth of the enriched fuel elements. The EBR core contains normal and enriched U, beta-quenched to promote dimensional stability. Length measurements of the irradiated rods were made and plotted against % burnup. It was found that the growth of normal uranium was small compared with that of the enriched uranium. A series of notched tensile specimens of type 347 stainless steel was irradiated in the reactor to integrated fluxes of 3×10^{18} , 1×10^{19} , 3×10^{19} , and 2.2×10^{20} nvt. (W.L.H.)

15443 APAE-41(Vol.I)

Alco Products, Inc., Schenectady, N. Y.
INTERMEDIATE HEAT EXCHANGER PRELIMINARY DESIGN. Feb. 28, 1959. 235p. Contract AT(11-1)-666. OTS.

A conceptual study of design of sodium-cooled reactor intermediate heat exchangers (IHx) is presented. The purpose of this unit is to transfer heat to a nonradioactive fluid which circulates through the steam generator. Since the unit is inaccessible, trouble free operation is emphasized. A detailed description of the preliminary design and analysis is given. (See also APAE-41 Vols.II and III.) (J.R.D.)

15444 APAE-41(Vol.II)

Alco Products, Inc., Schenectady, N. Y.
STEAM GENERATOR PRELIMINARY DESIGN. Feb. 28, 1959. 286p. Contract AT(11-1)-666. OTS.

A conceptual study on design of sodium-cooled reactor steam generators was conducted. Included is a detailed description of the preliminary design and analysis, based on the use of known materials and existing methods of fabrication. (See also APAE-41 Vols. I and III.) (J.R.D.)

15445 APAE-41(Vol.III)

Alco Products, Inc., Schenectady, N. Y.
INTERMEDIATE HEAT EXCHANGER AND STEAM GENERATOR CONCEPT STUDY. Feb. 28, 1959. 206p. Contract AT(11-1)-666. OTS.

Development of preferred designs in accordance with A.E.C. specifications for sodium-cooled reactor intermediate heat exchanger and boiler are reported. The field of possibilities was narrowed by consideration of all con-

cepts and elimination of designs which do not meet basic requirements. The work performed from the initial concept to selection of the preferred design prior to final rating is covered. (See also APAE-41 Vols I and II) (J R D)

15894 NDA-2140-2

Nuclear Development Corp of America, White Plains, N Y and Carborundum Co, Niagara Falls, N Y
CARBIDE FUEL DEVELOPMENT Phase I Report—Period of May 15 to September 15, 1959 R Bolomey, S Lazerus, J Sapir, G Sofer, H Steinmetz, A Strasser, J Weisman, R Dial, C McMurtry, F Saulino, and K Taylor Oct 15, 1959 107p Project IV Contract AT(30-1)-2303 OTS

A combination of UC and PuC is proposed as a fuel which has the potential for reducing the fuel cycle cost of fast breeder reactors. A $3\frac{1}{2}$ year development program is outlined, the purpose of which is to fabricate the fuel and evaluate the fuel properties having the most significant effect on fuel cycle cost. Analytical evaluation of PuC-UC in existing fast breeder reactors was completed with respect to heat transfer, physics, and cost. It was found that if certain reasonable fuel performance goals can be achieved a fuel cycle cost reduction of a factor of 2 to 3 is possible. Facilities to handle and irradiate plutonium were designed and partially constructed. The facilities will be capable of synthesis and fabrication of carbides by dry powder techniques, chemical analysis, x-ray diffraction, metallography, fuel-clad compatibility studies, hardness testing, thermal cycling. Irradiation capsules were designed to irradiate clad fuel rod sections to 2% burnup at maximum fuel temperatures $>1900^{\circ}\text{F}$. High purity, better than 99 wt %, UC pellets were fabricated by pressing and sintering. Evaluation studies were initiated. The ease of dissolution of unirradiated UC was confirmed. Dissolution studies of UC with simulated fission products were initiated (auth)

16446 KAPL-M-WMC-3

Knolls Atomic Power Lab Schenectady N Y
TRIP REPORT U S U K FAST REACTOR MEETING
ANL APRIL 30 TO MAY 3 1956 W M Cashin and J K Davidson [1956] Dec 1 May 5 1960 15p Contract W-31-109-Eng-52 OTS

Fast reactors are discussed with particular reference to the EBR-I and -II. A brief summary of the KAPL fast oxide breeder program is presented. (C J G)

16458 PRDC-TR-30

Power Reactor Development Co, Detroit
MONTHLY TECHNICAL REPORT [FOR] DECEMBER 1959
38p. Contract AT(11-1)-476 OTS

Research and development activities on the Fermi Fast Breeder Reactor are briefly summarized in terms of core design, materials, nuclear engineering, instrumentation, liquid metal and steam systems, test operations, and environs monitoring. (For preceding period see PRDC-TR-29.) (W D M)

16462 TID-5855

Duquesne Light Co., Shippingport, Penna
RADIATION SURVEY OF REACTOR PLANT CONTAINERS AND COMPONENTS AFTER SHUTDOWN "D" SURVEY. SECTION III SECOND PERFORMANCE CORE 1, SEED 1 Test Results DL-S-184 (T-612076) Second issue, Mar 22, 1960 8p OTS

Tests were performed to determine the amount of radioactive material build-up in the horizontal and vertical legs of the 1AC hairpin loop of the Shippingport Pressurized Water Reactor. The time required for this radiation level

to decrease to a steady state value was determined (C J G)

16468 TID-6032

Atomic Power Development Associates, Inc., Detroit.
SURVEY OF THE RADIATION LEVELS IN THE CONTAINMENT VESSEL OF THE ENRICO FERMI ATOMIC POWER PLANT. PART V. GAMMA RADIATION LEVELS ON THE OPERATING FLOOR OF THE CONTAINMENT BUILDING. a. LEVELS ABOVE THE EQUIPMENT COMPARTMENT. Technical Memorandum No. 16. W. F. Chalton and H. E. Hungerford. [Dec. 22, 1959]. 47p. Contract AT(11-1)-476. OTS.

The results are presented of a survey of calculated gamma-ray levels at many points on the surface of the operating floor of the containment building for the Enrico Fermi reactor. That portion of the floor surveyed lies directly above the equipment compartment. The calculations were made with the aid of an IBM-650 electronic computer. The main source of radioactivity which gives rise to gamma radiation above the floor is the radioactive sodium-24 in the primary coolant system. This system was considered to be completely filled with sodium, and activated to an equilibrium activity of 0.05 curies/cc, which corresponds to infinite reactor operation at 500 megawatts power. No fission product contamination was considered for these calculations. The operating floor is 5 feet thick and of concrete and steel. The results of the survey indicate that above the equipment compartment the surface dose on the operating floor will in no case exceed 0.9 mr/hr at the expected full operating power of 430 megawatts. Included as appendices are derivations and methods of corrections from one set of concrete and steel thicknesses to another (auth)

16576 ANL-6101

Argonne National Lab, Ill
CHEMICAL ENGINEERING DIVISION SUMMARY REPORT [FOR] OCTOBER, NOVEMBER, DECEMBER 1959 Feb 1960 170p Contract W-31-109-eng-38 OTS

Chemical-Metallurgical Processing A direct-cycle pyrometallurgical fuel-processing plant is being constructed in conjunction with EBR-II. The gamma-irradiation testing of the 175-watt white fluorescent mercury vapor lamp was continued to an integrated exposure of 2×10^3 rad. Irradiation tests of Shell APL grease were completed and the estimated useful life of this grease in the Air and Argon Cells is 2 and 3 years, respectively. Tests of the three types of d-c motors used in the operating manipulator of the Argon Cell indicate an expected useful cell operating life of 2 to 3 years. Continued irradiation tests of mineral-insulated cable show no catastrophic breakdown of the electric insulation even after an accumulated gamma dose of 8.4×10^3 rad. The scheme currently under consideration for processing melt-refining residues involves a reduction of skull oxides by a solution of Mg in liquid Cd. Molten salt fluxes had variable effects on the rate of oxide reduction in dilute Mg systems. Work was continued on development of processes for EBR-II blanket materials. A large-scale metal-distillation unit to demonstrate metal distillation at rates up to 100 kg/hour is under construction. Two medium carbon steel thermal-convection loops were built and operated to ascertain the extent of corrosion under adverse thermal conditions. Data were obtained for the solubilities of V, Fe, and Ni in liquid Cd at temperatures from 400 to 650°C . Corrections were made to data on the solubility of Th in liquid Zn. Solubilities of U in Si-free Zn-Mg solutions were found to be significantly higher than when silicon was present. The peritectic temperature in the U-Cd system was found to be $474 \pm 1^{\circ}\text{C}$. The activity coefficient of U in Al at 4.8% U and 686°C is estimated to be 1.25×10^{-3} , based on a distribution coefficient of U between Cd and Al.

and the activity coefficient of U in pure Cd. The free energy of formation of UPb_3 , determined by galvanic cell measurements, varied between -14.8 to -10.0 kcal/mole at temperatures of 400 to 800°C. Oxygen and fluorine bomb calorimetry are being used to obtain heats of combustion and heats of formation for sulfur and fluorine compounds of such elements as Mo, W, B, and Zr. Fuel Cycle Applications of Volatility and Fluidization Techniques. Several schemes involving the volatility of UF_6 were proposed for processing irradiated fuels. These include a direct fluorination process, the Aqueous Dissolution Fluorination (ADF) Process and the Fused Fluoride Volatility Process. The Direct Fluorination Process involves direct fluorination of oxide-matrix fuels with fluorine or other fluorination agents to produce volatile UF_6 and PuF_6 . Additional studies were made in the Fused Salt Fluoride Volatility Process for the recovery of U from U-Zr-matrix fuels. Additional studies of a one-step fluid-bed process for the conversion of UF_6 to UO_2 were made in the newly installed 3-inch Monel reactor. Reactor Safety. The program to clarify the factors governing the pyrophoric characteristics of the metals U, Zr, Th, and Pu was continued with studies of ignition phenomena and the combustion process. Reactor Chemistry. The determination of cross sections of U^{236} with neutrons from the Van de Graaff is reported. A two-step alkaline-permanganate procedure was developed for room-temperature decontamination of stainless steel surfaces. Transportation of decontamination reagents using foaming techniques is being investigated. (For preceding period see ANL-6068.) (W. L. H.)

16705 ERI-2240-6-F

Michigan. Univ., Ann Arbor. Engineering Research Inst.

STUDY OF THE FEASIBILITY OF AQUEOUS RECOVERY OF SPENT FUELS. PART 6. PARTIAL MATERIAL BALANCE FOR AQUEOUS RECOVERY PLANT. J. G. Lewis and H. A. Ohlgren. June 1954. Decl. Apr. 21, 1960. 21p. (HAO-20). OTS.

Work done for Dow Chemical-Detroit Edison and Associates, Atomic Power Development Project at Request of Consumers Power Co. (Jackson County).

A study of the feasibility of the aqueous recovery of spent nuclear fuels is presented. A partial material balance is described for the aqueous recovery plant being studied. The plant will consist of separate processing facilities for core elements and blanket elements but will be interchangeable in the event one facility fails to function properly. Flowsheets of the proposed separations process are included. (J.R.D.)

17578 NAA-SR-Memo-5054

Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.

BIBLIOGRAPHY OF REPORTS ON FAST REACTOR TECHNOLOGY. B. R. Hayward and M. Bloomfield. Mar. 23, 1960. 11p. OTS.

A bibliography is given of reports in the major fields of fast reactor technology. These fields are: fuels and materials, reprocessing, general technology, safety, and physics. 131 references are cited. (B.O.G.)

17590 ANL-6137

Argonne National Lab., Ill.

SOME STEADY-STATE THERMAL CHARACTERISTICS OF A THREE-LOOP REACTOR POWER SYSTEM. T. R. Bump and H. O. Monson. Mar. 1960. 22p. Contract W-31-109-eng-38. OTS.

The three-loop power system which is to be used with Experimental Breeder Reactor No. II (EBR-II) was analyzed to determine the coolant flow rate requirements at various power levels, coolant temperatures at various

power levels, effects of heat exchanger sizes (system optimization), and effects of control errors. An intermediate heat exchanger, preheater, evaporator, and superheater are included in the EBR-II power system. Constant thermal resistances and physical properties, and perfect insulation were assumed in the analysis. Among other things, the study showed that at low power levels, excessive thermal stresses are produced at the cold end of the intermediate heat exchanger unless a high sink temperature is used. It was also found that a short cut may be used to determine approximate system conditions at all power levels, that system optimization requires compromises, and that system flow rate control is possible through high-low measurements of two coolant temperatures in the primary system. (auth)

17622

FIRST TREAT RESULTS—MELTDOWN TESTS OF EBR-2 FUEL. Edmund S. Sowa (Argonne National Lab., Ill.). *Nucleonics* 18, No. 6, 122-4(1960) June.

In order to study the meltdown behavior of EBR-2 fuel elements under power surges, single EBR-2 fuel pins were exposed to transients in the TREAT reactor. In normal operation, the clad-metal interface will operate around 540°C; in the meltdown tests, temperatures varied from 970 to 1030°C. Damage commences in the range 970 to 1015°C by penetration of the cladding close to the base and ejection of U through this penetration. Above 1015°C, failure is more drastic with ejection and desposition of U on the graphite mount. (D.L.C.)

17628

THE FAST RESEARCH REACTOR BR-2. A. I. Leipunskii (Lejpunskij), D. I. Blokhintsev (Blochincev), I. N. Aristarkhov (Aristarchov), I. I. Bondarenko, O. D. Kazachkovskii (Kazačkovskij), M. S. Pinkhasik (Pinchasik), Yu. Ya. Stavisskii (J. J. Smavisskij), E. A. Stumbur, F. I. Ukraintsev (Ukrainev), and L. N. Usachev (Usachev). *Kernenergie* 1, 245-7(1958) Apr. (In German)

A brief description is given of the fast research reactor now in operation and its principal experimental and auxiliary equipment. The reactor is used for physical studies with fast neutrons. The core is made up of Pu rods, the lateral reflector is of enriched U. Heat is removed from the core by Hg and from the V reflector by air. The nominal total power capacity of the reactor is 150 kw, of which ~100 kw is generated in the core. (tr-auth)

17644 KAPL-1536

Knolls Atomic Power Lab., Schenectady, N. Y.

REPORT OF THE CHEMISTRY AND CHEMICAL ENGINEERING SECTION FOR FEBRUARY, MARCH, APRIL 1956. Decl. May 3, 1960. 98p. Contract W-31-109-Eng-52. OTS.

Studies were continued on the following waste treatment by alkaline electrolytic destruction of nitrate, fabrication, dissolution, and solvent extraction of UO_2 - PuO_2 fuels; radiochemical properties of Cm and Pu isotopes; preparation and properties of PuF_6 ; corrosion and particle size measurements of UO_2 -NaK slurries; alumina-water equilibria; and electrorefining and electrowinning of U from oxide. (For preceding period see KAPL-1491.) (C.W.H.)

18106 BMI-1409

Battelle Memorial Inst., Columbus, Ohio.

PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING JANUARY 1960. Russell W. Dayton and Clyde R. Tipton, Jr. Feb. 1, 1960. Decl. Apr. 20, 1960. 88p. Contract W-7405-eng-92. OTS.

Solid-state studies of oxide additions to uranium oxides

are being conducted in correlation with the high-temperature stability and physical properties of these solid solutions. The program to determine the effects of fast neutron irradiation on properties of AISI Type 347 stainless steel has continued. Cladding materials being studied for possible use in the EBR are unalloyed Nb, Nb-Cr, Nb-Zr, and Nb-Ti-Cr. The evaluation of selected Nb alloys for possible service in pressurized-water reactors was continued. The creep strength of Zircaloy-2 at elevated temperature is to be determined and compared for two test conditions in-reactor under a fast neutron flux and out-of-reactor. A study is being made of new approaches to the problem of measuring the oxygen content of sodium. The development of Nb-rich Nb-U alloys as high-temperature fuels is reported. Development of Th-U-base alloys with improved radiation stability and corrosion resistance continued. Results of various studies designed to support experiments of fission-gas release from UO_2 during and after irradiation are reported. Cermets of 80 vol % UO_2 dispersed in stainless steel, Mo, Cr, and Nb are currently being fabricated by pressure-bonding techniques for physical and mechanical testing. An investigation is being made of the mechanism and kinetics of solid-phase bonding of metals under application of heat and pressure. Various methods of preparing UC powder and dense UC bodies by powder-metallurgy techniques are being investigated. Reliable techniques for the production of high-quality cast shapes of UC are being developed. An investigation of the effects of fission and irradiation on the structure and physical properties of UC is in progress. Research was continued on the problem of growing large single crystals of UO_2 by fusion methods. A basic study of the migration of hydrogen in zirconium under the influence of a thermal gradient is being conducted. Work was continued on the determination of Fe in portland cement by means of a complexometric titration with a radiometric end point. The long-term program for determining the creep and stress-rupture properties of Zircaloy-2 sheet material has continued. Investigations on the development of a fuel-element leak detector have continued. Self-welded specimens of the 12 experimental alloys and unwelded specimens of vacuum-metal Hastelloy F were exposed to boiling Sulfex and Niflex solutions for five 24-hr periods. Previous experience with Al-35 wt.% U alloy has shown that improvements in casting and fabricating characteristics can be obtained when suitable ternary additions are used. The effects of impurities present in commercial-grade materials on the ternary eutectic temperature of Al-U-Ni alloy are being investigated. A program is currently being conducted to determine the changes induced in the properties of UC by irradiation in a test reactor. Work continued on the development of Al_2O_3 -clad UO_2 fuel particles. Data are presented on the room-temperature tensile properties of unirradiated and irradiated Ta. Ceramic-type fuels composed of ~20 vol.% UO_2 in BeO, 20 vol.% UO_2 in Al_2O_3 , 20 vol.% UC in graphite, and 20 vol.% UC_2 in graphite are being considered as potential fuels for the MGCR. Reference materials were selected and fabrication techniques are being evaluated for use in preparing fuel plates and absorber plates for the SM-2 reactor. Pellets of BeO-25 vol.% UO_2 are being prepared for loop and capsule exposures. (For preceding period see BMI-1403.) (W.L.H.)

18637 NAA-SR-Memo-5011

Atomics International, Div. of North American Aviation, Inc., Canoga Park, Calif.

A PRELIMINARY NUCLEAR ANALYSIS OF A SODIUM COOLED FAST REACTOR (SCFR) WITH UC FUEL ELEMENTS IN A SKEWED HEXAGONAL ARRAY. H. Jackel. Feb. 24, 1960. 18p. OTS.

Core dimensions and nuclear parameters were calculated for a sodium-cooled fast reactor containing UC fuel elements in a skewed hexagonal array. (C.J.G.)

18649 TID-6033

Atomic Power Development Associates, Inc., Detroit
DETAILED CONCEPTUAL DESIGN OF THE GRAPHITE PRIMARY SHIELD FOR THE ENRICO FERMI FAST BREEDER REACTOR Technical Memorandum No 23
R J Beaudry, W F Chaltrion, and H E Hungerford
Dec 15, 1959. 98p Contract AT(11-1)-476 OTS

The detailed final conceptual design is presented for the graphite portion of the primary shield which surrounds the reactor vessel of the Enrico Fermi Atomic Power Plant. This shield will consist of blocks of plain and borated graphite or semi-graphite material arranged in layers within the primary shield tanks in specified patterns at each elevation. Each configuration was accurately determined by calculation. All objects in the shield area, such as pipes and pipe hangers, wiring, junction boxes, and other obstructions, were taken into consideration in the design. The boron concentration for the borated material is either 5 or 7 wt % boron. In some areas in the thicker portions of the shield, brick filler blocks were specified. Steel gamma shielding was included in the upper regions of the shield (auth)

18671 ANL-6122

Argonne National Lab., Ill.

PROCEEDINGS OF THE CONFERENCE ON THE PHYSICS OF BREEDING, OCTOBER 19-21, 1959. G. Fischer, C. Kelber, D. Meneghetti, P. Persiani, D. Shaftman, and A. B. Smith, eds. 399p. OTS.

Abstracts were prepared on 26 of 28 papers presented at the Conference on the Physics of Breeding held at Argonne National Laboratory on October 19 to 21, 1959. (C.J.G.)

18673 ANL-6122(p.77-92)

Los Alamos Scientific Lab., N. Mex.

A COMPUTATIONAL SURVEY OF IDEALIZED FAST BREEDER REACTORS. William H. Roach. p.77-92 of PROCEEDINGS OF THE CONFERENCE ON THE PHYSICS OF BREEDING, OCTOBER 19-21, 1959.

Multigroup calculations of nuclear parameters of idealized fast breeder reactors are presented. The systems U^{235} -depleted uranium, Pu^{239} -depleted uranium, and U^{233} -thorium were set up as critical radii problems and solved using the S_4 approximation in the SNG code. Critical mass, conversion ratio, and breeding ratio were computed for each system. (C.J.G.)

18674 ANL-6122(p.93-100)

Argonne National Lab., Lemont, Ill.

THE SENSITIVITY OF BREEDING RATIO IN FAST REACTORS TO UNCERTAINTIES IN CROSS SECTIONS David Okrent. p.93-100 of PROCEEDINGS OF THE CONFERENCE ON THE PHYSICS OF BREEDING, OCTOBER 19-21, 1959.

The sensitivity of the breeding ratio in fast reactors to uncertainties in cross sections was investigated. The uncertainties investigated include the effects of various types of structural materials, the effects of fission products, and the effects of different assumptions on inelastic scattering, alpha, and fission product capture properties. (C.J.G.)

18675 ANL-6122(p.101-15)

Los Alamos Scientific Lab., N. Mex.

BREEDING: INTERNAL OR EXTERNAL? Robert M. Kiehn. p.101-15 of PROCEEDINGS OF THE CONFERENCE ON THE PHYSICS OF BREEDING, OCTOBER 19-21, 1959.

Material composition, geometrical shapes, spectrum changes, lattice arrangements and their effects on new fuel production and heat transfer are examined for internal and external breeding systems. The effects of changes in the above parameters on internal breeding in a system con-

sisting of a molten Pu-Co-Ce alloy contained in tantalum and cooled by sodium were studied. (C.J.G.)

18676 ANL-6122(p.116-33)

General Electric Co. Atomic Power Equipment Dept.,
San Jose, Calif.

PARAMETRIC ANALYSIS OF A PuO_2 - UO_2 FUELED FAST REACTOR. Paul Greebler and Peter Aline. p.116-33 of PROCEEDINGS OF THE CONFERENCE ON THE PHYSICS OF BREEDING, OCTOBER 19-21, 1959.

The physics characteristics of a PuO_2 - UO_2 fueled fast reactor were investigated over a range of design parameters. The physics properties investigated include the breeding ratios, doubling time, required plutonium fuel loading, sodium temperature coefficient of reactivity, and the excess operating reactivity which must be held down by control rods as a function of the refueling interval. (C.J.G.)

18677 ANL-6122(p.137-48)

Argonne National Lab., Ill.

BREEDING IN FAST PLUTONIUM METAL SYSTEMS WITH EBR-II-TYPE RECYCLE. Harry H. Hummel. p.137-48 of PROCEEDINGS OF THE CONFERENCE ON THE PHYSICS OF BREEDING, OCTOBER 19-21, 1959.

The effect of using a fuel processing system similar to that employed by the EBR-II reactor for an 800-liter core plutonium-fueled fast reactor is investigated. The effect of fission product poisoning on the breeding properties of such a reactor is discussed. (C.J.G.)

18678 ANL-6122(p.149-62)

Atomic Power Development Associates, Inc., Detroit.

Pu-U AND U^{235} -Th CERMET-TYPE FAST BREEDERS. John B. Nims. p.149-62 of PROCEEDINGS OF THE CONFERENCE ON THE PHYSICS OF BREEDING, OCTOBER 19-21, 1959.

The breeding characteristics of neutron adsorption and economy, critical mass, and breeding ratio were determined for the following cermet systems PuC-U-Mo and U^{235}C -Th. (C.J.G.)

18679 ANL-6122(p.177-85)

Argonne National Lab., Ill.

BREEDING IN COUPLED FAST-THERMAL SYSTEMS. Robert Avery. p.177-85 of PROCEEDINGS OF THE CONFERENCE ON THE PHYSICS OF BREEDING, OCTOBER 19-21, 1959.

Neutron inventory and breeding ratio were calculated for two coupled fast-thermal systems. One system is cooled with sodium and consists basically of a thermal system surrounding the fast core. The other system is cooled with water and D_2O steam. (C.J.G.)

18796 ANL-5668

Argonne National Lab., Ill

CHEMICAL ENGINEERING DIVISION SUMMARY REPORT [FOR] OCTOBER, NOVEMBER, AND DECEMBER 1956. Mar 1957 Decl. Mar. 21, 1960 111p. Contract W-31-109-eng-38 OTS

A study of mass transfer in a continuous-flow mixing chamber, fluoride volatilization separations process, corrosion testing in fused fluoride systems, behavior of Pu during volatilization of UF_6 , preparation of PuF_6 by fluorination of PuF_4 , particle size distribution and U content of JF_4 , stoichiometric behavior of the reaction $\text{IF}_5 + \text{F}_2 + \text{I}_2$, ignition data on stainless steel, were reported. Studies were continued on the analysis of crude green salt from U ore concentrates, fluidized-bed green salt pilot plant, calcination of reactor fuel wastes, reactor chemistry,

chemical-metallurgical separation processes, waste processing operations, and the operation of the gamma-irradiation facility. (W.L.H.)

19008 ANL-5730

Argonne National Lab., Ill

CHEMICAL ENGINEERING DIVISION SUMMARY REPORT [FOR] JANUARY, FEBRUARY, AND MARCH 1957 Decl. Mar 21, 1960 132p Contract W-31-109-eng-38 OTS

Fluoride Volatilization Separations Process Development of a fused fluoride process for dissolution of uranium-zirconium fuel alloys continued. In corrosion tests to find a suitable container material, Ni was found to be susceptible to a sulfur-type attack. Hastelloy B showed promise, and graphite offers excellent chemical resistance but poor mechanical strength. The dissolution rate of Zr in NaF - ZrF_4 as affected by impingement of the HF sparge was studied. Production of UF_6 by fluidized bed fluorination of UF_4 from ore concentrates was studied. The preparation, melting point, vapor pressure, and vapor density of VF_6 are given. Preliminary dissolution and recovery runs in semi-works equipment are discussed. Fluidization Fluidized-bed techniques have been applied to conversion of $\text{UO}_2(\text{NO}_3)_2$ to UF_4 and to calcination of radioactive liquid wastes. Activities of the Green Salt Pilot Plant and shakedown runs of the shielded waste calciner are described. Reactor Chemistry Studies continued on the kinetics and mechanism of oxidation of U, Th, and Zr. Data are given for oxidation of U in oxygen from 125 to 295°C and 20 to 800 mm pressure, and for Zr from 400 to 900°C and 200 mm O_2 pressure. The ratio of capture to fission cross sections for U^{233} and U^{238} in EBR-I have been determined as a function of position. Chemical-Metallurgical Separations Processes Development of pyrometallurgical processing of spent reactor fuels continued. Work is reported on melt refining and casting of U-Pu, iodine volatility problem, the system U-B-Ta, the distribution coefficients for Pu between U-Cr and Mg and U and Mg, extraction of Pu from U by liquid Mg, Ce removal by dross refining, adsorption of volatilized metals on surface active materials, and fractional crystallization of U with Zn. Analytical Research A study of the behavior of radionuclides with HF and HF-methyl ethyl ketone in ascending paper chromatograms is described. (T.R.H.)

19308 ANL-5709

Argonne National Lab., Ill

METALLURGY DIVISION QUARTERLY REPORT [FOR] OCTOBER, NOVEMBER, AND DECEMBER 1956. Dec. 31, 1956. Decl. Feb. 4, 1960. 64p. Contract W-31-109-eng-38 OTS.

Progress is reported on the following studies: the fabrication, testing, and evaluation of the Experimental Boiling Water Reactor fuel elements, fabrication of ThO_2 - UO_2 fuel pellets, fabrication of fuel and blanket rods for the Experimental Breeder Reactor, properties and irradiation of uranium-plutonium-fissium alloys, corrosion of Al alloys in phosphoric acid, aqueous corrosion of Ta-U-Zr alloys at 290°C, effect of radiation on the corrosion behavior of the Si-U system, dimensional stability of U-Zr alloys under irradiation, the effects of radiation on uranium hydride compacts, the recrystallization of cold-rolled U sheet annealed at various temperatures, self-diffusion in U, phase studies and decarburization of C-U systems, preparation of α -uranium single crystals for elastic constants, sintering of ceramic materials, corrosion mechanisms of Zr and Al at elevated temperatures, and radial distribution of neutron density in reactor fuel rods. (W.L.H.)

19314 CF-60-4-118

Oak Ridge National Lab., Tenn.

CLADDING SURVEY FOR THE ENRICO FERMI REACTOR

U-15 Wt.% Mo BASE DISPERSION-TYPE FUEL ELEMENT. M. M. Martin and R. J. Beaver. Apr. 29, 1960. 39p. OTS.

Potential cladding materials for a flat-plate fuel element containing a dispersion of UC or UO_2 in U-15 wt.% Mo alloy were surveyed on the bases of compatibility with the fissile compounds, matrix material, protective cover materials, and liquid sodium as well as the feasibility of fabricating fuel plates by roll cladding. Radiative-capture cross sections, thermodynamic data, eutectic and intermediate compound formation, mechanical properties, and corrosion by 1000°F Na are reported for austenitic stainless steels, chromium, nickel, niobium, molybdenum, tantalum, vanadium, and zirconium. It was recommended that "A" nickel (molybdenum barrier), Zr-3 wt.% Al, Nb-2 wt.% Cr, and Fansteel 82 be selected for investigation. (auth)

19496 APDA-130

Atomic Power Development Associates, Inc., Detroit. IRRADIATION TESTING OF ENRICO FERMI PROTOTYPE FUEL PINS IN THE CP-5, 1957-1959. M. A. Silliman, A. A. Shoudy, W. G. Blessing, and P. R. Huebotter. Apr. 1960. 56p. OTS.

A program was conducted to proof-test zirconium-clad molybdenum-uranium alloy fuel pins and to verify the predicted allowable burnup of this alloy based on previous MTR capsule irradiation tests. Tests of three full-length pins were completed with burnups ranging from 0.3 to 1.0 at.%. These results verified predicted high-temperature allowable burnups and showed that under certain conditions excessive swelling could occur at low temperatures. Test procedures are described, and photographs showing microstructures are included along with various experimental data. (J.R.D.)

19962 ANL-6028

Argonne National Lab., Ill. THERMAL CYCLING OF EBR-I, MARK III FUEL. G. K. Whitham and E. S. Sowa. Apr. 1960. 26p. Contract W-31-109-eng-38. OTS.

A brief description of the EBR-I, Mark III fuel fabrication process is followed by an exposition of the thermal cycling tests on fuel samples. Results of these tests, coupled with observations on the irradiation behavior of the fuel, resulted in the selection of the optimum heat treatment subsequently used in the fabrication of the Mark III core loading. (auth)

19963 ANL-6124

Argonne National Lab., Ill. and Land-Air, Inc., Chicago. THE INTERNAL FEEDBACK OF EBR-I MARK-III. J. C. Carter, D. W. Sparks, and J. H. Tessier. Feb. 1960. 72p. Contract W-31-109-eng-38. OTS.

The reactor is considered to constitute a closed-loop, nonlinear mechanical system with forcing functions resulting from variations in neutron density and the flow of NaK. The significant sources of internal feedback are found to be the variation in volume of the uranium and the variation in the density of NaK. Resistance to the free motion of uranium in response to thermal expansion accounts for the significant nonlinear properties of the system. This resistance results from the physical characteristics of the redundant structure constituting the core, blankets and containing shell. The mechanical system is transformed into a dynamically similar electronic system which is subject to the transformed operating conditions of the reactor. All the equations defining the time-dependent physical phenomena were developed from an analysis of the mechanical system. Only the constants in the nonlinear equations of motion of the materials of the core and blankets were synthesized from low-power operation of the reactor. The re-

lationship between signal and response of the two systems is in good agreement over all conditions of operation. (auth)

20013 TID-8523

Division of Reactor Development, AEC; Argonne National Lab., Ill. and Atomic Power Development Associates, Inc., Detroit.

FAST BREEDER REACTOR PROGRAM. Jan. 1960. 13p. OTS.

The fast breeder power reactor program is discussed relative to capital costs, thermal efficiency, operating and maintenance experience, specific power, material utilization and cycle flexibility, nuclear considerations, burnup, fuel fabrication, and fuel losses. (W.D.M.)

21105

IMPROVEMENTS IN OR RELATING TO SUPPORT STRUCTURES FOR NUCLEAR REACTOR FUEL ELEMENTS.

John Tatlock and John Alexander Forbes Glass (to United Kingdom Atomic Energy Authority). British Patent 836,043. June 1, 1960.

A device was invented for the support of fuel elements in a fast reactor core in such a way that they are closely nested together, while controlling their displacement and providing more room for coolant flow than can be provided by conventional tube-plate support structures. The invention consists of a nest of tubes with long tubes alternating with short tubes, the tubes being engaged by means of slots and hexagonal faces. (D.L.C.)

21144 TID-3556

Office of Technical Information Extension, AEC. SELECTED REACTORS OF THE POWER REACTOR DEMONSTRATION PROGRAM. A Literature Search. James M. Jacobs, comp. June 1960. 35p. OTS.

A bibliography of 314 progress and topical reports is given for ten reactors of the Power Reactor Demonstration Program: Elk River, Enrico Fermi, Florida, Hallam, Parr Shoals, Pathfinder, Piqua, Puerto Rico, Saxton, and Yankee power reactors. The reports are grouped by the reactor to which they pertain. (D.L.C.)

21147 TID-8518(Bk. 1)

Atomic Energy Commission, Washington, D. C. CIVILIAN POWER REACTOR PROGRAM. PART III. BOOK I. STATUS REPORT ON FAST REACTORS AS OF 1959. 1960. 102p. GPO.

A description of fast reactor types is given and the objectives of the AEC program on fast reactors are outlined. General research and development programs completed and in process on physics, fuel and materials, heat transfer, fluid flow, coolant chemistry, reactor safety, and components and systems are discussed. Reactors completed, under design and construction, and under study are described. (W.D.M.)

21559 ANL-6145

Argonne National Lab., Ill. CHEMICAL ENGINEERING DIVISION SUMMARY REPORT FOR JANUARY, FEBRUARY, MARCH 1960. 191p. Contract W-31-109-eng-38. OTS.

Chemical-Metallurgical Processing. A direct-cycle fuel-reprocessing plant using pyrometallurgical procedures is being designed as part of the Experimental Breeder Reactor No. II project. The reduction of uranium oxide was investigated, using pure Mg and solutions of Mg in Zn and Cd. Cadmium solutions of U were shown to be stable in Types 405 and 410 stainless steel containers at temperatures up to 550°C. The liquid metal corrosion loop in which a U-Mg-Cd alloy is being circulated at 550°C has been in

trouble free operation for 3000 hrs. Recovery of Pu from Mg solution by distillation of Mg was demonstrated on 1-g Pu scale. The solubility of Th in liquid Cd was measured over the temperature range from 1.9×10^{-3} per cent at 348°C to 1.8×10^{-2} per cent at 658°C . The solubility of Mn in liquid Cd was found to range from 0.27% at 414°C to 1.43% at 661°C . The solubility of Ni in liquid Cd was measured. The partition of U between liquid Al and liquid Cd was studied as a function of U concentration. The reaction of Al with a liquid Cd solution containing U, Zr, and Ce was studied. The free energy of formation of the U-Pb intermetallic compound UPb_3 was measured between 374 and 846°C by means of a galvanic cell method. Magnetic susceptibility measurements on the intermetallic compound CeCd_{11} were made over a range of temperature from 4 to 295°K . Fuel Cycle Applications of Volatility and Fluidization Techniques. The Direct Fluorination Process is currently aimed toward the processing of the Zircaloy-clad, UO_2 fuel typical of the Dresden Reactor. The direct fluorination of dense UO_2 pellets submerged in an inert fluidized medium was carried out in a 3-inch reactor at 500°C at fluorine concentrations up to 30%. Experimental work has continued to determine the mechanism of the nickel-fluorine reaction. In the development of the ADF process, current emphasis is being directed toward recovery of U from low U-Zircaloy-2 fuel alloys. A series of four dissolutions was carried out semicontinuously in the graphite pilot plant dissolver according to the Fused Salt Fluoride Volatility Process flowsheet. Additional studies of the one-step fluid-bed process for the conversion of UF_6 to UO_2 with steam and hydrogen are reported. Reactor Safety. The oxidation, ignition, and combustion processes of U, Zr, Th, and Pu are being studied in order to provide information leading to an understanding of the reactions. Reactor Chemistry. Data are being obtained on the neutron capture cross sections of U^{236} as a function of neutron energy. The Reactor Decontamination Program is directed principally to boiling water reactors. A stainless steel loop simulating the action of a boiling water reactor was operated with Y^{90} and Cs^{137} tracer activities. A program of research directed toward the selection and evaluation of materials for nuclear superheaters was initiated. (For preceding period see ANL-6101) (W L H)

21900

FUEL AND BREEDER ELEMENTS FOR NUCLEAR REACTORS. Werner Gutmann. Kerntechnik 2, 251-2(1960) July-Aug. (In German)

Considerations on the construction of reactor fuel elements in order to ensure optimum operating safety are described. It was proposed to arrange fuel elements of uranium monocarbide, plutonium monocarbide, or thorium monocarbide molded forms in a casing with a protective coating of carbides or silicides of elements of the IV and V group of the periodic system. (tr-auth)

22611 PRDC-TR-33

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] MARCH 1960.
30p. Contract AT(11-1)-476. OTS

A preliminary fuel bundle using two fuel cartridges of half-length fuel plates was designed for Type B core loading for the Fermi Fast Breeder Reactor. The effect of decay power of fission products on reactor kinetics was investigated. For Type A core, equilibrium core and blanket compositions and loading were determined for the Na-bonded pin. Axial and radial variations in blanket heating were determined. Gas generation rates in the neutron source and in the Zr-5 wt. % Be brazing alloy were determined. Modifications to the shielding of the offset handling mechanism are discussed. Various tests performed on the Reactor Components Test Facility and fuel element endurance

are described. (For preceding period see PRDC-TR-32) (C J.G.)

22628

PRICING BRED REACTOR FUEL. Milton C. Edlund (Babcock and Wilcox Co., Lynchburg, Va.) Nucleonics 18, No. 8, 64-5, 110(1960) Aug.

A computation was made to determine whether the buy-back prices set by the AEC for U^{233} and Pu are realistic. It is stated that a reasonable pricing system is to assign a value to the bred fuel that makes the energy cost of the equilibrium fuel just equal to the energy cost of the initial fuel cycle, before recycling starts. Starting with this definition, it is shown that, relative to the current prices for U^{235} , U^{233} is worth \$18 to \$20/g and Pu \$6 to \$8/g. These figures include a 0.5 mil/kwh penalty for the extra expense of fabricating recycle elements. This difference of a factor of three in the suggested prices for Pu and U^{233} has two sources: (1) in a thermal reactor U^{233} produces more fission neutrons per capture than does Pu, and (2) the uranium replaced by U^{233} in the Th- U^{233} is fully enriched and hence more expensive than the slightly enriched uranium with which Pu competes in the Pu- U^{238} cycle. (B O G)

23224 NUMEC-TM-P-11

Nuclear Materials and Equipment Corp. Apollo, Penna.
DEPLETED VS NATURAL URANIUM IN Pu REACTORS
K H Puechl. May 2 1960. 4p. OTS

The relative merits of depleted and natural U as source materials in Pu-fueled reactors are discussed in terms of fuel costs. (W L H)

23681 APDA-139

Atomic Power Development Associates, Inc., Detroit.
THE DOPPLER EFFECT IN FAST NEUTRON REACTORS (thesis). Richard Benjamin Nicholson. June 1960. 145p. OTS

Submitted to Cornell Univ.

A new analysis of the Doppler effect in fast neutron reactors is made. The effect is evaluated for U^{235} and U^{238} at several neutron energies between 1 and 200 kev. A new method is presented for U^{238} at neutron energies below 30 kev where the previous method is shown to be invalid. The new method is based on the assumption that the individual resonances are isolated and do not overlap. The importance of inelastic scattering is evaluated and the effects of other neglects previously made without complete justification are studied in some detail. The newer data on low energy resonance parameters and high energy cross sections are examined to determine the best choice for the statistical distributions of neutron and fission widths and the distribution of resonance spacing. A crude numerical calculation for the Doppler temperature coefficient of reactivity in the Fermi Fast Breeder Reactor gives about $-2 \times 10^{-6}/^\circ\text{C}$ at 550°C . The results indicate that the Doppler effect from U^{235} can almost always be neglected as small but in some of the larger reactors such as a large fast oxide breeder U^{238} may contribute a negative Doppler effect of the order -10^{-5} which would be of considerable importance, especially with respect to reactor safety. (auth)

23758 PRDC-TR-34

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] APRIL 1960
22p. Contract AT(11-1)-476. OTS

Research and development activities on the Fermi Fast Breeder Reactor are summarized in terms of core design, materials and metallurgy, nuclear engineering, reactor vessel engineering, electrical engineering and instrumentation, liquid metal and steam systems, and test operations. Environmental radioactivity data are also included. (J R D)

24465 ANL-6099

Argonne National Lab., Ill.

METALLURGY DIVISION ANNUAL REPORT FOR 1959.

171p Contract W-31-109-eng-38. OTS.

Engineering Metallurgy. Investigations were made on the following: development programs for Experimental Breeder Reactor No. 2 (EBR-II); fabrication of core and blanket components for Argonne Fast Source Reactor (AFSR); fabrication of spiked fuel rods for Experimental Boiling Water Reactor (EBWR) (Core 1A); fuel for Transient Test Reactor Facility (TREAT); fuel elements for Experimental Breeder Reactor No. 1 (EBR-I) (Mark-IV); canning of plutonium loading for Zero Power Reactor No. 3 (ZPR-III); development of ceramic fuel materials; irradiation evaluations of various experimental fuel and control-rod materials; postirradiation examinations of full-scale reactor fuel elements; development of corrosion-resistant fuel and jacketing materials; nondestructive testing developments; and metallurgical assistance to the Fast Reactor Safety Program. Basic Metallurgy. Investigations were made on the following: preparation of high-purity materials, physical metallurgy of uranium; constitution and properties of uranium and plutonium alloys; alloying properties; x-ray and neutron-diffraction studies; problems in metal physics; corrosion research; irradiation effects; and ceramic-materials research. (W.L.H.)

25004 AERE-M-597

United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England.

USE OF HASSITT PROGRAM FOR MULTIGROUP CALCULATIONS FOR FAST REACTORS. T. P. Moorhead. Dec 1959 9p.

25013 LAMS-2433

Los Alamos Scientific Lab., N. Mex.

A PROPOSED FAST REACTOR INSTALLATION PROJECT FRE-EX Richard J Watts Feb 26, 1960 14p Contract W-7405-eng-36 OTS

A conceptual design for a fast reactor installation is described. This design is based on the argument that the design of highly rated core configurations is so uncertain that some method of minimizing the investment risk of an experimental installation is desirable. The plan is based on the availability of a remote test area such as the Nevada Test Site as a minimum-expense facility (auth)

25538 DP-479

Du Pont de Nemours (E. I.) & Co. Savannah River Lab Aiken, S. C.

REPROCESSING OF POWER REACTOR FUELS. Quarterly Progress Report No. 9 [for] October 1, 1959 to January 1, 1960. Leon H. Meyer, comp. Mar. 1960. 12p. Contract AT(07-2)-1. OTS

Pressure and mechanical vibration are effective in maintaining continuous anodic dissolution in HNO_3 of stainless steel contained in an electrolytically inert anode basket. Zirconium metal can be electrolytically disintegrated in HNO_3 by the same technique, in this case 85% of the zirconium precipitates as ZrO_2 and only 15% remains in solution. When Zr-10% U alloy is electrolytically disintegrated in HNO_3 , 60 to 80% of the zirconium precipitates as ZrO_2 , carrying 10% or more of the uranium. The constituents of a charge of PRDC elements, zirconium and U-Mo alloy, were dissolved in a two-step process with HNO_3 and HF. The rate of dissolution of irradiated PRDC fuel is one to two times that of unirradiated fuel. (For preceding period see DP-439.) (auth)

25873

PLUTONIUM-URANIUM-TITANIUM ALLOYS. (to U. S.

Atomic Energy Commission). British Patent 843,307. Aug. 4, 1960.

Plutonium-uranium-titanium alloys are proposed for use as fast breeder reactor fuel because they have the following advantages: (1) Pu^{239} serves as breeder fuel for fast (epithermal) neutron fluxes; (2) uranium serves to dilute plutonium and thus reduces cooling difficulties without slowing fast neutrons down to a point where they are captured by Pu^{239} . The uranium is converted to plutonium by reactor operation, thus replacing the spent plutonium; and (3) titanium improves the metallurgical properties of plutonium-uranium alloys (fabrication, working, and corrosion resistance) and does not moderate fast neutrons. Such alloys can be made by mixing uranium and plutonium in atomic proportions from 9:1 to 1:1 and adding 15 to 60 (preferably 32 to 42) at. % titanium. The superior properties of these alloys are attributed to the phase having a hexagonal crystal structure and the composition $(\text{U}, \text{Pu})_2\text{Ti}$. (D.L.C.)

25999

TERNARY ALLOYS OF URANIUM, COLUMBIUM, AND ZIRCONIUM. F. G. Foote (to U. S. Atomic Energy Commission) U. S. Patent 2,947,621. Aug. 2, 1960

Ternary alloys of uranium are described which are useful as neutron-reflecting materials in a fast neutron reactor. They are especially resistant to corrosion caused by oxidative processes of gaseous or aqueous origin and comprise uranium as the predominant metal with zirconium and niobium wherein the total content of the minor alloying elements is between 2 and 8% by weight.

26032 LA-2432

Los Alamos Scientific Lab., N. Mex.

REVISED NEUTRON FLUX, SPECTRUM, AND TISSUE DOSE MEASUREMENTS AT THE GODIVA II CRITICAL ASSEMBLY. Joseph A. Sayeg. Apr. 1960 24p. Contract W-7405-eng-36. OTS.

Addendum to LA-2310

Neutron-flux, spectra, and tissue-dose measurements were made at the Godiva II critical assembly by foil activation techniques (C.J.G.)

26470 NYO-2948

Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif.

STUDY OF REMOTE MILITARY POWER APPLICATIONS REPORT NO. 12 EVALUATION AND SELECTION OF APPLICABLE REACTOR CONCEPTS Jan 1960 Revised July 1960 61p Contract AT(30-1)-2441 OTS

An evaluation of the reactor concepts under consideration for remote military power plants is presented. The concepts include water-cooled and -moderated reactors, both direct and indirect cycle, organic-cooled and -moderated reactors, heavy-water-cooled and -moderated reactors, gas-cooled reactors, sodium-cooled graphite-moderated reactors, fast reactors, and fluid-fuel reactors. The limitations and advantages, technological status, economics, and future potential of each reactor are reviewed. From the reviews it is concluded that direct-cycle boiling-water and pressurized-water reactors are most suitable for applications requiring power before 1965 (C.J.G.)

26494

METHOD AND APPARATUS FOR IMPROVING PERFORMANCE OF A FAST REACTOR (to U. S. Atomic Energy Commission) British Patent 843,636. Aug. 4, 1960

A fertile-fissionable material arrangement is given for fast-breeder reactors in which the fissionable material forms a hollow core and the fertile material forms two blankets surrounding the core from both outside and inside.

The advantages of such an arrangement are that all the neutrons are utilized, the neutron flux at the center is flattened out giving better performance, and the operating temperatures in the center are lowered. The fissionable material is a plutonium-uranium alloy, and the fertile material is depleted uranium. Details are given on the reactor and its operation. (D L C)

26495

IMPROVED FUEL-BREEDER ELEMENT FOR A NUCLEAR REACTOR (to U S Atomic Energy Commission) British Patent 847 847 Sept 14 1960

A unique fuel-element design is proposed for breeder reactors in which the fertile and fissionable materials are separated, the fissionable material being in the nonaqueous fluid phase. The fertile material is fabricated as the core of a fuel element with the fissionable material dissolved or suspended in a molten inorganic medium, e.g., bismuth or bismuth-tin for uranium or a fused fluoride mixture for UF_4 surrounding the core and in contact with the container (stainless steel or zirconium). Because the solubility of uranium in the above molten solvents is low, it should be enriched >90% U^{235} or U^{233} . The advantages of this fuel-element design are ease of fabrication, ease of reprocessing for fertile material recovery, high burn-up, resistance to radiation damage or growth, and improved heat transfer

owing to the power level not being limited by the center temperature. The fuel element is particularly suitable for use in sodium-cooled graphite-moderated thermal reactors. (D L C)

26498 ANL-6194

Argonne National Lab., Ill

FAST FUEL TEST REACTOR—FFTR CONCEPTUAL DESIGN STUDY R. Brubaker, H. H. Hummel, A. McCarthy, A. Smaardyk, and J. H. Kittel Aug 1960 101p Contract W-31-109-eng-38 OTS

The Fast Fuel Test Reactor (FFTR) is a nuclear facility for the purpose of irradiating samples of fuels and structural components for use in fast reactors. The core consists of a plate type element in a square configuration. Beryllium metal between the fuel elements is used to obtain a neutron energy spectrum in the hard intermediate region. Cooling of the core and test specimens is accomplished by means of liquid sodium. The design concept was carried through in sufficient degree in the following areas of preliminary concern: number and size of irradiation facilities, sample power requirements, plant layout to evaluate site requirements, plant and nuclear design parameters to evaluate essential equipment requirements, plant-capital-cost estimate, annual-operating-cost estimate, and estimate of construction time schedule. (W D M)

Volume 15, 1961

1013 NP-tr-482

NUMERICAL METHODS OF DESIGNING NUCLEAR REACTORS. G. I. Marchuk. p.279-317, Chap. XII of "Chislennye Metody Rascheta Yadernykh Reaktorov." Translated from a publication of the Publishing House of Atomic Energy, Moscow, 1958. 43p. OTS.

Analysis of the basic and conjugate equations of fast reactors, reduction of these equations to multigroup types, and certain methods for solving kinetic equations are discussed. (W.D.M.)

1063 PRDC-TR-35

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] MAY 1960. 12p.
Contract AT(11-1)-476. OTS.

Research and development activities on the Fermi Fast Breeder Reactor are briefly summarized in terms of core design, nuclear engineering, reactor vessel engineering, mechanical handling, electrical and instrumentation systems, liquid metal and steam systems, and test operations. (For preceding period see PRDC-TR-34.) (W.L.H.)

1064 PRDC-TR-36

Power Reactor Development Co., Detroit.
MONTHLY TECHNICAL REPORT [FOR] JUNE 1960. 13p.
Contract AT(11-1)-476. OTS.

Research and development activities on the Fermi Fast Breeder Reactor are briefly summarized in terms of core design, materials and metallurgy, nuclear engineering, mechanical handling, electrical and instrumentation systems, liquid metal and steam systems, and test operations. (For preceding period see PRDC-TR-35.) (W.L.H.)

1908

PLUTONIUM-URANIUM ALLOY. (to U. S. Atomic Energy Commission). British Patent 849,713. Sept. 28, 1960.

Pu-U alloys suitable for use as fuel in fast breeder reactors and having good fabrication characteristics can be prepared by mixing an alloy comprising Pu and U in the ratios 1:9 to 1:1 and 7 to 40 at. % Fe, Ni, Co, and/or Mn. The latter group of metals does not moderate or capture fast neutrons and improves the working properties of the alloy by keeping the amount of the ζ phase small. From 7 to 40 at. % of these metals will keep the ζ phase small, although 14.3 at. % is the amount that will form the preferred tetragonal crystal structure of the U_2Mn type. Six examples of the preparation of such alloys are given. (D.L.C.)

1940 LMSD-703084

Lockheed Aircraft Corp. Missiles and Space Div., Sunnyvale, Calif.
HIGH-SPECTRUM REACTORS AS POSSIBLE THERM-
IONIC PRIMARY HEAT SOURCES. John E. Cobb.
Sept. 1960. 50p. OTS.

A one-space dimensional, multigroup, multiregion analysis was performed on a series of fast and semifast neutron spectrum refractory reactor types. Diffusion theory was used with transport corrections applied in the description of the neutron-transport process. The boundary conditions and selection of the energy groups used in the analysis gave a very good approximation to criticality for the types of reactors studied. Three primary reactor types were investigated pending final heat-transfer studies to determine the type which can best supply primary power to the thermionic units at the best mass ratio. The three types studied were bare critical, reflected, and internally moderated and externally reflected. (auth)

2296 ANL-6215

Argonne National Lab., Ill.
REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT [FOR] AUGUST 1960. Sept. 16, 1960. 59p. Contract W-31-109-eng-38. OTS.

General research and development on water- and sodium-cooled reactors are reported along with specific developments on EBWR, BORAX-V, EBR-I, and EBR-II. Thermal and fast reactor safety studies are summarized in terms of fuel-coolant chemical reactions, kinetics of oxidation, ignition of reactor metals, and TREAT development. Developments in reactor fuels and materials, reactor components, separations, and advanced reactors are discussed in some detail. (W.D.M.)

2297 ANL-6234

Argonne National Lab., Ill.
REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT [FOR] SEPTEMBER 1960. Oct. 14, 1960. 76p. Contract W-31-109-eng-38. OTS.

2332 GEAP-3347

General Electric Co. Atomic Power Equipment Dept., San Jose, Calif.
FAST OXIDE BREEDER-STRESS CONSIDERATIONS IN FUEL ROD DESIGN. K. M. Horst. Mar. 28, 1960. 34p. Contract AT(04-3)-189. OTS.

Expressions were determined for calculating the optimum fuel-rod geometry which will contain the maximum fission-gas pressure. Consideration was given to stresses from both internal gas pressure and heat transfer through the cladding wall. The allowable gas pressure is shown to reach a maximum at a specific ratio of cladding diameter to wall thickness. Inconel-X, niobium-alloy, and type 347 stainless-steel claddings are discussed. (W.D.M.)

2349

ADVANCED SYSTEMS. A PERSONAL APPRAISAL. Alvin M. Weinberg (Oak Ridge National Lab., Tenn.). Nuclear Eng. 5, 463-5(1960) Oct.

Advanced civilian power reactors are reviewed. The

most important trends noted in advanced reactor development are the trend toward unclad or ceramic-clad solid-fuel-element systems, the renewed interest in high-temperature liquid fuels, and the continued work on fast and slow neutron breeders. Reactors having these characteristics are discussed. (M.C.G.)

2357

BREEDER ECONOMICS. Jack Chernick (Brookhaven National Lab., Upton, N. Y.). *Nuclear Sci. and Eng.* 8, 365 (1960) Oct.

In a study of U. S. uranium reserves, the great differences in reserves of high-grade and low-grade ores made it clear that prices must be included in any discussion of nuclear fuel. Reserves of high-grade uranium are sufficient to develop a large scale commercial power industry, but insufficient to sustain it in the absence of a breeder economy. The depletion of both economically recoverable fossil fuels and low-grade uranium at low fuel utilization is expected after a period of several decades. It is postulated that during this period a breeder economy based on uranium and thorium ores in this price range must be established to prevent an excessive rise in fuel costs. (M.C.G.)

3047 (NMI-4801) COEXTRUSION OF ZIRCALOY-CLAD U-2 w/o Zr ROD FOR THE EBR-I MARK III CORE LOADING. H. F. Sawyer. Nuclear Metals, Inc., Cambridge, Mass. Apr. 30, 1958. 53p. Contract AT(30-1)-1565. Sponsor Agreement No. S-27. OTS.

Technical data are given concerning the fabrication of Zircaloy-clad U-2 wt. % Zr alloy rod from extrusion billets. The development of high-yield extrusion billets for fabricating the enriched core and blanket rod is described. (auth)

3147 (ANL-6010) SUMMARY REPORT ON IRRADIATION OF PROTOTYPE EBR-II FUEL ELEMENTS. J. H. Monaweck and E. S. Sowa. Argonne National Lab., Ill. Sept. 1960. 32p. Contract W-31-109-eng-38. OTS.

The series of irradiations performed in the CP-5, MTR, and ETR on promising prototype EBR-II fuel elements is summarized. Type 304-clad uranium-zirconium and uranium-fissium alloy fuel pins were exposed to conditions approaching the anticipated parent-core environment. Special facilities were designed to promote maximum fuel centerline temperatures ranging from 700 to 1200°F and fuel atom burnups from 0.4 to 1.56%. The exposure history and the effects of irradiation are described and illustrated photographically. These include dimensional and density changes, fission-gas build-up, and fission-product contamination of the sodium thermal bond as a function of fuel burnup and operating temperature. The lesser physical changes evidenced in the stainless-steel-clad uranium-fissium alloy support its superiority over the zirconium-uranium alloy and its ultimate selection for the initial core loading of the EBR-II. (auth)

3598 (ANL-6205(p.201-13)) OSCILLATOR TESTS IN BRITISH FAST REACTORS. A. R. Baker. United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England.

The oscillator experiments performed in the zero-power fast reactors, Zeus and Zephyr, are described. Brief mention is made of the experiments proposed for the 60-Mw Dounreay Fast Reactor. A description of an experiment to measure the Doppler effect is included because it requires the same type of equipment as oscillator tests of stability and because the Doppler effect, if large, is an important contributor to the power coefficient. (W.D.M.)

3601 (ANL-6205(p.232-49)) RECENT EBR-I STABILITY STUDIES. R. R. Smith. Argonne National Lab., Idaho Falls, Idaho.

The nature of the problems which existed in the earlier EBR-1 loadings is briefly reviewed. The experimental program was divided into two phases: one dealing with the fully ribbed rigid core and the other dealing with various configurations in which stabilizing ribs were removed. Transfer function measurements were carried out at various power levels to test the linearity of the feedback. (W.D.M.)

3602 (ANL-6205(p.251-68)) ANALYSIS OF THE POWER FEEDBACK RELATIONS IN FAST REACTORS. F. Storrer. Atomic Power Development Associates, Inc., Detroit.

A method that APDA proposes to use in the analysis of results of Fermi Fast Reactor oscillator tests is described. The method provides a quick and easy check on the agreement between experimental and predicted feedback characteristics. It is illustrated by an analysis of some of the experimental data obtained from the EBR-1 Mark III tests. A simple stability criterion is presented which indicates without lengthy calculations that the Fermi Reactor will be stable. (W.D.M.)

3603 (ANL-6205(p.269-77)) KINETICS STUDIES ON EBR-I AND EBR-II. PART I. D. Okrent. Argonne National Lab., Ill.

Calculations performed in the course of kinetic studies of EBR-1 and EBR-2 are discussed. The relatively good agreement between experiments on bowing effects and calculations that were made is briefly discussed. (W.D.M.)

3604 (ANL-6205(p.278-9)) KINETICS STUDIES ON EBR-I AND EBR-II. PART II. H. H. Hummel. Argonne National Lab., Ill.

The thermal feedback (the response of the reactor to unrestrained thermal expansion) is considered. Calculation of the temperature distribution in the reactor and the reactivity change associated with expansion of its various parts are discussed. (W.D.M.)

3615 (NDA-Memo-14-114) ARTIFICIAL NEUTRON SOURCE REQUIREMENT FOR THE SUB CRITICAL REACTOR. J. DeFelice. Nuclear Development Corp. of America, White Plains, N. Y. Nov. 25, 1955. 9p.

Calculations were performed to obtain an order-of-magnitude estimate of the artificial neutron source required in the Fermi subcritical reactor. The source is needed so that the flux in the graphite shield will be sufficiently high to be measured by the flux-detecting instruments. The measurements will determine the multiplication of the reactor while it is subcritical. The problem was to determine the neutron flux 1 ft into the graphite shield when a neutron source is placed at the center of the core with all the scram rods inserted. (W.D.M.)

3643 (ANL-6217) SOME CALCULATIONS FOR A PLUTONIUM-FUELED ZPR-III. A. R. Baker. Argonne National Lab., Ill. Sept. 1960. 25p. Contract W-31-109-eng-38. OTS.

Calculations of critical size, reactivity coefficients, and delayed neutron fractions are described for mockups of a plutonium-fueled EBR-I (Mark IV) and five possible metal-fueled power reactors. The calculations for critical size were performed using the SNG transport-theory code. The reactivity coefficients were estimated by SNG calculations or by one-group perturbation-theory desk calculations. The delayed-neutron fractions were estimated by an approximate method, which starts with a calculation of β

for a bare core and then makes a correction for the small effect of the blanket. The reactivity effects studied were the effects of fuel-plate and whole-core expansion and control-rod worths. (auth)

4708 (ANL-6253) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT [FOR] OCTOBER 1960. (Argonne National Lab., Ill.). Nov. 15, 1960. 64p Contract W-31-109-eng-38 OTS.

Research and development activities are reported on EBWR, BORAX-V, ZPR-III, ZPR-VI, ZPR-IX, Juggernaut, EBR-I, and EBR-II. Fast- and thermal-reactor safety studies are summarized. Nuclear technology and general support are treated in terms of applied nuclear and reactor physics, reactor fuels and materials, reactor components, separations, and advanced reactor development (For preceding period see ANL-6234.) (W.D.M.)

4772 (WADC-TN-59-415) FAST VERSUS THERMAL REACTORS FOR PROPULSION AND POWER APPLICATIONS. [Period covered]: September 1, 1959 to October 15, 1959. John E. Brooke and Irving A. Peltier (Wright Air Development Div. Propulsion Lab., Wright-Patterson AFB, Ohio). Nov. 30, 1959. 34p. (AD-238401).

Nuclear characteristics of fast and thermal reactors are compared on the basis of effects of moderator concentration on fuel concentration, flux lethargy, and size of reactors. The concentration of moderator is divided into three ranges, from zero to full concentration. In the small range, neither the fuel concentration nor the reactor size is a strong function of moderator concentration, and the flux spectrum is essentially the same as the fission spectrum. Also reactor size, radius, is inversely proportional to the square of the fuel concentration. In the high range, neither the fuel nor the size is a strong function of moderator concentration, but the flux lethargy spectrum shifts to include a very large percentage of thermal neutrons. In the intermediate or transition range, both the fuel and size are strong functions of moderator concentration, and the flux lethargy spectrum shifts very rapidly toward the thermal range when moderator is increased. The reactor size, however, is a very weak function of fuel concentration. The intermediate range for the U^{235} -C reactor is actually about one order of magnitude wide and shifts toward the increasing moderator concentration direction as the size of the reactor decreases. For reactors below a certain size, the transition from no thermalization to maximum thermalization cannot take place, and these reactors have fast or epithermal flux lethargy spectra regardless of moderator concentration. If void space is filled with heavy, nonabsorbing nuclei, fuel concentrations required for fast reactors can be reduced appreciably. This fact points out an apparent disadvantage of gas-cooled reactors over liquid-metal-cooled fast reactors. (auth)

4824 (TID-7599(p.185-98)) A HIGHLY ACTIVE PILOT PLANT FOR REPROCESSING EXPERIMENTAL FAST REACTOR FUELS. K. Hartley and T. W. Allen (United Kingdom Atomic Energy Authority. Industrial Group. Dounreay Works, Caithness, Scotland).

A highly active pilot plant is described. It provides for investigation of the complete reprocessing of newly developed fast reactor fuels. The three cells and their auxiliaries occupy a single-story building of approximately 3000 sq ft floor area built as a flexible research tool. The building is situated adjacent to the irradiated fuel examination block from where it may conveniently receive the irradiated fuels. (auth)

4846 (TID-7599(p.429-40)) REMOTE ASSEMBLY OF REPROCESSED FUEL SUBASSEMBLIES FOR EBR-II. R. H. Olp (Argonne National Lab., Ill.).

Remotely operated assembly equipment and procedures used in constructing reactor core subassemblies for the first core loading of the Experimental Breeder Reactor No. 2 are described. All this equipment was designed for totally remote operation, maintenance, and replacement in a radiation field of up to 10^6 r/hr, resulting from the reprocessing of spent reactor subassemblies. The operation of this remote equipment is controlled through an electrical programming console exterior to the cell and Argonne type master slave manipulators. (auth)

5791 (NDA-14-82) DOPPLER TEMPERATURE COEFFICIENTS OF REACTIVITY IN FAST REACTORS. H. Yamauchi and J. De Felice (Nuclear Development Associates, Inc., White Plains, N. Y.). June 17, 1955. 40p For Atomic Power Development Associates, Inc.

Formulas and data required for calculating temperature coefficients of reactivity due to the Doppler effect are given. The results of coefficient calculations for the APDA reference design, the EBR, and an ANL critical assembly are reported. (auth)

5792 (NDA-2134-1) A SURVEY OF FAST REACTOR SAFETY PROBLEMS. P. Mittelman and M. Silberstein (Nuclear Development Corp. of America, White Plains, N. Y.). Mar. 29, 1960. 18p. Contract AT(30-1)-2303. OTS

One of the major considerations facing the future development and practical application of fast reactors is the successful solution of the safety problems associated with this type of reactor. A survey of fast reactor fundamental safety problems was conducted at Atomic Power Development Associates, ANL, NRTS, and LASL. Problems that were felt to require additional effort are summarized (W.D.M.)

5851 (ANL-6183) CHEMICAL ENGINEERING DIVISION SUMMARY REPORT [FOR] APRIL, MAY, JUNE 1960. (Argonne National Lab., Ill.). 178p. Contract W-31-109-eng-38 OTS.

Chemical-Metallurgical Processing. A direct-cycle fuel-reprocessing plant using pyrometallurgical procedures is being designed and constructed as part of the Experimental Breeder Reactor No. 2 project. Cable samples sealed with Temporell No. 741 were irradiated to 3.0×10^5 rad. Tests of the absorber-type fume trap were successful. Preparations were continued for high-activity-level demonstrations of the melt-refining process for EBR-2 core fuel. An attractive and efficient procedure for removing uranium skull material from a melt-refining crucible is to convert it to free-flowing oxide powder by oxidation with a dilute oxygen-argon mixture at 700 to 800°C. Two complete runs were made for both demonstrating and pin-pointing difficulties in the drag-out or skull-reclamation process. Investigation of the reduction of uranium oxides by liquid magnesium alloys was continued. Two runs were made to demonstrate recovery of plutonium from a magnesium solution by distillation of the magnesium. Operation of a mild steel corrosion loop for circulating a U-Mg-Cd alloy at 550°C was terminated after 4800 hr of trouble-free operation. The solubility of titanium in liquid cadmium was found to vary between 0.047 at. % at 449°C to 0.15 at. % at 651°C. The solubility of uranium in liquid lead over the temperature range from 415 to 802°C is given. The solubility of uranium in cadmium-magnesium solutions containing up to 72 wt. % Mg at 610°C was found to decrease smoothly from about 2 wt. % in pure cadmium to about 0.002% in 72 wt. % Mg. The partition coefficients of a number of representative fissile and fission product elements between the immiscible liquids aluminum and cadmium at 680 to 688°C were measured. The heat of combustion of tungsten disulfide in oxygen was determined to be

$\Delta E_c^0/M = -1939.29 \pm 0.55$ cal/g for the reaction. The heat of formation of ZrF_4 was determined by combustions of Zr in F_2 . Two series of calorimetric combustions of B in F_2 were carried out. Fuel-Cycle Applications of Volatility and Fluidization Techniques. The Direct Fluorination Volatility Process is being developed to recover uranium and plutonium from spent uranium oxide, Zircaloy-clad fuel elements of the type used in the Dresden nuclear power plant. A study of the kinetics and mechanism of the decomposition of PuF_6 vapor was initiated. The reaction of PuF_6 with SF_4 was investigated. The reaction of SF_4 with UO_3 to produce UF_6 may be used as a basis for a fuel-recovery process. Conversion of UO_2 to UO_3 by means of air and nitric acid vapor and also by means of oxygen and NO_2 was investigated between 350 and 500°C. Several aspects of the Direct Fluorination Process are being studied for the recovery of uranium from highly enriched U-Zr alloy fuels. Experiments incorporating a 2-zone reaction scheme for reacting Zr with HCl-HF gas mixtures were initiated. Laboratory data were obtained on reaction rates of Zircaloy-2 with various mixtures of HF and HCl at various temperatures. A series of fused salt dissolutions of synthetic Zr-U fuel assemblies was carried out in the graphite pilot plant dissolver according to a semicontinuous flowsheet. The conversion of UF_6 to UO_2 by direct reaction with steam and hydrogen is being investigated. The first phase of the experimental work in the 6-in. multistage fluid-bed reactor designed to achieve controlled downward transport of solids using external vibrators and without the use of internal downcomers was completed. Reactor Safety. The considerable inhibition of the burning propagation rates of uranium and zirconium by presence of small percentages of halogenated hydrocarbons in air was studied further. The study of ignition of metal powders has continued with studies of ignition of spherical zirconium powders. Preliminary ignition and burning propagation rate studies with plutonium binary alloy foils containing 2 at. % Al, Fe, C, Ni, Mn, and Cr were completed. The experimental program to determine rates of reaction of molten reactor fuel and cladding metals with water is continuing. Reactor Chemistry. Data are being obtained on the neutron-capture cross sections of U^{238} as a function of neutron energy. Corrosion studies are being made of various turbine constructional materials in alkaline permanganate procedure solutions. Several loop runs were made at 200 psig with irradiated uranium. Corrosion of the uranium pellets occurred within 25 to 30 hr. Construction work is proceeding on apparatus to study the effects of superheated steam on materials of construction. Analytical Research and Development. A fusion technique was developed for dissolution of PuO_2 . Two analytical methods were developed for determination of plutonium. Routine Operations. The operation of the radioactive waste-processing facility and the γ -irradiation facility continued without incident. (For preceding period see ANL-6145.) (W.L.H.)

6043 (ANL-6211) NITRIDATION OF CRUCIBLE SKULLS FROM MELT REFINING. T. R. Johnson, G. F. Brunzie, and R. K. Steunenberg (Argonne National Lab., Ill.), Sept. 1960. 23p. Contract W-31-109-eng-38. OTS.

A description is given of the EBR-2 fuel cycle. The effects of nitrogen on the crucible skulls from melt refining are described at length. (W.L.H.)

6160 PROMPT CRITICAL ASSEMBLY GAMMA RAY SOURCES Charles S. Shapiro (International Business Machines Corp., Owego, N. Y.) Nuclear Sci. and Eng. **8**, 515-17(1960) Dec

The method of obtaining the γ dose rate during a prompt critical burst from the "Godiva" reactor is presented. In this measurement, the fission-product γ -ray contribution to the total γ -ray dose must be determined. The determi-

nation of this quantity is described along with the results. Conclusions are drawn as to the relative contributions to the total dose from prompt, capture, and inelastic collision γ -ray sources (auth)

6161 SPECTRAL COMPARISONS WITH HIGH ENERGY ACTIVATION DETECTORS James Grundl and Arthur Usner (Los Alamos Scientific Lab., N. Mex.) Nuclear Sci. and Eng. **8**, 598-607(1960) Dec

The responses of the activation detectors $P(n,p)$, $Al(n,p)$, $Fe(n,p)$, $Al(n,\alpha)$, and $Cu(n,2n)$ to the neutron spectra at the center of the bare U^{235} and Pu^{239} critical assemblies, Godiva and Jezebel, are the basis of a precision comparison of these two spectra above 2 Mev. Results, interpreted in terms of the spectral function $E^{1/2}e^{-BE}$, show that the average energy of this component of the spectrum in Godiva is $4.7 \pm 0.8\%$ lower than in Jezebel. From some preliminary measurements a close resemblance to the corresponding fission neutron spectra is indicated in this energy region. Results of photoplate measurements in the leakage spectra of both assemblies are considered and analyzed. A discussion and determination of effective thresholds is appended (auth)

6368 METHOD OF MAKING COMPOSITE FUEL ELEMENTS. (to U. S. Atomic Energy Commission). British Patent 853,394. Nov. 9, 1960.

A method for joining the sections of a composite fuel blanket element for a fast breeder reactor and for capping the ends of a fuel element consisting of a uranium rod clad in Zircaloy-2 is outlined in which the exposed uranium is etched to the depth of $\sim 1/10$ mil, a thin disk of zirconium is placed between the ends to be joined, pressure is applied to the joint, the joint is welded with an inert gas-shielded arc to fuse the cladding to 50% of its thickness, and the joint is heated by induction to form a diffusion bond between the uranium and the zirconium disc. A specific example of this method and welding conditions is presented. (D.L.C.)

7034 (CEA-tr-A-804) CONTRIBUTION À LA THÉORIE DES RÉACTEURS À NEUTRONS RAPIDES. (Theory of Fast-Neutron Reactors). R. Schulten (Max-Planck-Institut für Physik, Göttingen). Translated into French from "Zur Theorie der Reaktoren Mit Schnellen Neutronen. (NP-6268) [1954?]. 11p.

This paper was previously abstracted from the original language and appears in NSA, Vol. 11, abstract No. 6899.

7044 GODIVA II AN UNMODERATED PULSE-IRRADIATION REACTOR. T. F. Wimett, R. H. White, W. R. Stratton, and D. P. Wood (Los Alamos Scientific Lab., N. Mex.) Nuclear Sci. and Eng. **8**, 691-708(1960) Dec

Design features of Godiva II, the improved pulsed-reactor successor to Lady Godiva, are discussed together with characteristics of power excursions, and performance is compared with that of the original Godiva. Measurements of the wait time between stepwise reactivity insertion and the occurrence of a burst are presented and compared with theory based on a statistical model of fission chains. Analytical and numerical solutions of the reactor equations are developed to reproduce experimental data and extrapolate to higher energy release. Consideration is also given to perturbations arising from room-returned neutrons. Two different modes of operation are discussed and some design problems of Godiva-type pulsed reactors are briefly mentioned. Typical bursts are illustrated with peak powers up to 13,000 Mw and widths at half-maximum down to 35 μ sec (auth)

7057 THE PHYSICS OF NUCLEAR REACTORS.

Yu. I. Klimov. *Priroda* 49, No. 10, 42-5(1960) Oct. (In Russian)

It is noted that a series of five reactors (BR-1, BR-2, BR-3, BR-4, and BR-5) were successively constructed to develop the theory and practice of fast neutron reactors. The last reactor of this series (BR-5) put on stream in the summer of 1958 with a power of 5000 kilowatts can be considered as the prototype of future, fast neutron, power reactors. Brief mention is made of three research reactors (VVR-2, VVR-S, and IRT) which use distilled water as a moderator and heat transfer medium. A schematic drawing of the IRT research reactor is presented. Twelve research reactors were built in the USSR and various satellite countries. The names of people associated with the development of these reactors are given prominence as Lenin prize winners. (TTT)

7091 (PRDC-TR-38) MONTHLY TECHNICAL REPORT [FOR] PERIOD AUGUST 1960. (Power Reactor Development Co., Detroit). 60p. Contract AT(11-1)-476. OTS.

Fermi Fast Breeder Reactor design progress and major component fabrication status are summarized. Reports are included from an environmental radioactive survey. (W.L.H.)

7110 COMPUTATIONAL SURVEY OF IDEALIZED FAST BREEDER REACTORS. William H. Roach (Los Alamos Scientific Lab., N. Mex.). *Nuclear Sci. and Eng.* 8, 621-51(1960) Dec.

A calculational survey of idealized fast breeder reactors is reported. Representative fuel (Pu^{239} , U^{235} , U^{233}), fertile (U^{238} , Th^{232}), structural (stainless steel), and coolant (sodium) materials are used in the calculations, references providing information as to the substitutional effect of other materials. The calculational method is defined, input parameters are discussed, and results, such as critical mass, critical volume, initial breeding ratios, and spectral indices for the reactors, are tabulated. (auth)

7111 IMPROVEMENTS RELATING TO NUCLEAR REACTORS. Stanley Edgar Barden (to A.E.I.-John Thompson Nuclear Energy Co., Ltd.). British Patent 853,269. Nov. 2, 1960.

The reactivity of a fertile-fissile fuel reactor can be increased if the reactor is designed so that the ratio fertile-fissile atoms varies periodically in successive zones. Two possible fuel arrangements for accomplishing this are described for heterogeneous and homogeneous reactors. (D.L.C.)

8297 IMPROVEMENTS IN OR RELATING TO FAST BREEDER REACTORS. (to Walther and Cie Aktiengesellschaft). British Patent 856,946. Dec. 21, 1960.

A liquid-fuel fast-breeder reactor with continuous fuel processing and low activity outside the reactor is designed, comprising a reactor vessel filled with a liquid fissile metal, an alloy, a suspension or salt and a liquid breeder vessel blanketing the reactor vessel. Pipes are installed in both the reactor and breeder vessels for cooling, withdrawing a minor part of the fissile and fertile liquids for continuous processing, and returning the processed liquids to the vessels. One drawing is included. (D.L.C.)

8375 POWER BREEDER REACTOR. (to United States Atomic Energy Commission). British Patent 857,959. Jan. 4, 1961.

A power breeder reactor is designed so that reactivity changes due to operating power and temperature changes and mass volume changes of the core due to temperature

differentials are minimized. It comprises a core section of vertical multifaced subassemblies surrounded by a blanket section. Means are provided for preventing lateral movement of the subassemblies near their lower ends, so that volume changes are minimized. (D.L.C.)

8378 POWER BREEDER REACTOR. H. O. Monson (to U. S. Atomic Energy Commission). U. S. Patent 2,961,393. Nov. 22, 1960.

An arrangement is offered for preventing or minimizing the contraction due to temperature rise, of a reactor core comprising vertical fuel rods in sodium. Temperature rise of the fuel rods would normally make them move closer together by inward bowing, with a resultant undesired increase in reactivity. According to the present invention, assemblies of the fuel rods are laterally restrained at the lower ends of their lower blanket sections and just above the middle of the fuel sections proper of the rods, and thus the fuel sections move apart, rather than together, with increase in temperature.

8896 THE MELT REFINING OF IRRADIATED URANIUM: APPLICATION TO EBR-II FAST REACTOR FUEL. VI. THE BEHAVIOR OF PLUTONIUM IN THE MELT-REFINING PROCESS. V. G. Trice, Jr. and N. R. Chellew (Argonne National Lab., Ill.). *Nuclear Sci. and Eng.* 9: 55-8(Jan. 1961).

The behavior of plutonium in melt refining of plutonium-bearing uranium alloys was studied as functions of the initial plutonium concentration in the charge, time, and temperature of refining and the carbon impurity added to the charge. Under all conditions studied, the percentage recovery of plutonium in the purified metal was slightly lower than that of uranium. The concentration of plutonium in uranium-20 wt.% plutonium-10 wt.% fission alloy (EBR-II type fuel) melt refined for 3 hr at 1300 or 1400°C was $0.31 \pm 0.23\%$ lower than the plutonium concentration in metal charged to the process. (auth)

8897 THE MELT REFINING OF IRRADIATED URANIUM: APPLICATION TO EBR-II FAST REACTOR FUEL. VII. THE EVOLUTION OF XENON AND KRYPTON. A. Schneider and N. R. Chellew (Argonne National Lab., Ill.). *Nuclear Sci. and Eng.* 9: 59-63(Jan. 1961).

The release of the fission gases, xenon and krypton, from molten irradiated uranium was studied experimentally at total initial gas contents of 0.002 to 138 ppm. At concentration levels anticipated in spent EBR-II fuels, over 99% of the xenon and krypton is expected to be removed from the molten metal. (auth)

8898 THE MELT REFINING OF IRRADIATED URANIUM: APPLICATION TO EBR-II FAST REACTOR FUEL. VIII. THE BEHAVIOR OF RARE EARTHS, YTTRIUM, BARIUM, STRONTIUM, AND CESIUM. N. R. Chellew, G. A. Bennett, and V. G. Trice (Argonne National Lab., Ill.). *Nuclear Sci. and Eng.* 9: 64-70(Jan. 1961).

Melt refining of synthetic EBR-II alloy in oxide crucibles at temperatures between 1200 and 1400°C under an inert atmosphere resulted in a separation of rare-earth elements, yttrium, and alkaline-earth elements from ingot fractions. The rate of cerium removal is shown to behave as a first-order reaction. Cesium is quantitatively volatilized from highly irradiated uranium fuels refined at 1200°C for 4 hr. (auth)

8899 THE MELT REFINING OF IRRADIATED URANIUM: APPLICATION TO EBR-II FAST REACTOR FUEL. IX. SORPTION AND RETENTION OF SODIUM AND CESIUM VAPOR ON STATIONARY BEDS AT ELEVATED TEMPERATURE. J. Wolkoff and A. A. Chilenskas (Ar-

gonne National Lab., Ill.) Nuclear Sci. and Eng., 9 71-7 (Jan. 1961).

Materials capable of trapping alkali metal vapors at temperatures up to 800°C were tested and a number of surface-active materials proved efficient. A sufficient quantity of molecular sieves, activated alumina, or activated carbon can be contained in a crucible cover to trap and hold the evolved alkali metal vapors from an EBR-II melt refining furnace charge of 10 kg. The cover design is under development (auth)

8900 THE MELT REFINING OF IRRADIATED URANIUM APPLICATION TO EBR-II FAST REACTOR FUEL. X. THE BEHAVIOR OF ZIRCONIUM. N. R. Chellew and V. G. Trice (Argonne National Lab., Ill.). Nuclear Sci. and Eng., 9 78-81 (Jan. 1961).

Melt refining experiments in which irradiated uranium or synthetic EBR-II alloys were melted in oxide crucibles showed that removal of zirconium as an oxide does not occur at temperatures up to 1400°C. Removal of zirconium by carbon scavenging of the melt was approximately 50% in 5-hr melts at 1400°C, the mechanism and products of reactions which led to this separation are discussed. Zirconium contamination of uranium-cerium alloys melted in lime-stabilized zirconia crucibles was negligible. (auth)

8901 THE MELT REFINING OF IRRADIATED URANIUM APPLICATION TO EBR-II FAST REACTOR FUEL. XI. BEHAVIOR OF IODINE IN MELT REFINING. N. R. Chellew and M. Ader (Argonne National Lab., Ill.). Nuclear Sci. and Eng., 9 82-6 (Jan. 1961)

Melt refining of synthetic EBR-II alloy in stabilized zirconia crucibles at 1400°C for 3 hr under one atmosphere of helium pressure resulted in almost complete volatilization of iodine from ingot and skull fractions. Removal of iodine was strongly dependent on refining temperature. Volatilized products were primarily iodides of metals initially present as minor oxide constituents of the crucible, no evidence for evaporation as free iodine was found in the melt refining operation (auth)

8902 THE MELT REFINING OF IRRADIATED URANIUM APPLICATION OF EBR-II FAST REACTOR FUEL. XII. THE BEHAVIOR OF RUTHENIUM, MOLYBDENUM, PALLADIUM, RHODIUM, TECHNETIUM, ANTIMONY, CADMIUM, AND TELLURIUM. N. R. Chellew and G. A. Bennett (Argonne National Lab., Ill.). Nuclear Sci. and Eng., 9 87-90 (Jan. 1961)

Ruthenium, molybdenum, palladium, rhodium, technetium, and antimony are retained by the bulk metal when uranium containing macrolevels of accompanying fission elements is melt-refined in oxide crucibles at 1200 to 1400°C. Under similar conditions, cadmium and tellurium are nearly quantitatively removed, the latter element by a proposed mechanism involving the formation of a stable compound with cerium (auth)

10368 (ANL-6218) CALCULATION OF HETEROGENEITY EFFECTS IN ZPR-III FAST ASSEMBLIES USING THE DSN PROGRAM. D. Meneghetti and M. F. Loomis (Argonne National Lab., Ill.). November 1960. 33p. Contract W-31-109-eng-38.

Reactivity changes, flux variations, and detector-response variations for ZPR-III fast critical assemblies, due to variations in heterogeneities caused by various intradrawer configurations of fuel and diluent plates and deviations of reactivities from ideal homogeneous cores, are calculated as reflective boundary slab cells using the DSN code. For these cells the calculated multiplication constants, for a given assembly plate configuration, by $N = 4, 8$, and 16 approximations successively increase such that the eigenvalues obtainable at $N = 16$ are still generally far from the

limiting values at large N . This aspect makes for uncertainty, especially if differences in k_{eff} values from homogeneous are desired. (auth)

10369 (ANL-6295) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT [FOR] DECEMBER 1960. (Argonne National Lab., Ill.) Jan. 16, 1960. 68p. Contract W-31-109-eng-38

Research and development are reported on the following reactors: water-cooled, EBWR, BORAX-V, sodium-cooled, EBR-1, and EBR-II. General studies were also made on nuclear superheating, reactor safety, and nuclear technology (For preceding period see ANL-6269) (W. L. H.)

10503 (CF-59-10-79(Rev.)) REACTOR MATERIALS EFFICIENCY. J. W. Ullmann and C. E. Guthrie (Oak Ridge National Lab., Tenn.). Oct. 23, 1959. 14p.

The materials efficiency, in terms of electrical-megawatts per metric ton of natural uranium required for inventory, was calculated for the reactors included in the 10-year proposals and is presented as a function of reactor type and size. (auth)

10554 ON THE FAST BREEDER REACTOR. Shoji Nomoto, Yorio H. Tomioka, and Tomoo Nakamura (Japan Atomic Energy Res. Inst.) J. At. Energy Soc. Japan, 2 622-37 (1960) (In Japanese)

A review is presented of the physical characteristics and present status of the fast breeder reactor. The neutron energy distribution, neutron spatial distribution, power distribution, and some consideration of the design of the fast reactor are described in order to evaluate the breeding potentiality. The kinetics and control characteristics of the fast reactor, such as the prompt neutron life time, effective delayed neutron fraction, reactivity change due to temperature change, and power instability are discussed. The experimental results with critical assemblies are summarized (auth)

10561 CAPITAL-COSTS REDUCED FOR FAST BREEDERS. John G. Yevick and Alfred Amorosi (Atomic Power Development Associates, Inc., Detroit). Nucleonics, 19, No. 2, 64-9 (Feb. 1961)

The APDA Plutonium Fueled Fast Breeder Reactor (PFFBR) represents a 50% saving in capital cost over the EFFBR and EBR-II designs fixed in 1955. Technological advances, improved understanding of design requirements, and operation of prototypes and models over the intervening five years resulted in a new fuel material, redesigned auxiliaries, and cheaper shielding. The design improvements involved plant layout, reactor vessel, plug, refueling equipment, sodium pumps, heat exchangers, and steam generator. The 150 Mw(e) PFFBR will operate on recycle plutonium with only depleted uranium makeup. The new shield materials and arrangement allow easier access and cost savings (T.R.H.)

10562 BRITISH NUCLEAR POWER STATIONS. Rolt Hammond. London, Macdonald & Co., Ltd., 1961. 189p.

This survey of the British nuclear power program includes an outline of the plan for the next 20 years, descriptions of nuclear power stations, discussions of some of the thermonuclear research in progress, and a review of the special problems of nuclear power. The Calder Hall, Hinkley Point, Berkeley, and Hunterston Power Stations are described. Electrical control and switchgear for the reactors are also described. The research going on at the Dounreay Fast Breeder Reactor and at Harwell is reviewed (M. C. G.)

11118 (APAE-78(Vol.I)) INTERMEDIATE HEAT EXCHANGER AND STEAM GENERATOR FINAL DESIGN.

THERMAL AND MECHANICAL DESIGN. (Alco Products, Inc., Schenectady, N. Y.). Sept. 30, 1960 382p. Contract AT(11-1)-666.

The final designs are given for a sodium-to-sodium intermediate heat exchanger and a sodium-to-water steam generator. The basic design of the heat exchanger is a shell with fixed tube sheets and sine wave tubes; this offers the advantage of tubes operating essentially independently of the shell or adjacent tubes. The steam generator is designed basically as a shell-and-tube unit made of three sections (1) a boiler section, (2) a steam drum, and (3) a superheater section. A thermal design analysis is given for the steam generator. Fabrication concepts are outlined for the heat exchanger and steam generator. The design data and drawings are presented. Mechanical and thermal design calculations, developed during the final design phase, are included. Cost estimates were prepared for the heat exchanger, boiler section, and superheater section for 30- and 70-Mw capacity. (B.O.G.)

11119 (APAE-78(Vol.II)) **INTERMEDIATE HEAT EXCHANGER AND STEAM GENERATOR FINAL DESIGN. CHEMICAL AND STRESS ANALYSIS.** (Alco Products, Inc., Schenectady, N. Y.). Sept. 30, 1960. 450p. Contract AT(11-1)-666.

Chemical and stress analyses were made of a sodium-to-sodium intermediate heat exchanger and a sodium-to-water steam generator as an extension and modification to previous analyses. The chemical analysis covers the sodium cover-gas system and the effects of sodium water reactions in the event of a leak in the steam generator. The stress analysis was made of those thermal transients that would be physically possible with the intermediate heat exchanger and steam generator design. Attention was given to methods of operation that would minimize the magnitude and frequency of thermal shocks. A structural design basis is included for handling stress analysis of combined mechanical, hydrostatic, and thermal stresses. Conditions for using creep and stress rupture properties are given. (B.O.G.)

11120 (APAE-78(Vol.III)) **INTERMEDIATE HEAT EXCHANGER AND STEAM GENERATOR FINAL DESIGN. SPECIFICATIONS.** (Alco Products, Inc., Schenectady, N. Y.). Sept. 30, 1960. 90p. Contract AF(11-1)-666.

Twenty-three material, inspection, and welding specifications are presented for the various parts of the intermediate heat exchanger and steam generator. Tables are given to indicate the applicable parts and assemblies to which the specifications apply. (B.O.G.)

11912 (ANL-6212) **SOME PHYSICS CALCULATIONS ON THE PERFORMANCE OF LARGE FAST BREEDER POWER REACTORS.** S. Yiftah and D. Okrent (Argonne National Lab., Ill.). Dec. 1960. Contract W-31-109-eng-38. 32p.

Critical mass, spectrum, breeding ratio, and coolant removal coefficients were calculated for a series of large Pu-U²³⁵-fueled sodium-cooled fast-breeder power reactors, using a new 16-group cross-section set based in part on recent microscopic cross-section measurements. The parameters studied include reactor size, plutonium isotopic content, and type of structural material. Reactors cooled with Pb-Bi eutectic and those containing U²³⁵-Th fuel were examined. (auth)

12451 (ANL-6269) **REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT** N Hilberry (Argonne National Lab., Ill.) November 1960 70p Contract W-31-109-eng-38

General research and development are reported on water-cooled and sodium-cooled reactors. Studies are also re-

ported on reactor safety and nuclear technology (For preceding period, see Report ANL-6253) (W L H)

12452 (ANL-6307) **REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, JANUARY 1961.** (Argonne National Lab., Ill.). Feb. 15, 1961. 71p. Contract W-31-109-eng-38.

Modifications of EBWR are described, and data from analysis of control-rod operational crud deposit are tabulated. Development and construction progress on Borax V are summarized. Research for ZPR III is reported on a series of uranium oxide critical cores. Comparative data on composition, critical masses, and central fission ratios are tabulated for these assemblies. Developmental work on ZPR-VI, ZPR-IX, and Juggernaut is summarized, followed by a summary of EBR-I operation and development of EBR-II. Studies of reactor steam superheating concepts are reported from which conclusions are listed. In reactor safety studies, data were obtained on the extent of water reaction with various metal combinations for use in cores as function of energy input. In other work in-pile melt down studies were conducted. Operation and development of TREAT are described. Nuclear technological developments are summarized in sections on physics, fuel development, reactor components and materials, separations development, and advanced reactor development. (For preceding period see ANL-6295.) (J.R.D.)

12453 (APAE-78(Vol.IV)) **INTERMEDIATE HEAT EXCHANGER AND STEAM GENERATOR FINAL DESIGN OPERATION AND MAINTENANCE.** (Alco Products, Inc., Schenectady, N. Y.). Sept. 30, 1960 89p Contract AT(11-1)-666

Information is given on installation and shipping, scram and casualty shutdowns, leaks, and instrumentation and control (W.L.H.)

12522 (ANL-6266) **INSTABILITY STUDIES WITH EBR-I, MARK III** R R Smith, J F Boland, F D McGinnis, M Novick, and F W Thalgott (Argonne National Lab., Ill.) Dec 1960 92p Contract W-31-109-eng-38

The results of instability studies suggest that the operational characteristics of the fully ribbed and rigid Mark III loading are governed by feedback processes that guarantee safe and stable operation under normal operating conditions. No evidence of positive reactivity effects was noted for the fully-ribbed rigid core. Logical extrapolations of full-power full-flow test data indicate that the reactor could be brought into a resonant condition for power levels exceeding 1000 Mw. Strong nonlinearities were observed in power coefficients and were considered to pose no serious operational problems above 200 kw in the power range associated with the tests. An application of the Nyquist stability criterion to the extrapolated fully sheared data results in the conclusion that the reactor would attain resonance instability at full flow at ~11 Mw. Logical extrapolations of test data for one-third flow results in resonance instability at ~10 Mw, a power level ~20 times that designed for one-third flow. Rib shearing was shown to result in an unexpected and unexplained increase in the magnitude of the delayed structural power coefficient component. An empirical fit of feedback data to a model describing the dynamic and static behavior of the partially sheared core resulted in the values for the respective prompt negative, rod-bowing, and delayed structural power coefficient components of -2.21×10^{-6} , $+0.543 \times 10^{-6}$, and $-0.873 \times 10^{-6} \Delta k/k/kw$. The overall behavior of the Mark III core, whether fully ribbed and rigid, fully ribbed and loose, or even partially or fully sheared, is one of extreme stability. The inclusion of stabilizing ribs and a system of tightening rods in the Mark III design eliminated, or at least reduced to a point beyond de-

tection, the prompt positive rod-bowing component that existed in the Mark II (B O G)

12649 (ANL-6231) CHEMICAL ENGINEERING DIVISION SUMMARY REPORT, JULY, AUGUST, SEPTEMBER, 1960. (Argonne National Lab., Ill.). Contract W-31-109-eng-38. 182p.

Chemical-Metallurgical Processing. The first of a series of demonstrations at high-activity levels of the melt-refining process for EBR-II core fuel was completed. An experiment was also completed on the evolution of fission-product krypton and xenon from an irradiated fuel pin as it was heated to a temperature above the melting point. In tests of alternate materials for use in a melt-refining furnace, a fibrous potassium titanate grain retainer was found to be a very effective heat insulator, but to have less strength than rigid Fiberfrax retainers. The skull remaining in the zirconia crucible after a melt-refining operation must be processed to recover, as partially purified metal, the fissionable material for return to the fuel cycle. Several essentially quantitative reductions of uranium dioxide and skull oxides were achieved in times of less than 8 hr at 800°C in dilute magnesium-zinc solutions and in magnesium containing a small percentage (0.5 to 2) of sodium as a wetting agent. Data and equations for solubilities of other elements in liquid cadmium are included. The partition coefficients of a number of representative fissile and fission product elements between the two immiscible liquids, lead and zinc, at about 735°C were measured. The free energy of formation of the uranium-thallium intermetallic compound UTl_3 was measured, using a galvanic cell method. Galvanic cell measurements with the cerium-zinc system at 443 to 742°C were also made. The determination of the heat of formation of zirconium tetrafluoride by combustions of zirconium in fluorine was completed. A value of -372.44 kcal/mole was obtained for the heat of formation of molybdenum hexafluoride. Calorimetric combustions of boron in fluorine are being continued. Exploratory combustions of magnesium, aluminum, zinc, and cadmium are being carried out. **Fuel Cycle Application of Volatility and Fluidization Techniques.** Investigation of the stoichiometry of the reaction of plutonium hexafluoride with sulfur tetrafluoride was completed. The investigation of the stoichiometry of the reactions of sulfur tetrafluoride with uranium trioxide, uranium dioxide, uranyl fluoride, and U_3O_8 was also completed. Radiation decomposition of plutonium hexafluoride in the presence of uranium hexafluoride at 2.6% per day, was found to be very similar to the value obtained for plutonium hexafluoride alone. In terms of wt.% hydrogen chloride, the solubility of hydrogen chloride in liquid hydrogen fluoride is 0.7 at -38°C and 2.4 at -78°C . An investigation of the corrosive effect of hydrogen fluoride-hydrogen chloride gas mixtures on A-nickel has been initiated. Additional studies of metal fluorinations were concerned with the mechanisms of the nickel-fluorine and nickel-oxygen reactions. Work continued on engineering-scale studies of the Direct Fluorination Process applied to dense uranium dioxide pellet fuel. Decladding runs were made to investigate the effect of close packing of fuel elements on operation of the fluid-bed reactor. Mock-up tests of heat transfer from the surface of fixed packing submerged in a fluid bed were made to determine the effects of bed height and gas velocity. A final dissolution was performed in a series of dissolutions of synthetic fuel assemblies to demonstrate operation of the graphite pilot-plant dissolver. Examination of the walls of the graphite dissolver showed the expected presence of fused salt in the middle of the lamplack insulating zone. A fused-salt charge was allowed to freeze in the dissolver to determine the effect that this would have on dissolver components. Extensive damage to internal graphite downlines and heaters was found. **Reactor Safety.** In the continued study of the ignition of uranium it was confirmed that an aggregate

or array of pieces ignites at a considerably lower temperature than an individual piece of the same specific area. Continued studies of the effects of halogenated hydrocarbons on the burning propagation of uranium and zirconium foils indicate the possibility of preventing continued combustion of individual zirconium foils or turnings thicker than 0.05 mm (2 mils). A self-consistent mechanism for the isothermal oxidation of uranium has been devised. The experimental program to determine rates of reaction of molten reactor fuel and cladding metals with water is continuing. A series of in-pile experiments was conducted in TREAT with oxide-core pins, cermet-core pins, and uranium wires. **Reactor Chemistry.** Neutron-capture cross sections for U^{238} were obtained for neutrons at 0.475 to 1.74 Mev. The Slurrax Reactor chemical supporting program was terminated and a final report is being written. In the Reactor Decontamination Program a study is being made with a stainless-steel loop that simulates the action of a boiling-water reactor into which mixed fission products can be introduced. It was found that the presence of oxygen during the contamination portion of an experiment reduced the effectiveness of decontamination by the alkaline permanganate-citrate procedure. Corrosion studies of various turbine materials in typical decontamination reagents were continued. Pitting of chromium-molybdenum steels in citrate solutions adjusted to pH 10 was found to be appreciably reduced when hexamethylenetetramine was added as an inhibitor. In the loop runs, radioactivity was introduced in the form of irradiated uranium which reacted with the hot water to simulate a fuel rupture. Analysis of loop liquid over a 12-day period indicated that the concentrations of uranium and fission products did not remain constant but exhibited variations both upward and downward. The fission products Zr^{95} , Ru^{103} , and Ba^{140} - La^{140} were deposited on the metal sample strip in the disengaging section in the same relative proportions as they were present in the circulating liquid. Cerium-141 deposition in the liquid phase was lower than that suggested by its relative content in the circulating liquid. (For preceding period see ANL-6183.) (J.R.D.)

12662 (TID-8533) STUDY OF NUCLEAR POWER PLANTS CAPITAL AND POWER GENERATION COSTS 44 AND 12 65 MWE (GROSS) FOR UNITED STATES ATOMIC ENERGY COMMISSION, SAN FRANCISCO OPERATIONS OFFICE, BERKELEY, CALIFORNIA (Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif.). Oct. 31, 1960. 190p

This report may also be identified as KE-60-19

Results of studies to develop basic parameters are presented for normalization of reactor capital cost, fuel cycle cost, and operation and maintenance cost. Included are estimates of capital and power generation costs for central station nuclear power plants with gross-rated capacities of 12 65 and 44 Mw(e). Data are presented for pressurized-water, boiling-water, organic-cooled, sodium-graphite, and sodium-fast reactor concepts. Also included are data for heavy-water reactors and helium-cooled graphite-moderated reactors. (J.R.D.)

13272 (NASA-TN-D-769(p.25)) LIQUID-METALS CORROSION RESEARCH. Vincent J. Rutkauskas (Argonne National Lab., Ill.)

Refractory metals and alloys were exposed to flowing sodium at 650°C to select suitable materials for fuel jackets of future EBR-II fuel loadings. The effects of oxygen concentration and temperature on the corrosion resistance of refractory metals and stainless steels were studied. An experiment is planned to determine the effects of heat transfer and fluid flow in two-phase sodium systems. (M C G)

13909 (APDA-133) FAST REACTOR CORE DESIGN PARAMETER STUDY. (Atomic Power Development As-

sociates, Inc., Detroit). Mar. 1960. Contract AT(11-1)-800. 321p.

Parametric studies of 11 fast reactor fuel systems were made to determine the important design and economic factors on which the cost of producing electric power depend. Some of the studies were applied to a U^{235} -fueled fast reactor and others to a plutonium-fueled fast reactor. The method used consisted of designing a number of cores of different characteristics for use in the reactor power plant and calculating the fixed charges and fuel cycle costs for each design, using the values for fuel burn-up, fabrication costs, and aqueous reprocessing costs developed as part of the studies. Thermal considerations influenced both the core designs and economics to a greater extent than any other factor. It was found that the general method of approach to parametric studies is very important. When a variable was treated independently, a larger variation in results was obtained than actually existed. Fixed charges and total power costs decreased as plant size was increased. Fuel cycle costs increased with coolant temperature. There appeared to be an optimum power density for each fuel system studied. An optimum pressure drop per unit length of core was observed. It was found that if the burn-up could be increased for a given design, the total power cost would be decreased. For fuel systems where clad thickness determined permissible burn-up, there was a broad range of clad thickness for which almost no effect on total power costs was noted. For fuel elements whose burn-up was independent of clad thickness, increased volume of cladding resulted in increased total power costs. A high fuel volume fraction was achieved with plates rather than pins for the same temperature burn-up conditions. Power costs were relatively unchanged by variations within the ranges of uncertainties of nuclear constants such as cross sections. (auth)

13961 FAST NEUTRON REACTION SYSTEM. W. H. Zinn. U. S. Patent 2,975,117. Mar. 14, 1961.

A nuclear reactor is described in which nuclear fission is caused by high-energy neutrons. There is a central reactive zone containing closely spaced vertical rods of high fissionable material content. Surrounding this zone is a first reflector zone made up of vertical rods of uranium of normal isotopic content. A second reflector zone contains natural uranium in wedge-shaped blocks laid around the first reflector zone to form a hollow cylinder. This second zone has one or more movable doors that can be opened varying amounts to control the reactor by varying neutron leakage. The whole second reflector may be dropped from position for emergency control. The second reflector zone is enclosed in a graphite moderator-reflector to return thermal neutrons to the reflectors for plutonium conversion.

13969 (ANL-6261) A LUNAR POWER PLANT. R. H. Armstrong, J. C. Carter, H. H. Hummel, M. J. Janicke, and J. F. Marchaterre (Argonne National Lab., Ill.). Dec. 1960. Contract W-31-109-eng-38. 137p.

A concept of a nuclear power plant to be assembled on earth and operated on the moon is presented. The two principal design objectives are reliability and high specific power. Wherever there is an incompatibility between these two objectives, the decision favors reliability. The design is based on the premise that the power plant must be designed on the basis of current technology and with a minimum amount of research and development. The principal components consist of a fast reactor in a direct cycle with a mercury-vapor turbine. The high-frequency generator, hydrogen compressor for the generator cooling system, mercury-recirculating pump, and condensate pump are on an extension of the turbine shaft. The mercury vapor is condensed and the hydrogen cooled in wing radiators. The reactor is of a construction quite similar to EBR-I Mark III

for which there is a large amount of operating experience. The radiator is a vertical tube-and-fin type built in concentric cylindrical sections of increasing diameter. The curved headers are connected by swivel joints so that, upon arrival, the radiator can be quickly unfolded from the compact cylindrical package it formed during transportation. (auth)

13970 (ATL-505A) EVALUATION OF MERCURY COOLED BREEDER REACTORS. Donald W. Battles (Advanced Technology Labs. Div. of American-Standard, Mountain View, Calif.). Dec. 14, 1960. 22p.

A technical and economic evaluation of a mercury-cooled fast breeder reactor is presented. The objectives of the program were to establish the technical feasibility of a fast breeder reactor cooled with boiling mercury and to evaluate the long-range potential of such a reactor power plant for production of economic power. Details of the conceptual design of a 100-Mw(e) reactor and system are discussed. The power cost from a mercury cooled fast breeder reactor was estimated as 21.4 mills/kwh which is competitive with the power cost for the initial Enrico Fermi plant. It was concluded that this reactor concept is technically feasible and has promising long-range economic potential. (M.C.G.)

13977 (GEAP-3287) FAST OXIDE BREEDER-REACTOR. PART I. PARAMETRIC STUDY OF 300(e) MW REACTOR CORE. P. Greebler, P. Aline, and J. Sueoka (General Electric Co. Atomic Power Equipment Dept., San Jose, Calif.). Nov. 15, 1959. Contract AT(04-3)-189. 59p.

Physics scoping studies of a 300-Mw(e) PuO_2-UO_2 -fueled fast-breeder reactor are reported. Physics design parameters that effect fuel costs, full conservation, and reactor safety were evaluated for use in the selection of parameters for a reference design. The total breeding ratio varied from 1.1 to 1.5 in the range of parameters considered. Plutonium core loading ranged from 500 to 1500 kg. Doubling time was found to be reduced by high-density fuel and low steel content. A compromise figure on fuel-rod range of sizes (about 200 mils) yields a \$2.5 operating reactivity and a small, negative sodium temperature coefficient. (J.R.D.)

13999 (TID-11025) ANALYSIS AND SIMULATION STUDY OF THE OPERATING CONTROL SYSTEM FOR THE ENRICO FERMI ATOMIC POWER PLANT. Report No. 59-12. (Holley Carburetor Co., Warren, Mich.). Nov. 1, 1959. 302p. For Power Reactor Development Co., Detroit.

Includes Addendum No. 1 (EM60-1). Jan. 15, 1960.

An analog simulator was developed to simulate one of the three circuits of the Enrico Fermi Power Plant. Included in the simulation were the reactor, intermediate heat exchanger, mixing and transport delay phenomena, and steam-system hydraulics. Revisions and additions to the steam-system simulation and refinements of the automatic feedwater-steam control subsystem, as simulated, are outlined in the addendum. (J.R.D.)

15209 (ANL-6328) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, FEBRUARY 1961. (Argonne National Lab., Ill.). Mar. 15, 1961. Contract W-31-109-eng-38. 80p.

Design, development, and testing efforts were continued on BORAX-V, EBR-I, EBR-II, EBWR, JUGGERNAUT ZPR-III, ZPR-VI, and ZPR-IX. An evaluation program is outlined for Pebble Bed Reactor designs. Fast and thermal reactor safety studies were conducted. Experimental and theoretical studies in applied nuclear and reactor physics are described. Developments made in reactor components, fuels, and materials are discussed. Heat engineering studies were conducted on steam separation, and velocity and void distri-

butions in two-phase systems. Fluidization and fluoride volatility separation, and chemical-metallurgical separation processes were studied. Advanced reactor concepts that were discussed included: Basic Radiation Effects Reactor, Biogeonuclear Reactor, Fast Reactor Test Facility, compact high-power density fast reactors, AHFR hydraulic test loop, Packed Bed Reactor, and direct conversion. (For preceding period see ANL-6328.) (B.O.G.)

15228 ON THE DESIGN AND MANAGEMENT OF FAST REACTOR BLANKETS. S. A. Hasnain and D. Okrent (Argonne National Lab., Ill.). Nuclear Sci. and Eng., 9: 314-22 (Mar. 1961).

The performance of some blanket designs is studied using economically optimized cycling based on a simple economics model. For an 800-liter core fast reactor having a 45-cm radial blanket and an average core power of 1-Mw per liter, it appears that the outermost blanket elements make enough plutonium to pay for the cost of their fabrication and processing, unless the core power density falls well below the expected value. A cyclic motion of elements in the inward radial direction has little effect on the economics if optimum cycling is followed. Moving the blanket elements may have engineering advantages however, such as a uniform buildup and burnup, and less variation in power locally with time. A paste blanket with radial inward motion and axial mixing has a similar behavior. Inclusion of moderating material in a fast reactor blanket is not promising for a high-power density reactor using optimum cycling, but it may prove valuable if blanket fluxes get very low or the residence times of the blanket elements are limited. (auth)

15261 NUCLEAR FAST BREEDER REACTOR. (to U. S. Atomic Energy Commission). French Patent 1,215,419. Apr. 19, 1960.

Three types of fast breeder reactors are described. The reactor cores are supported inside a container so as to be spaced from the walls. Molten fuel including plutonium is arranged inside spaces bounded by the walls of suitable parallel tubes, which are in contact at their ends through headers inside the core. The amount of plutonium within the core is sufficient for critical reactivity. The tubes form a heat exchanger transferring the heat to molten sodium flowing through the core. Suitably the reactors have breeding blankets of natural uranium. Control of the reactivity may be effected by variation of the amount of molten fuel within the reactors. (NPO)

15267 FUEL ELEMENT FOR NUCLEAR REACTORS. (to General Electric Co.). French Patent 1,220,386. Jan. 4, 1960.

Solid fertile material of cylindrical or other suitable shape is covered by a layer containing fissile material, which in its turn is enclosed or bounded by an immediately adjacent container wall, acting as a heat transfer surface for the element. The fertile material consists of U^{238} or Th^{232} or of U alloyed with either Al, Be, Cr, Fe, Pb, Mn, Ni, Sn, Ti or Zr. As fissile material U^{235} is initially used and later on Pu^{239} or U^{233} . The volume of the "fissile" layer comprises 1 to 40 vol.% (generally 20%) of the total volume of the fuel element, not counting that of the container. This construction can be adapted to rod-shaped, tube- or plate-type fuel elements. (NPO)

15275 (ANL-6301) IDAHO DIVISION SUMMARY REPORT, JULY, AUGUST, SEPTEMBER 1960. (Argonne National Lab., Ill.). Contract W-31-109-eng-38. 146p.

Experimental Breeder Reactor I. The fully ribbed and rigid Mark III loading of EBR-I was found to be governed by feedback processes which guarantee safe and stable operation under normal operating conditions and to give a

large radial contribution to the power coefficient. Nonlinearities in the power coefficient were investigated and found to be no problem. If the stabilizing ribs are removed from the fuel rods, a strong positive effect appears which is associated with the inward bowing of fuel rods. The prompt positive coefficient observed in Mark II is discussed from the standpoint of Mark III tests. A 800-Mwh irradiation run was made on a number of samples, and some brick cladding failures are reported. Data are given for the dimensional changes in EBR-I, Mark III fuel rods used for a total of 2,682 Mwh operating time; the fuel rods usually increased in diameter and decreased in length, and some bowing was observed. The growth and temperature profiles of the fuel rods are compared, and the effects of radial restraint on the rod growth are discussed. The EBR-I, Mark IV core design is then discussed. The fuel rod will incorporate four plutonium-10 at.% aluminum fuel slugs with two depleted uranium blanket slugs. Calculations were made on the critical mass of Mark IV, which is shown to be 28.3 kg of total plutonium. Zero-power Reactor III (ZPR-III). A mockup of EBR-II was studied in ZPR-III, and the worth of the mockup control rods was evaluated with tantalum and B_4C followers. An EBR-II B_4C oscillator rod experiment was made in which the excess reactivity was measured as a function of the angular position of the oscillator. The worths of sodium, aluminum, and stainless steel were mapped throughout the core and blanket. From substitution experiments, there appears to be some spectral differences at the center of the clean cores, depending on whether the core is filled with sodium or aluminum. A two-dimensional mapping of the worth of U^{235} and U^{238} was also performed, and their fission rates were determined. Since the EBR-II shields and thimble holes have been redesigned, new mockups were made and their counter responses studied. Transient Reactor Test Facility. The equipment and procedures used to obtain constant power or flat top pulse bursts for transient testing of fuel elements are described. The poison sections of the control rods were modified by mixing epoxy resin with graded boron carbide to prevent poison movement. The radiation effects of pulsed bursts on pressure transducers were studied, and it was found that the extraneous pressure signal following the instantaneous power, but not that following the integrated power, can be eliminated. Boiling Reactor Experiment V (Borax V). A general review is given of work done to date, and the design of the reactor and plant is discussed in detail. The core structure is discussed, particularly the spring which allows differential expansion between the core structure and the reactor vessel. A comparison of boiling fuel rods with different diameters is given. The superheater fuel assembly was redesigned with 4 instead of 5 fuel plates because the maximum surface temperature has been lowered from 1200 to 1100°F. The reactor control system is compared with those of previous BORAX reactors and EBWR. The construction of the control rods is discussed; the control rod drives which are to be used are those originally used on EBWR. The fuel handling system is discussed. The programs for testing superheater-fuel assembly seals and developing in-core instrumentation are described. Argonne Fast Source Reactor (AFSR). The present status of AFSR is discussed. Data are presented for the neutron fluxes at various points in AFSR and for the AFSR dimensions. (D.L.C.)

15291 (PRDC-TR-39) MONTHLY TECHNICAL REPORT [ON APDA ACTIVITIES FOR] SEPTEMBER 1960. (Power Reactor Development Co., Detroit). Contract AT(11-1)-476. 26p.

Research and development activities on the Fermi Fast Breeder Reactor are summarized under the following headings: core design, materials and metallurgy, nuclear engi-

neering, health physics, reactor vessel engineering, mechanical handling, instrumentation, liquid metal and steam systems, and test operations. An environmental radioactivity report is also included. (M.C.G.)

15292 (PRDC-TR-40) MONTHLY TECHNICAL REPORT [ON APDA ACTIVITIES FOR] OCTOBER 1960. (Power Reactor Development Co., Detroit). Contract AT(11-1)-476. 26p.

Development of the Fermi Fast Breeder Reactor is reported in terms of core design, nuclear engineering, materials and metallurgy, mechanical handling, instrumentation, liquid metal and steam systems, and test operations. The status of the PRDC preoperational, environmental survey and the results of airborne dust sampling and analyses are given. (M.C.G.)

15293 (PRDC-TR-41) MONTHLY TECHNICAL REPORT [ON APDA ACTIVITIES FOR] NOVEMBER 1960. (Power Reactor Development Co., Detroit). Contract AT(11-1)-476. 18p.

Development of the Fermi Fast Breeder Reactor is reported in terms of core design, materials and metallurgy, nuclear engineering, mechanical handling, liquid metal and steam systems, instrumentation, and test operations. Results of the airborne dust sampling and analysis program are included. (M.C.G.)

16006 (ANL-6177) A REPORT ON SOME ATTEMPTS TO CAST CENTRIFUGALLY FUEL ELEMENTS OF SMALL DIAMETER. F. L. Yaggee (Argonne National Lab., Ill.) Mar. 1961 Contract W-31-109-eng-38 22p

The applicability of the centrifugal casting technique to the production of multiple castings of fuel pins of small diameter and of thin fuel plates was investigated. Fuel pins measuring 0.185 in. in diameter by $4\frac{1}{4}$ in. long of unalloyed uranium and of a uranium-2 wt % zirconium alloy were cast successfully in batches of sixteen pins per melt. Sixteen different metals and alloys were used as mold materials. Smaller and longer fuel pins, 0.165 in. in diameter by $9\frac{3}{4}$ in. long, of similar compositions were cast successfully in brass and copper molds. Thirty-six pins of the same diameter and length were cast simultaneously in each casting run. Attempts to cast centrifugally thin uranium plates measuring 9 in. long by 2 in. wide by 0.04 in. thick proved to be only partially successful, but encouraging. These plates were cast into graphite molds at the rate of six plates per run. The maximum usable length of the unalloyed uranium plates cast did not exceed six inches. (auth)

16651 (TID-12489) A CROSSCORRELATION METHOD FOR MEASURING THE IMPULSE RESPONSE OF REACTOR SYSTEMS J. Douglas Balcomb, Elias P. Gyftopoulos (Massachusetts Inst. of Tech., Cambridge), and Howard B. Demuth (Los Alamos Scientific Lab., N. Mex.). [1960?] 29p.

A crosscorrelation method for the study of the small signal response of nuclear reactor systems is presented. The technique and equipment necessary for the implementation of the method are discussed. Results from experiments performed on the Godiva and Kiwi-A3 reactors are reported. The experiments showed that the method gives very satisfactory results under adverse experimental conditions. (auth)

16713 (NDA-2116-9) SPONGE FUEL EVALUATION. Quarterly Technical Report for period Ending December 31, 1960. J. M. McKee (Nuclear Development Associates, Inc., White Plains, N. Y.). Feb. 28, 1961. Contract AT(30-1)-2303. 12p.

In the program of evaluation of sponge fuel (uranium

powder vibrated to maximum density in a container tube and infiltrated with sodium), the irradiation of capsule NDA 4-2 was completed at 0.88% burnup of all uranium atoms and the capsule was replaced in the reactor by capsule NDA 4-3. The thermal conductivity of capsule 4-3 was measured and found to be identical with that of 4-2. The irradiation and temperature history of capsule 4-3 and the fabrication of capsule 4-4 are discussed. (D.L.C.)

16714 (NDA-2147-4) REVIEW OF INVESTIGATIONS PERTINENT TO THE PREDICTION OF THE THERMAL BEHAVIOR OF FAST REACTORS UNDER ABNORMAL OPERATING CONDITIONS. F. E. Beers (Nuclear Development Associates, Inc., White Plains, N. Y.). Mar. 15, 1961. Contract AT(30-1)2303. 41p.

Results are summarized for a survey of methods for predicting fuel element and coolant behavior in sodium-cooled fast reactors under abnormal operating conditions. Circumstances prior to, during, and after meltdown are considered, and the behavior of fuel and coolant during transients is discussed. Reactor excursions, theory of fuel element thermal transients, reactor coolant transients in single and two-phase flows, and specific topics such as two-phase flow patterns, two-phase pressure drop, and transient voids in boiling coolant reactors are also treated. Some applicable mathematical treatments are given along with an annotated bibliography. (D.L.C.)

16761 FAST REACTOR SAFETY STUDIES IN TREAT A STATUS REPORT C. E. Dickerman, E. S. Sowa, and D. Okrent (Argonne National Lab., Ill.) Nucleonics, 19 No. 4, 114; 116, 118, 121; 150 (Apr. 1961).

Results of tests in TREAT on Fermi-1 and EBR-2 Mark-1 fuel elements are reported. Neutron flux pulses from 0.2 to 30 sec duration are applied to the test fuel elements, with temperatures from 950 to 1015°C being achieved. The extent and nature of the fuel element failures, both in air and stagnant sodium, are reported. (T.F.H.)

17278 (ORNL-3077) FABRICATION DEVELOPMENT OF UO₂-STAINLESS STEEL COMPOSITE FUEL PLATES FOR CORE B OF THE ENRICO FERMI FAST BREEDER REACTOR. J. H. Cherubini, R. J. Beaver, and C. F. Leitten, Jr. (Oak Ridge National Lab., Tenn.). Apr. 18, 1961. Contract W-7405-eng-26. 96p.

The development of an inexpensive composite fuel plate with a high burnup potential for application in a 500°C sodium environment as Core B of the Enrico Fermi Fast Breeder Reactor is described. The dispersion fuel product consists of 35 wt. % spheroidal UO₂ dispersed in type 347B stainless steel powder and clad with wrought type 347 stainless steel. Nominal over-all dimensions of Type II design fuel plates are 18.97 in. long x 2.406 in. wide x 0.112 in. thick with 0.005-in. cladding. Reliable processing methods for achieving a uniform distribution of spheroidal UO₂ in the matrix powder and cladding the sintered powder compact by roll bonding are described. Examination of experimental plates reveals that the degree of UO₂ fragmentation and stringering encountered during processing is primarily a function of the degree of cold work employed in the finishing operation and the starting quality of the UO₂ powder. Cladding studies indicate that a sound metallurgical bond can be achieved with an 87.5% reduction in thickness at 1200°C and that close processing control is required to meet the stringent tolerances specified. The developed process meets all criteria except possibly the surface finish requirement; occasionally, pitting occurs due to scale embedded during hot working. Detailed procedures covering composite plate manufacture are presented. (auth)

17738 (ANL-6343) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT (Argonne National Lab III) Mar 1961 Contract W-31-109-eng-38 86p

Research and development activities are reported on EBRW BORAX-V, ZPR-III, ZPR-VI ZPR-IX EBR-I and EBR-II Thermal- and fast-reactor safety studies are summarized Applied nuclear and reactor physics, reactor fuels development, reactor materials development, reactor components development heat engineering separations processes and advanced reactor development are discussed (M C G)

17749 (NDA-2147-3) A STATUS REPORT ON REACTIVITY COEFFICIENTS IN FAST REACTORS AND METHODS OF INVESTIGATING THEIR EFFECTS ON REACTOR STABILITY J. Agresta (Nuclear Development Corp of America, White Plains, N Y) Mar 1, 1961 Contract AT(30-1)-2303 (XIII) 19p

Methods for predicting the stability of fast reactors in the presence of both positive and negative reactivity coefficients are surveyed Experimental methods of determining reactor stability are reviewed excursion, transient, and oscillator Sources of reactivity coefficients are discussed for both zero power and nonzero power operation Representative reactivity coefficient values are presented, and positive coefficients due to Doppler effect, fuel bowing, and sodium coolant expansion are discussed. (D L C)

17760 CALCULATION OF PARAMETRIC INFORMATION FOR FAST REACTORS R E Reid Nuclear Energy, 158-60, 164(Apr 1961)

A first-order perturbation method is developed for reducing computer time of transport theory calculations, the method is designed for small fast reactors, with or without reflectors The method calculates the flux and adjoint simultaneously from their respective transport equations, and solves for the desired eigenvalue in an action that eliminates first-order errors The method is outlined for a SNAP-type or ZPR-III reactor (T F H)

17763 ON INFLUENCE OF ANISOTROPIC SCATTERING ON CRITICALITY OF A MULTIPLYING SPHERE Yu S Sigov Vestnik Moskov Univ, Ser III, Fiz Astron, 15: No 4, 79-85(June-Aug, 1960) (In Russian)

A generalized Carlson method is used for evaluating the influence of anisotropic scattering on the critical radius and spatial neutron flux density distribution in homogeneous, spherical fast reactors without reflectors The results derived by the transport equation and by the developed method are correlated An attempt was made to determine the applicability limits of transportation approximation (R V J)

17788 CONTAINMENT STUDIES OF THE ENRICO FERMI ATOMIC POWER PLANT W McGuire and G P Fisher (Cornell Univ, Ithaca, N Y) p 114-33 of "Nuclear Reactor Containment Buildings and Pressure Vessels" London, Butterworths, 1960

An engineering analysis is presented of the ultimate containment capability of the Fermi Fast Breeder Reactor plant The nuclear excursion considered is assumed to resemble in amount and rate of energy release the detonation of a 1,000 lb charge of conventional high explosive It is shown that while extensive internal disruption is to be anticipated, the outer containment vessel, as designed, will be capable of containing all the effects considered without deforming plastically (auth)

17811 NEUTRONIC REACTOR DESIGN TO REDUCE NEUTRON LOSS. F. T. Mills (to U. S. Atomic Energy Commission). U. S. Patent 2,982,709 May 2, 1961.

A nuclear reactor construction is described in which an unmoderated layer of the fissionable material is inserted between the moderated portion of the reactor core and the core container steel wall which is surrounded by successive layers of pure fertile material and fertile material having moderator The unmoderated layer of the fissionable material will insure that a greater portion of fast neutrons will pass through the steel wall than would thermal neutrons As the steel has a smaller capture cross-section for the fast neutrons, then greater numbers of the neutrons will pass into the blanket thereby increasing the over-all efficiency of the reactor

17825 (APDA-142) REMOVAL OF SODIUM FROM CORE SUBASSEMBLIES WITH WHITE OIL AND ULTRASONICS Z R Kanaan and C R Nash (Atomic Power Development Associates, Inc, Detroit) Mar 1961 46p

A method for removing sodium from fuel and blanket subassemblies that have been irradiated in the Enrico Fermi Atomic Power Plant reactor is described In this method, the subassemblies are flushed with white oil aided by ultrasonic agitation, and the sodium is carried away from the surfaces of the subassemblies by the oil Data gathered from experimental studies of the method are described; conceptual flow diagrams, materials balance sheets and equipment arrangements applicable to the Fermi plant are discussed, and sodium cleaning methods currently used are summarized (D L C)

17836 (PRDC-TR-42) MONTHLY TECHNICAL REPORT [ON APDA ACTIVITIES], DECEMBER 1960 (Power Reactor Development Co, Detroit) Contract AT(11-1)-476 22p

Research and development activities on the Fermi Fast Breeder Reactor are summarized in terms of core design, materials and metallurgy, nuclear engineering, health physics, mechanical handling, electrical and instrumentation systems, liquid metal and steam systems, and test operations Results of the airborne dust sampling and analysis program are given (M C G)

17862 RAPSODIE Georges Vendryes (Commissariat à l'Énergie Atomique, [Paris]) Energie nucléaire, 3 25-46 (Jan-Feb, 1961) (In French)

The fast-neutron high-flux reactor Rapsodie is greatly different from those which have been studied and constructed up to the present time in French nuclear centers It will use a concentrated fuel material (plutonium and enriched uranium) with sodium as heat transfer agent, and it will have no moderator The objective and principal characteristics of Rapsodie are described The reactor itself, the liquid circuits, installations for fuel element manipulation, and reactor control are then discussed (tr-auth)

17983 (ANL-6287) CHEMICAL ENGINEERING DIVISION SUMMARY REPORT, OCTOBER, NOVEMBER, DECEMBER 1960 (Argonne National Lab, Ill). Mar 1961 Contract W-31-109-eng-38 208p.

Chemical-metallurgical processing studies were made of pyrometallurgical development and research, and fuel processing facilities for EBR-II Fuel-cycle applications of fluidization and volatility techniques included laboratory investigations of fluoride volatility processes, engineering-scale development, and conversion of UF_6 to UO_2 Reactor safety studies consisted of metal oxidation and ignition kinetics, and metal-water reactions Reactor chemistry investigations were conducted to determine nuclear constants and suitable reactor decontamination methods Routine operations are summarized for the high-level gamma-irradiation facility and waste processing (B.O.G.)

18482 (BMI-1513) HIGH-TEMPERATURE NIOBIUM-BASE ALLOYS FOR SODIUM-COOLED REACTORS. John A. DeMastry, Frederick R. Shober, and Ronald F. Dickerson (Battelle Memorial Inst., Columbus, Ohio) Apr. 14, 1961 Contract W-7405-eng-92 30p

In work to develop fuel-element cladding material for the EBR-II, binary Nb alloys containing 2.37 wt % Cr, 3.34 wt % Zr, or 5.21 wt % V and a ternary alloy containing 1.56 wt % Zr and 1.09 wt % Cr were fabricated by forging at 550°C and rolling to sheet or by rod rolling at 1200°C in a stainless steel can. All four alloys were completely recrystallized after 90% cold work (reduction in thickness) and a 1-hr anneal at 1149°C. Tensile tests on the annealed materials at 650, 800, 1000, 1150, and 1315°C indicated that Cr, Zr, and V are potent strengtheners of Nb. The 2.37 wt % Cr alloy had a 0.2% offset yield strength of 38,000 psi at 1000°C, and the 5.21 wt % V alloy had a yield strength of 38,300 psi. At 1315°C, the 3.34 wt % Zr alloy had a 0.2% offset yield strength of 14,700 psi, the 1.56 wt % Zr-1.09 wt % Cr alloy had a yield strength of 13,500 psi at 1315°C. Stresses to produce creep rates of 0.001, 0.01, and 0.1% per hr were determined for the binary alloys at 650, 800, and 1000°C. The V alloy exhibited the greatest creep resistance, requiring stresses of 52,700, 40,000, and 15,500 psi to produce a creep rate of 0.001% per hr at 650, 800, and 1000°C, respectively (auth)

18567 IRRADIATION TESTING OF ENRICO FERMI PROTOTYPE FUEL PINS IN THE CP-5 REACTOR. M. A. Silliman, A. A. Shoudy, P. R. Huebotter, and W. G. Blessing (Atomic Power Development Associates Inc., Detroit) p. 17-32 of "Symposium on Radiation Effects and Radiation Dosimetry" Philadelphia, American Society for Testing Materials, 1960

Three Enrico Fermi prototype fuel pins were irradiated in CP-5 reactor to average burnups of 0.27, 0.58, and 0.85% of the total atoms fissioned in the U-10 wt % Mo fuel alloy. Through correlation of diameter measurements, temperature, and post-irradiation metallographic analysis, it is concluded that the radiation stability of the alloy is heavily dependent on the metallurgical condition of the material. The γ phase, stable above 1060°F, resists radiation more effectively than the $\alpha + \delta$ phases, which are thermally stable at temperatures less than 1045°F. Data from the CP-5 program suggest a more severe burnup limitation than was previously concluded. The disparity is believed to be due to the influence of the fission-rate parameter on the microstructure of the material during irradiation. Fission events, if they occur with sufficient frequency, maintain the γ phase at temperatures below 1060°F. If the critical rate is not attained, the alloy will transform thermally in accordance with the U-Mo equilibrium diagram and the macroscopic temperature of the material (auth)

19064 (ANL-6332) PRELIMINARY DESIGN OF A BASIC RADIATION EFFECTS REACTOR (BRER). D. R. MacFarlane, R. R. Rohde, B. Toppel, I. Charak, and H. Unger (Argonne National Lab., Ill.) Mar. 1961 Contract W-31-109-Eng-38 61p

The Basic Radiation Effects Reactor (BRER) is a small fast core surrounded by a segmented radial reflector. The NaK-cooled fast core operates at a thermal power of 1 Mw, with all the reactor heat being rejected to the atmosphere through a secondary heat exchange system. The secondary heat exchange system is another NaK loop which dissipates heat to the atmosphere by means of an air-blast cooler. The reactor core is composed of small-diameter rods of uranium-zirconium alloy, arranged in a close-packed triangular pattern. The maximum core loading is approximately 60 kg of U^{235} . Reactor control is effected by moving control rods in the reflector region immediately adjacent to the core. Reactor instrumentation and fuel handling are similar to other heterogeneous reactor systems. Relatively

large volumes for experiments are available in the large radial reflector surrounding the core. The physics of the BRER system was investigated, using a 15-group set of cross sections, for a series of reflector materials. The materials studied were lead, aluminum, iron, zirconium, depleted uranium, and natural uranium. Based on the criterion of producing two widely spaced and relatively sharply peaked neutron spectra, these preliminary calculations indicate that a major portion of the reflector would be lead, with an aluminum region starting at some intermediate point and extending to the outer edge of the reflector. (auth)

19515 (TID-6881) DESCRIPTION OF INTERMEDIATE HEAT EXCHANGER AND STEAM GENERATOR SELECTIONS. Final Report. R. W. Schroeder and M. A. Chionchio. Feb. 25, 1959. Includes Supplement MECHANICAL DESIGN BASIS FOR HEAT EXCHANGER COMPONENTS FOR NUCLEAR POWER PLANTS. (Griscom-Russell Co., Massillon, Ohio). Oct. 30, 1958. Contract AT(11-1)-664. 151p.

A discussion is given of the development studies of heat exchangers and steam generators for sodium-cooled reactor systems. Design specifications are included for offset-tube and U-tube intermediate heat exchangers, and involute-tube and offset-tube steam generators with removable bundles. Estimated costs for 70-Mw designs are included. Recommendations for the proposed development program are described (B.O.G.)

19523 FERMI AND HALLAM STEAM GENERATORS. Frank Boni and Philip S. Otten (Griscom-Russell Co., Massillon, Ohio) Nucleonics, 19 No. 6, 58-61 (June 1961)

Design characteristics of the heat exchangers used in the Fermi Fast Breeder Power Reactor and the Hallam Power Reactor are outlined. Particular attention is devoted to thermal stress reduction and leak handling measures, and the materials used in the steam generators of the two reactors are described (T.F.H.)

19579 INSTRUMENTS IN THE FAST REACTOR. Atom, No. 54, 10-11 (Apr. 1961).

Nuclear instrumentation for the Dounreay fast reactor is described. Methods for measuring the fast neutron flux, the liquid metal coolant flow, the inlet and outlet temperatures, and the coolant oxide content are outlined (T.F.H.)

19815 MANUFACTURE OF FUEL ELEMENTS FOR BREEDER REACTORS. (to DEGUSSA). Belgian Patent 575,379. Priority date, Feb. 4, 1958.

The process consists in sintering uranium or plutonium carbide, on their own, or mixed with ZrC, NbC, TaC, in order to obtain a solid piece having 15 to 30% porosity. This is then impregnated, under vacuum, at 3400°F with thorium metal. The final product can be irradiated in a reactor and reach temperatures up to 2600°F. (EURATOM)

20343 REACTIVITY EFFECTS OF PROTACTINIUM-233 BUILDUP IN U^{233} FAST BREEDER REACTORS

Arthur J. Goldman (Nuclear Development Corp. of America, White Plains, N.Y.) Nuclear Sci. and Eng., 10 91-2 (May 1961)

The effects of Pa^{233} production in U^{233} fast breeder reactors are studied. The Pa^{233} is formed by the reaction $Th^{232}(n,\gamma)Th^{233} \rightarrow \beta^- + Pa^{233}$, it decays by the reaction $Pa^{233} \rightarrow \beta^- + U^{233}$. In a reactor using U-Th fuel elements, the reactivity during shutdown increases because of the 27-day decay of Pa^{233} . An example is given. The excess reactivity effects are functions of bred fuel atom distributions and of fuel reactivity worth (T.F.H.)

20375 (ANL-6355) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT. (Argonne National Lab., Ill.). Apr. 1961. Contract W-31-109-eng-38. 73p.

Progress on reactor programs and in general engineering research and development programs is summarized. Research and development are reported on water-cooled reactors including EBWR and Borax-V, sodium-cooled reactors including ZPR-III, IV, and IX, Juggernaut, and EBR-I and II. Other work included a review of fast reactor technology, and studies on nuclear superheat, thermal and fast reactor safety, and reactor physics. Effort was also devoted to reactor materials and fuels development, heat engineering, separation processes and advanced reactor concepts. (J.R.D.)

20428 (PRDC-TR-43) MONTHLY TECHNICAL REPORT [ON APDA ACTIVITIES], JANUARY 1961. (Power Reactor Development Co., Detroit). Contract AT(11-1)-476. 20p.

Research and development activities on the Fermi Fast Breeder Reactor are summarized in terms of core design, materials and metallurgy, mechanical handling, electrical and instrumentation systems, liquid metal and steam-systems, and test operations. (B.O.G.)

21735 STRUCTURAL RESPONSE TO OSCILLATING COOLANT TEMPERATURES IN EBR-I, MARK II. R. G. Matlock and R. R. Smith (Argonne National Lab., Ill.). Trans. Am. Nuclear Soc., 4: No. 1, 12(June 1961).

21740 CALCULATION OF DOPPLER REACTIVITY EFFECT IN A $\text{PuO}_2\text{-UO}_2$ FUELED FAST REACTOR. P. Greebler and B. A. Hutchins (General Electric Co., San Jose, Calif.). Trans. Am. Nuclear Soc., 4: No. 1, 17(June 1961).

21743 THE EFFECT OF FAST REACTOR CALCULATIONS OF FLUX DEPRESSION IN SCATTERING RESONANCES. R. B. Nicholson, P. F. Zweifel, and J. H. Ferziger (Atomic Power Development Assn., Inc., Detroit). Trans. Am. Nuclear Soc., 4: No. 1, 19(June 1961).

21770 A MECHANISM EXPLAINING THE INSTABILITY OF EBR-I, MARK II. R. R. Smith and R. G. Matlock (Argonne National Lab., Idaho Falls, Idaho). Trans. Am. Nuclear Soc., 4: No. 1, 71-2(June 1961).

21791 ENGINEERING CONSIDERATIONS FOR REMOTE REFABRICATION OF EBR-II FUEL ELEMENTS. A. B. Shuck and J. E. Ayer (Argonne National Lab., Ill.). Trans. Am. Nuclear Soc., 4: No. 1, 117-18(June 1961).

21792 EQUIPMENT FOR REMOTE INJECTION CASTING OF EBR-II FUEL. H. F. Jelinek and G. M. Iverson (Argonne National Lab., Ill.). Trans. Am. Nuclear Soc., 4: No. 1, 118-19(June 1961).

21793 EQUIPMENT FOR REMOTE DEMOLDING, SIZING, AND INSPECTION OF EBR-II CAST FUEL PINS. N. J. Carson, Jr. and S. B. Brak (Argonne National Lab., Ill.). Trans. Am. Nuclear Soc., 4: No. 1, 119-20(June 1961).

21794 LEAK TESTING OF EBR-II FUEL RODS. A. P. Grunwald (Argonne National Lab., Ill.). Trans. Am. Nuclear Soc., 4: No. 1, 120(June 1961).

21795 ASSEMBLY, SODIUM BONDING, AND BOND TESTING EBR-II FUEL RODS. T. C. Cameron and N. F. Hessler (Argonne National Lab., Ill.). Trans. Am. Nuclear Soc., 4: No. 1, 120-2(June 1961).

21813 IMPROVEMENTS IN OR RELATING TO NUCLEAR REACTORS. John Tatlock and John Alexander Forbes Glass (to United Kingdom Atomic Energy Authority). British Patent 867,969. May 10, 1961.

A support for a fast reactor core of fuel elements with means allowing the insertion and withdrawal of neutron generating material for control purposes is described. It is comprised of a core of vertically oriented fuel elements having groups of fuel elements supported in carriers which are retained against gravity by association at their lower ends with operating mechanisms for controlled movement of carriers together with their groups of fuel elements into and out from the core and the carriers together with their groups of fuel elements being free to fall from the core under gravity by a coupling releasable in the mechanisms. (N.W.R.)

21839 (NAA-SR-5190) A COMPARISON OF TYPICAL FUEL CYCLE COSTS FOR THERMAL AND NONTHERMAL SODIUM-COOLED REACTORS. K. W. Foster (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). May 30, 1961. Contract AT(11-1)-GEN-8. 51p.

Preliminary nuclear, heat transfer, and fuel cycle cost data were prepared for thermal and nonthermal sodium-cooled reactors in order to determine the reactor type best suited for demonstrating economic nuclear power. UC and PuC-UC fuels are considered. At a burnup of 20,000 Mwd/MT, the various reactor types are ranked from least to most expensive in the following order: (1) paste fast, (2) solid-rod thermal, (3) solid-rod fast-thermal coupled, and (4) solid-rod fast. For a burnup of 50,000 Mwd/MT, (4) will be more economical than either (2) or (3). Since further development of (1) is required, the solid-rod thermal reactor with a fuel cycle cost of ~1.7 mills/edwh is recommended as the best reactor type for demonstration purposes. (D.L.C.)

21887 FEASIBILITY STUDY OF THE URANIUM-233--THORIUM FAST BREEDER REACTOR SYSTEM. A. J. Goldman (Nuclear Development Corp. of America, White Plains, N. Y.). Trans. Am. Nuclear Soc., 4: No. 1, 33(June 1961).

21904 RADIOACTIVE WASTE HANDLING IN THE NUCLEAR POWER INDUSTRY. A Report of the Technical Appraisal Task Force on Nuclear Power to the Board of Directors of the Edison Electric Institute. EEI Publication No. 60-46. New York, Edison Electric Institute, 1960. 93p. \$5.00.

Problems and prospects in the handling of radioactive wastes from reactors, fuel processing plants, and nuclear weapons are studied. Outlines are given of the primary sources of radioactive materials and the types, effects, and permissible doses of radiation. Methods for containment or dispersal of gaseous, liquid, and solid wastes are described. Economics, regulations, standards, and insurance aspects of waste handling are studied. Waste handling at the six large U. S. power reactors—Shippingport, Yankee, Indian Point, Dresden, Enrico Fermi, and Hallam—are discussed. Normal and abnormal wastes, wastes from accidents, and utilization of wastes are investigated. (T.F.H.)

22692 (ANL-6330) ANNUAL REPORT FOR 1960, METALLURGY DIVISION. (Argonne National Lab., Ill.). Contract W-31-109-eng-38. 273p.

Development and Fabrication Programs for EBR-II. The thermal conductivity of U-5 wt % Fs (fissium)—2.25 wt % Zr alloy was measured at 0 to 800°C. The heat enthalpy of U-5 wt % Fs is greater than that for U at temperatures up to 350°C, above 350°C, the enthalpy of U is higher. Phase

studies of U-Pu-Fs alloys with and without Zr additions are reported. Metallographic examinations were made on specimens taken from the top, center, and bottom of injection-cast U-20 wt % Pu-10 wt % Fs fuel pins. Diffusion penetration by U-20 wt % Pu-10 wt % Fs was measured for the following potential cladding materials: Ti, Mo, V, Ta, Y, Nb, Nb-5.3 wt % V, Zr, Zircaloy-2 and 347 and 430 stainless steel. Irradiation behavior of U-Fs and U-Pu-Fs alloys is discussed. Thermal cycling tests of EBR-II blanket slugs indicate the cycling behavior to be unaffected by differing cooling rates from the beta phase. Methods and equipment used in refabricating irradiated fuel into elements suitable for return to EBR-II are described in detail, and production results for both fuel pins and blanket rods are given. EBR-II steam generator fabrication is discussed. EBR-I Core IV Fuel Elements. Casting methods were evaluated for fabrication of Pu-1 wt % Al alloys. The following properties were measured for Pu-1 wt % Al: (1) thermal conductivity at 100 to 600°C, (2) hot deformation or slumping tendency at 7 to 60 psi load, and (3) hardness. Densities of five Pu-Al alloys of composition ranging from 0.26 to 3.06 wt % Al were determined. Fabrication of Zircaloy jacket components, uranium blanket slugs, and stainless steel hardware for EBR-I Core IV is described. EBWR Core 1-A Spike Fuel Elements. A ZrO_2 -CaO- UO_2 solid solution ceramic was fabricated in the form of pellets to permit EBWR operation at 100 Mw(t). Fabrication of Zircaloy-2 hardware and fuel element assemblies for the spike elements is described. Thermal cycling test results are presented for Al- U_3O_8 fuel rods. Borax V Boiling and Superheater Fuel Elements. Fabrication of Borax V boiling and superheater fuel elements is discussed in some detail. Ceramic Fuel Materials. Direct reaction of UO_2 and C as a means of producing UC was investigated in argon atmosphere. Reactions of UO_2 and C powders yielded a heterogeneous mixture with UO_2 predominating, while UO_2 reacts with petroleum pitch in alcohol to give UC with not more than 10% UO_2 . Metallographic results of UC are reported, a U_2C_3 surface layer was observed. PuC pellets were arc melted, and the following properties were studied for the compositions of 42.1 to 59.6 at.% C: density, microhardness, heat treatment behavior, and thermal expansion. The densification of uranium-thoria solid solutions during both air and H_2 sintering was studied, and the density changes of air-sintered U_3O_8 - ThO_2 were found to be influenced greatly by the composition, while H_2 sintering yielded the highest sintered densities. Weight losses and stoichiometry of air-sintered U_3O_8 - ThO_2 compositions were also measured. La_2O_3 - U_3O_8 mixtures were calcined and milled to give solid solutions which were quite stable in air. However, solubility limits and lattice parameters measured under three different heat treatments indicate stoichiometry differences. Deviations from stoichiometric $\text{MO}_{0.9}$ were measured for H_2 - and air-sintered La_2O_3 - U_3O_8 compositions. Three types of glasses were tested for possible use as a single-phase fuel material with U_3O_8 , one type, Na_2O - K_2O - SiO_2 , did not devitrify when mixed with various amounts of U_3O_8 . The sintered densities of various MgO-fueled glass mixtures were determined. Isothermal heat treatment of a 30% UO_2 - Na_2O -CaO- SiO_2 glass in a 304-L stainless steel capsule at 800°C for 650 hr gave evidence of a slight reaction approaching a bond. Sintered 85 wt % boron carbide-glass mixtures fabricated in the shape of pellets and rods were found to have good mechanical strength. Sintering studies of US and ThS were carried out and their electrical resistivities measured. Irradiation Evaluation of Fuel, Control, and Structural Materials. The effects of cladding and internal venting on the swelling characteristics of clad U-2 wt % Zr specimens were studied at burnups of 1.4 to 2.1 at.% and temperatures of 620 to 1260°C. The results indicate that vent holes combined with cladding restraint can extend the application of metallic fuels to higher temperatures. Dimensional stability of Th-5 wt % Pu and

Th-10 wt % Pu after 1.4 and 2.3 at.% burnups was found to be excellent. Metallographic examinations of irradiated Al alloy-clad, Pb-bonded ThO_2 - UO_2 pellets after cladding failures are reported. Irradiation testing of Th-U alloys containing 10 to 28 wt % U showed good irradiation stability up to 20 wt % U, higher U contents were less stable. The program for development of control materials containing lanthanides is briefly described. A scheme is outlined for monitoring the irradiation effects in the Borax V pressure vessels with specimens of the same material as the vessel. Development of instrumented high-temperature irradiation capsules for use in the CP-5 reactor is described. Examination of Full-Scale Fuel Elements. The results and conclusions of a metallurgical evaluation of failed Borax IV fuel elements are discussed. Since testing of EBWR Core IA fuel elements indicated possible water channel blocking at 100 Mw, a device was developed for monitoring the water channel gaps. Corrosion-Resistant Fuel, Jacketing, and Structural Materials. Corrosion rates of U-3 wt % Ti with small amounts of Nb and Ru were measured up to 234 days in water. Corrosion results indicate that iodide thorium has better corrosion resistance than Ca-reduced thorium. Corrosion of Th-U alloys with U contents ranging from 5 to 40 wt % was studied in 150°C water, and Th-U-Zr alloys were also investigated. A large number of Th alloys were evaluated with respect to aqueous corrosion, and Al, Y, and Zr were found to be beneficial additives. Aqueous corrosion study results are presented for a low Si-Al alloy A288 which indicate the alloy to be an improved Al alloy. Corrosion of a number of binary Al alloys was studied, and Co and Rh were found to be beneficial additives. Strengthening of Al alloys was studied. Corrosion of X8001 Al by H_3PO_4 and of Al-Ni-Fe by superheated steam was investigated. Studies of Zr alloy corrosion by superheated steam and absorption of H_2 are reported. Corrosion results of Fe containing various O_2 contents in water are summarized, higher O_2 contents appear to be beneficial. Corrosion by superheated steam was studied for the following Fe alloys: 304 and 347 stainless steels, Martin alloys 258, 259, and 261, and Croloy 2 1/4 and 5. 18-8 stainless steel and X8001-coated low-C steel were tested in dilute H_3PO_4 solutions. Mercury corrosion tests at 370°C and 100 psia were made on Ti, Ti alloy A110-T, and 304 stainless steel. Non-destructive Testing Developments. Various nondestructive techniques developed or being evaluated for use with reactor and nuclear materials are described: ultrasonics, eddy currents, neutron radiography, and gamma radiography. Fast Reactor Safety Program. Various joining problems encountered in thermocouple fuel element fabrication for TREAT are described. Dummy fuel rods and slot liner-capsule assembly for TREAT are described. Fabrication of ZrO_2 -CaO- UO_2 and UO_2 -304 ELC stainless steel fuel samples is discussed. Preparation of High-Purity Materials. High-purity Pu buttons produced by electrolysis with PuF_4 dissolved in KCl - LiCl were analyzed for their impurities and the results given. Densities of cast Pu of varying impurity content were determined. Cs_2UCl_6 was used in electrowinning U and U yields up to 73% were obtained, however, Cs_2PuCl_6 was found to be poor for either electrowinning or electrolysis of Pu. Texture Studies. High-purity rolled U sheet prepared in such a way that the starting grains are uniform and small was studied, recrystallization at low and moderate temperatures was found to result in formation of very fine grains and then discontinuous growth of larger grains. X-ray pole figures are presented for as-rolled U and for annealed-rolled U. Recrystallization at low temperatures was found to result in a preferred orientation different from that of as-rolled material. Recrystallization textures in single α zirconium crystals was studied under compressive deformation, and deformation mechanisms are derived for different crystals. Uranium and Plutonium Alloys. The temperature of forma-

tion of U_2Ru_3 is estimated to be near 1000°C. The structure of U_2Ru was derived from x-ray work. A thermal conductivity vs temperature plot is presented for various U-Fs alloys. Some results of a phase study of U-Pu-C are reported. A technique was developed for obtaining the lattice parameters of α uranium phases by the Debye-Scherrer method, and results are presented for pure α uranium.

Alloying properties The saturation magnetization of Fe-V alloys was measured with the alloys in the form of the metastable bcc α phase and after a series of anneals at 625°C which gave a σ phase. The results indicate CsCl-type ordering. Lattice contractions are plotted against radius ratios for CsCl-type compounds between Cu, Ag, Au and the lanthanides. Crystal structures were determined for Laves phases and for binary AB_3 compounds. The c/a ratios of hexagonal $TiNi_3$ -type APd_3 compounds were investigated as a function of the size factor. The occurrence of epsilon phases in various ternary systems is discussed. Intermetallic compounds which exist in binary systems of Nb with transition elements were investigated. A_2B phases are shown to exhibit either Ti_2Ni or Al_2Cu type phases, depending on the atomic size factor. Si_2Mo -type compounds were also studied. The miscibility of Laves-type phases in 51 ternary systems was determined, most were found to be completely miscible. The dissolution rates of Ti, V, Cr, Fe, Co, and Ni in liquid Sn and Al were determined at 600 to 800°C.

X-ray and Neutron Diffraction Studies A stepping mechanism to be used with a horizontal neutron spectrometer is described. Available data on the structure of α uranium are compared, the variance of the parameter y with temperature was measured. The crystal structures of $CuTi_2$ and Ti_2Ni were determined. Preliminary diffraction measurements on the structure of $Na_2[Ru(NO_2)_4(NO)(OH)] \cdot 2H_2O$ are reported. Magnetic structures of Mn-Co, Mn-Co-C, and Mn-Cu were derived from diffraction patterns made on null matrices. The results of an investigation of the neutron coherent scattering amplitudes of Ti isotopes are summarized. Neutron scattering amplitudes were determined for K^{39} and Rb^{85} in the form of chlorides. Ti washers, reacted with D_2 gas and then depleted to various degrees by an arc discharge, were examined to determine the state of D atoms in Ti. It was found that D reacts with Ti to form a metal deuteride with a structure different from that of the metal.

Metal Physics The temperature dependence of the elastic moduli of α uranium crystals was measured down to 1.1°K for C_{11} and to 35°K for C_{22} . Both sets of measurements indicate a sharp transition in the physical state at a temperature between 41 and 43°K. Revised calculations of the theoretical elastic moduli of α uranium are presented and discussed. The five elastic moduli of α zirconium crystals were measured at 19.7°C and elastic parameters calculated therefrom. Work on diffusion in U is summarized. Diffusion of Cr in β uranium was found to behave differently from self-diffusion. A theoretical study of the mechanisms of twinning in Zr is presented, and ZrH habit planes in Zr were identified.

Corrosion Research The effect of applied current on the solution potential of Al and Co in boiling distilled water is illustrated by graphs. Corrosion of a large number of ternary Zr alloys was studied in dry O_2 and steam. Emf measurements of the growing ZrO_2 scale on various Zr alloys indicate that the alloys can be grouped into three classes according to their emf behavior. The implications of the corrosion studies of Zr are discussed. The Al content of the corrosion product lost from 1100 Al in 70°C water was determined as a function of time, and the reaction chemistry is discussed. The amount of Al corrosion in water was also followed over a one-year period. pH measurements at a point close to the Al surface in 50°C water are reported as a function of time. Relative corrosion pitting of four Al alloys in $CuSO_4$ solution at 50°C was studied. A mechanism is proposed for the corrosion of Al alloys by superheated steam. The results of an electron

microscopic study of corrosion films formed on Al alloys and Fe by high-temperature water are presented.

Irradiation Effects A detailed investigation was made on the swelling behavior of U and U alloys after postirradiation annealing, some of the variables studied were annealing temperature, annealing time, and burnup. Evaporated Au films irradiated with fission fragments from natural U were studied by electron microscopy, results indicate that one fission fragment produces ~10 damage spots along its path in Au. The anisotropy of fission fragments escaping from U crystals into Ilford K-2 emulsions was determined to be less than 15%.

Ceramic Materials Research The sintering rates of single Al_2O_3 crystals were studied and a diffusion coefficient derived $D = 5 \times 10^8 \sqrt{p_{O_2}} \exp(-150,000/RT)$. Since the activation energy agrees well with that measured for O_2 diffusion, the rate-controlling mechanism in Al_2O_3 sintering appears to be O_2 diffusion. A sphere grinder and furnace for sintering studies are described. A detailed study of reduction of uranium oxides of compositions between U_3O_8 and UO_2 by CO and H_2 was made. The reaction rates were found to obey the following equations for H_2 , $dP/dt \propto g^{3/2} P \exp(-33,000/RT)$, and for CO, $dP/dt \propto g^{3/2} P^{1/2} \exp(-4,000/RT)$, where P = pressure and g = sample weight. The rates are relatively independent of the over-all oxide composition. (D.L.C.)

23067 (APDA-143) SUMMARY OF THE APDA FUEL DEVELOPMENT PROGRAMS. W. G. Blessing, J. S. Busch, J. G. Duffy, R. J. Hennig, W. H. Jens, F. W. Knight, E. C. Kovacic, D. O. Leaser, G. L. O'Neill, A. A. Shoudy, and R. G. Rateick (Atomic Power Development Associates, Inc., Detroit). Apr. 1961. 187p.

A summary is presented of the development status, through 1960, of work done on: Core A, the U-10 wt. % Mo fuel and U-3 wt. % Mo blanket, and Core B, the UO_2 -SS cermet and the UO_2 axial blanket for the Fermi Fast Breeder Reactor; advanced fast reactor fuels; and paste fuels. (B.O.G.)

23068 (DEG-Report-96) THE MULTI-GROUP ALBEDO METHOD. R. T. Ackroyd and J. M. Ball (United Kingdom Atomic Energy Authority. Development and Engineering Group, Risley, Lancs, England). May 15, 1961. 12p.

A multi-group version is given of the albedo method for calculating the critical sizes of fast cores with moderating reflectors. Fairly accurate results for critical sizes were obtained for bare spheres and cylinders of U^{235} and Pu^{239} without the need to adjust the core cross-sections, as in the simple albedo method which uses one group in the core and two groups in the reflector. The multi-group albedo method, because of its simplicity, is suitable for calculations on the interaction of arrays of fissile cores. The method can be applied also to cores of awkward shape without any difficulty. (auth)

23116 (ANL-6374) REACTOR DEVELOPMENT PROGRAM, PROGRESS REPORT, MAY 1961. (Argonne National Lab., Ill.). June 15, 1961. Contract W-31-109-eng-38. 61p.

General research and development on water-cooled and sodium-cooled reactors are reported along with specific developments on EBWR, BORAX-V, EBR-I, and EBR-II. Thermal and fast reactor safety studies are summarized in terms of fuel-coolant chemical reactions, kinetics of oxidation and ignition of reactor materials, core meltdown studies, and a sodium vapor pressure furnace. Evaluations were made of improved fast reactors for central station power and of a 50-Mwe Prototype Organic Power Reactor (POPR). Developments in instruments, reactor fuels and materials, reactor components, heat engineering, separa-

tions processes, and advanced reactors are discussed. (M.C.G.)

23145 COUPLED FAST-THERMAL POWER BREEDER REACTOR. Robert Avery (to U. S. Atomic Energy Commission). U. S. Patent 2,992,982. July 18, 1961.

A nuclear reactor having a region operating predominantly on fast neutrons and another region operating predominantly on slow neutrons is described. The fast region is a plutonium core and the slow region is a natural uranium blanket around the core. Both of these regions are free of moderator. A moderating reflector surrounds the uranium blanket. The moderating material and thickness of the reflector are selected so that fissions in the uranium blanket make a substantial contribution to the reactivity of the reactor.

23146 FAST NEUTRON REACTOR. Harry Soodak and Eugene P. Wigner (to U. S. Atomic Energy Commission). U. S. Patent 2,993,850. July 25, 1961.

A reactor comprising fissionable material in concentration sufficiently high so that the average neutron energy within the reactor is at least 25,000 ev is described. A natural uranium blanket surrounds the reactor, and a moderating reflector surrounds the blanket. The blanket is thick enough to substantially eliminate flow of neutrons from the reflector.

23766 (NDA-2147-5) FAST REACTOR SAFETY. Quarterly Progress Report for the Period Ending December 31, 1960. J. Agresta and F. Beers (Nuclear Development Corp. of America, White Plains, N. Y.) Mar. 15, 1961. Contract AT(30-1)-2303. 21p.

Investigations were made of fast reactor safety. Sodium two-phase flow regimes were charted to show the different flow pattern regions as a function of quality and total mass velocity. The pressure losses resulting from frictional resistance for two-phase sodium flow were obtained. Vapor-volume fraction and slip and pressure drops resulting from momentum changes were investigated. A systematic study of the effects on reactor stability of both negative and positive power coefficients which may be prompt or delayed was begun. The effect on stability of delayed negative power coefficients of both the conduction and convection types was determined. (M C G)

23908 FUEL AND BLANKET ELEMENTS FOR THE EXPERIMENTAL BREEDER REACTORS. R. Macherey (Argonne National Lab., Ill.). p.467-94 of "Fuel Element Fabrication with Special Emphasis on Cladding Materials. Volume 1." London and New York, Academic Press, 1961. (In English)

The fabrication, jacketing, assembly, welding, and bonding of the Mark I, II, and III loadings of Experimental Breeder Reactor I (EBR-I) and the Mark I loading of EBR-II are described. Inspection techniques employed in the evaluation of the elements, reactor performance of the EBR-I fuels, and irradiation results from test elements are also described. In addition, the trends in future high burnup fuel elements for EBR-II are discussed. (auth)

24590 (PRDC-TR-44) MONTHLY TECHNICAL REPORT, REPORT PERIOD—FEBRUARY 1961. (Power Reactor Development Co., Detroit). Contract AT(11-1)-476. 14p.

Research and development on the Fermi Fast Breeder Reactor are summarized in sections concerned with core design, materials and metallurgy, nuclear engineering, mechanical handling, instrumentation, liquid metal and steam systems, and test operations. (J.R.D.)

24592 (TID-3556(Rev.1)) SELECTED REACTORS OF THE POWER REACTOR DEMONSTRATION PROGRAM. A Literature Search. James M. Jacobs; comp. (Office of Technical Information Extension, AEC.). June 1961. 52p.

A bibliography of 552 reports is presented on eleven reactors of the Power Reactor Demonstration Program. The reactors are the Big Rock Point, Elk River, Enrico Fermi, Florida, Hallam, Parr Shoals, Pathfinder, Peach Bottom, Piqua, Puerto Rico, and Yankee power reactor. (D.L.C.)

25603 ACHIEVING HIGH BURN-UP IN FAST REACTORS. L. R. Blake (Dounreay Experimental Reactor Establishment, Caithness, Scotland). J. Nuclear Energy, Pts A & B. Reactor Sci and Technol, 14. 31-48 (Apr 1961). (In English)

A theory is proposed that relates irradiation swelling in metal fuel with burn-up. The theory is used to predict fuel element behavior under various fast reactor conditions. Agreement of the theory with known behavior is sufficient to substantiate the general conclusion that fission-product gas accumulation is chiefly responsible for swelling. It appears to be possible only to reduce and accommodate such swelling. Based on this principle, a fast reactor fuel element is suggested using unalloyed uranium, with substantial void-space, in a strong can. This element should be capable of 4 per cent average burn-up at 700°C center temperature under practical fast reactor conditions, which should be sufficient to ensure economic power. With oxide fuel it appears that, in order to satisfy the essential safety and economic requirements in a fast reactor, it is also necessary to incorporate considerable voidage in the fuel and to employ a strong can, safety requirements could also lead to double-canning. Such an arrangement could possibly attain 10 per cent burn-up, which is comparable to 4 per cent with metal fuel. A cermet of UO_2 - PuO_2 dispersed in uranium metal, in a strong can and with void space in the fuel, appears to be a suitable way of combining the good properties of both oxide and uranium fuel, and could result in a high burn-up fuel for fast reactors, though its behavior is more uncertain to predict. (auth)

25621 (ANL-6387) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT [FOR] JUNE 1961. (Argonne National Lab., Ill.). Contract W-31-109-eng-38. July 15, 1961. 69p.

Water Cooled Reactors. The irradiation performance of a prototype EBWR Core I fuel plate in a high-pressure water loop is reported. Preliminary data on decontamination of 304 stainless steel by oxalic acid-hydrogen peroxide are presented. EBWR cores IA and II and EBWR operation are discussed. Installation and testing of BORAX V reactor components are reported. Sodium Cooled Reactors. Fission and reactivity measurements are reported for ZPR-III fuel assemblies 36 and 22. Progress on the ZPR-VI and ZPR-IX is outlined. Fabrication of EBR-I Mark IV core, construction of EBR-II, and development of EBR-II components are discussed in detail. Reactor Safety. Studies of the reactor of U and Al with steam and water are described. Results are presented for TREAT reactor transients performed on fuel pins of UO_2 clad with 304 stainless steel and submerged in water. Preliminary data are presented for the ignition temperature of Zr. Fast reactor safety studies under way are outlined. Nuclear Technology. Neutron differential elastic scattering cross sections are presented for Th and Fe at 0.5 to 1.6 Mev. Reactivity measurements in the ATSR are reported. Operating conditions for ZPR-VII critical conversion experiments are outlined. Results of corrosion of Al powder tubing in water at 290 and 360°C and of Zr alloys in steam at 540°C are discussed. Heat content tables are presented for pure U and uranium-5 wt % fissium at 0 to 1200°C,

and heats of transformation and fusion are also given. Some results of studies of the properties of PuC are discussed. Nondestructive testing methods are discussed. Various projects in reactor materials and components development and heat engineering are described. A large number of studies was conducted on separation of uranium from fuel elements, especially by fluorination. Reactions of PuF_6 with various inorganic solids were studied, and the G value for radioinduced PuF_4 production from PuF_6 was determined. Operating results are presented for the two-step fluid bed process in which UF_6 was converted into UO_2F_2 by steam hydrolysis and the UO_2F_2 subsequently reduced to UO_2 . Pure U_3Si_2 was obtained by reaction of stoichiometric amounts of U and Si at 800°C . ThO_2 was found to be reduced to the metal in 2 hr at 750°C by 5 wt % Mg-Zn in a salt flux. Equations are given for the solubilities of Sc and Fe in liquid Cd. Work on U-Zn alloys and on distribution of U between liquid Pb and liquid Zn are discussed. Some advanced reactor concepts are discussed. (D.L.C.)

25644 (NAA-SR-Memo-5996) OPTIMIZATION STUDIES ON PASTE-FUELED FAST REACTORS. J. M. Zetterbaum and T. W. Kerlin (Atomics International, Div. of North American Aviation, Inc., Canoga Park, Calif.). Dec. 28, 1960. 21p.

The reference design is an unmoderated, sodium-cooled reactor using a paste fuel of uranium monocarbide in sodium. The core is a cylinder 5 ft in diameter and 5 ft in height. An 18-in. thick breeding blanket surrounds the core and an 18-in. thick graphite reflector surrounds the blanket. Various changes were made in the reference core to uncover any possible modifications for cost reductions and to evaluate the consequences of certain design modifications which might occur. Cases were studied for variations in fuel volume fraction in the core from 0.2 to 0.6, fertile material volume fraction in the blanket from 0.2 to 0.6, blanket thickness 3 in. to 24 in.; fuel materials of UC, U metal, UO_2 , PuC-UC, Pu-U metal, and PuO_2 - UO_2 , and liquid carrier in the paste of Na, Sn, or Pb. (auth)

25647 (NP-10458) ENRICO FERMI ATOMIC POWER PLANT REVISED LICENSE APPLICATION. PART A. GENERAL INFORMATION AND REQUEST FOR LICENSE. PART B. TECHNICAL INFORMATION AND HAZARDS SUMMARY REPORT, VOLS. 1 THRU 6. (Power Reactor Development Co., Detroit). July 1961. 209p.

Part A contains information regarding the Power Reactor Development Company, which operates the reactor, other information required by the AEC, and the specific license application. Part B consists of 6 volumes which comprise studies of the reactor and plant design, normal operations, and commissioning operations; the system verification programs; and the reactor site and environment considerations. (T.F.H.)

25734 (ANL-6333) CHEMICAL ENGINEERING DIVISION SUMMARY REPORT, JANUARY, FEBRUARY, MARCH 1961. (Argonne National Lab., Ill.). Contract W-31-109-eng-38. 250p.

Pyrometallurgical processes are being developed for recovering fissionable and fertile materials from short-cooled reactor fuels. The second laboratory demonstration of the melt-refining process with highly irradiated EBR-II-type fuel pins was completed. A 392-g charge of U-5% fission fuel pins irradiated to an estimated burnup of 0.4 total at.% and cooled 28 days was melt refined for three hours at 1400°C . Data were not obtained on the behavior of fission products. The effect of N concentration on the nitridation rates of unirradiated U-fission alloys in Ar-N atmospheres was determined. Experiments on the storage of fuel pins at 350°C in Ar atmospheres showed that the

presence of 5% N lowered product yields only slightly during subsequent melt-refining operations. Supplementary pouring techniques, such as the use of probes and mashers designed to break crusts over the melts, are moderately effective, but are a less desirable solution to the problem of maintaining high yields than the elimination of contaminants in the Ar atmosphere. A liquid metal process is under development for recovery of the fissionable material contained in melt refining crucible skulls produced in the EBR-II fuel cycle. Information obtained in separate studies of the individual process steps is listed. A systematic study is underway to ascertain the influence of atomic size, metallic valence, and electronic configuration on the coprecipitation of various metallic elements with the Cd-Ce intermetallic phase CeCd_{11} . Values for the coprecipitation coefficient λ , defined by the equation $\log(\text{tracer in solution}/\text{total tracer}) = \lambda \log(\text{carrier in solution}/\text{total carrier})$, were determined for Na, Li, K, Y, Ba, $\lambda = 0$; La, $\lambda = 1.49$; Th, $\lambda = 1.08$; Pr, $\lambda = 0.63$; Ga, $\lambda = 0.23$; Sm, $\lambda = 0.17$; U, $\lambda = 0.13$; Sr, $\lambda = 0.10$; Eu, $\lambda = 0.099$; Sc, $\lambda = 0.05$; and Zr, $\lambda = 0.04$. In pilot-plant studies the reaction of 304 stainless steel tube sections with Cl was investigated in a $1\frac{1}{2}$ -inch-diameter two-zone fluid-bed reactor. An average penetration rate of 4.6 mils/hr was obtained in a 4.7-hour run at 575°C using 87% Cl (in N). At 625°C , using the same Cl concentration, a 35-mil tube completely reacted in 3.8 hours. The effect of Cl dilution at 625°C was noticeable at concentrations below 48 vol.%. The rate of thermal decomposition of PuF_6 was studied at 140 to 173°C by a static method and from 150 to 250°C by a flow method. A study of the kinetics of decomposition established the mechanism of the reaction. The rate of the reaction has been formulated as concurrent first- and zero-order reactions with respect to PuF_6 pressure in the range between 50 and 1100 mm at 140 to 170°C . The reaction of elemental Br with PuF_6 has been investigated. The stoichiometry of the reaction was established. Primary products of the reaction are PuF_4 and BrF_3 . The utility of the reaction for the separation of U and PuF_6 was demonstrated. In studies of fission product behavior, mixtures of UO_2 - Ru^{106} were reacted with F at 400 and 500°C . It was found that at both of these temperatures Ru was volatilized at a rate equal to or faster than that of U volatilization as UF_6 . However, it was found that RuF_5 decomposed and was deposited on a colder portion of the walls of the fluorination vessel. Two series of in-pile, metal-water experiments were completed. Meltdowns were conducted on stainless steel-urania cermets, U wires, and on ceramic core, metal-clad fuel specimens. Cermets made of 90 wt.% stainless steel with urania showed similar behavior when submerged in water irrespective of whether the sample was in the form of pins or plates; metal temperatures greater than 1500°C were attained. The original geometry was changed into one or two large globules together with many fine particles; the more energetic transients also produced some fine (1-mil diameter) powder. Chemical analyses indicated that a separation of the urania from the stainless steel took place during the melting-quenching cycle of the reactor burst. The larger globules were depleted in U (0.017 to 0.055 wt.% U) whereas the fine particles were more concentrated in U (30 to 64 wt.% U). A Zircaloy-2 clad, ceramic core fuel pin subjected to a 648-Mw-sec pulse (606 cal/gm) on a 50 ms period gave nearly complete destruction of the specimen with 24.0% of the metal jacket reacting with the water. The data obtained to date from the various ceramic and cermet core fuel pins were correlated as a function of the energy of the reactor burst. Results at two different energies are summarized tabularly. (auth)

26114 (ORNL-3147) A SURVEY OF THE HAZARDS INVOLVED IN PROCESSING LIQUID METAL BONDED FUELS. J. B. Adams (Oak Ridge National Lab., Tenn.). Aug. 14, 1961. Contract W-7405-eng-26. 19p.

A survey of the character and magnitude of hazards involved in processing liquid metal bonded fuels was made and the scope of a preliminary experimental program outlined. Processing of SRE and CPPD fuels by mechanical decladding followed by controlled reaction of the collected NaK and Na is recommended over chemical dejacketing methods. Similarly, shearing of PRDC fuel and controlled exposure of the Na in the severed portions to water appears more desirable than chemical dissolution of the metallic cladding. (auth)

26115 (TID-11692) THE DOW ALUMINUM PYROMETALLURGICAL PROCESS. Annual Technical Progress Report Fiscal Year 1960. Robert J Teitel and Gilbert S. Layne (Dow Chemical Co., Midland, Mich.). Aug. 19, 1960 180p Contract AT(11-1)-768.

In the Dow Al Pyrometallurgical process U is dissolved in molten Al and precipitated as UAl_3 by the addition of Mg. The UAl_3 is dissociated by Mg or Zn treatments and metallic U recovered by heating the product in vacuo. Fission products or other impurities are separated from the U by taking advantage of their properties of volatility or insolubility in Al, or solubility in Al-Mg, Mg, or Zn. The basic steps of the DAP process were evaluated experimentally with respect to fission product and Pu separation, quantities of process metals required, U losses, container material requirements, general basic equipment, and processing techniques. From this information, a flow sheet of the process as it may be applied to an EBR II fast breeder reactor fuel cycle and as a "head-end" process for aqueous processing of U-Zr fuel elements was evaluated with favorable potentialities indicated. Detailed experimental data were obtained on experiments using from 1 to 100 grams of U. The results show that the separation of Ce (and therefore the rare earth fission products), Ba, Sr, Y, Zr, Mo and Ru can be maintained at a reasonably low level in a uranium based equilibrium "fissium" fuel. High recoveries of U from low U-Zr fuel elements were demonstrated. Pu can be recovered from Pu-Al melts by the addition of Mg. Very low concentrations of Pu can be coprecipitated (from these melts) with U. (auth)

27065 (ANL-6092) THE DESIGN AND CONSTRUCTION OF THE EBR-II INITIAL FUEL LOADING FACILITY. James E. Ayer and Arthur B. Shuck (Argonne National Lab., Ill.). June 1961. Contract W-31-109-eng-38. 23p

The need for the first core for EBR-II resulted in the design and construction of the Initial Fuel Loading Facility for this reactor. The plant was built to provide the required initial loading, to train personnel, and to test prototype equipment for the remote reprocessing of fuel materials in the EBR-II Fuel Cycle Facility. The facilities include remotely manipulated melting, casting and pin processing equipment, a degreaser, hoods and their atmospheric control system, a gas-purification system, fuel-element-assembly equipment, mold-preparation and balance room, bonding furnaces, a maintenance shop, and a change area. (auth)

27067 (CF-61-4-71) PREPARATION AND SPECIFICATIONS OF MINIATURE FERMI REACTOR CORE B FUEL SPECIMENS FOR IRRADIATION TESTING. J. H. Cherubini and R. C. Williams (Oak Ridge National Lab., Tenn.). Apr. 21, 1961. 22p

The preparation of miniature plate specimens containing dispersions of 33.2 wt % UO_2 in type 347 stainless steel for irradiation testing is described. These plates simulate the prototype roll-bonded stainless steel-clad composite plates intended for the Core B fuel element of the Enrico Fermi sodium-cooled reactor. The significant characteristics of the fabricated irradiation specimens are tabulated. The samples will be used to determine the irradiation behavior

of 33.2 wt.% UO_2 at temperatures of about 1150°F in controlled capsule tests in the Materials Testing Reactor. (auth)

27074 (NDA-2164-2) AN EVALUATION OF U^{233} -THORIUM FUELS FOR FAST BREEDER APPLICATION. Ralph Hankel and Arthur J. Goodman (United Nuclear Corp., White Plains, N. Y.). Aug. 15, 1961 Contract AT(30-1)-2303. 85p

An investigation of the breeding potential, energy costs, and long-term nuclear effects of a number of 300-Mwe reactors using alternate types of U^{233} -Th fuels and fuel element configurations was performed as part of an evaluation of fast U^{233} -Th breeder reactors. The fuel types evaluated were metal alloy, sponge, oxide, oxide-BeO mixtures, carbide, and cermet. Rods, plates, and annuli were the geometric shapes investigated. Over-all system doubling times, for a set of assumed reference conditions, ranged from 19.7 years for carbide fuel to 28.6 years for metal alloy fuel. Fuel cycle costs varied from 3.5 mill/kwh for oxide fuel to 7.5 mill/kwh for metal fuel. These doubling times and fuel costs are based on the best judgment of the near term burnup capability of the fuels considered. A carbide design with a fuel cost of 3.7 mill/kwh and a doubling time of 19.7 years was selected for further detailed evaluation. Of the geometric configurations studied, the rod type element led to lower fuel costs than either plates or annuli. The addition of 25% BeO to oxide fuel in order to increase power density lowered the system doubling time from 27.6 years to 24.3 years, but increased fuel costs from 3.53 mill/kwh to 3.86 mill/kwh. The long-term nuclear effects on breeding potential, reactivity, control requirements, and fuel handling were considered for each fuel material. No fuel material demonstrated long-term characteristics which would affect its feasibility of operation, furthermore, the effects for the several fuel materials are sufficiently similar as not to influence the selection of a preferred fuel. (auth)

27217 (TID-13305) SELECTED OPERATING EXPERIENCE OF COMMISSION POWER REACTORS. J. O. Roberts (Division of Reactor Development, AEC) [1961] 43p

For presentation at the American Inst. of Electrical Engineers, Summer General Meeting, June 18, 1961, Ithaca, N. Y.

The Shippingport station has demonstrated that the pressurized water reactor concept is capable of operating as a base or peak load plant. The EBWR has demonstrated that a direct cycle boiling water reactor can be operated without undue activity at the turbine. Availability factors greater than 90% were obtained for both concepts. The sodium and organic cooled concepts are in the reactor experiment stage; based on operating experience of reactor experiments, these concepts appear to have the characteristics necessary for central power station use. Maintenance experience was obtained on all types of reactors ranging from instrument calibration to major repairs and modification in areas of high radiation levels. Novel techniques and special tools were developed to perform many of the repairs. The manpower and time required for maintenance on the radioactive system is considerably higher than for similar type maintenance on conventional systems. Personnel radiation exposure levels have not exceeded AEC standards even though the level of activity in some areas requiring access were quite high. Radioactive waste, another problem area for nuclear plants, was successfully handled and disposed of without any major technical difficulty. (auth)

27225 THE FAST BREEDER REACTOR. R. V. Moore (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.) and R. Hurst. J. Brit. Nuclear Energy Conf., 6: 161-5 (July 1961)

The basic concepts of fast breeder reactors are discussed. Included are the characteristics, advantages and disadvantages, and performance of the fuel elements, fuel cycles, heat removal systems, control and safety, and the place of fast reactors in the United Kingdom power program. (N.W.R.)

27226 DESIGN AND CONSTRUCTION OF THE DFR CORE, REACTOR VESSEL, FUEL HANDLING EQUIPMENT AND SHIELDING. J. Tatlock (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.), A. G. Frame, P. K. Richards, and R. Bellinger. *J. Brit. Nuclear Energy Conf.*, 6: 166-88 (July 1961).

The main problems encountered in the design, manufacture, and construction of the fuel element, core, reactor vessel, fuel element handling equipment, and shielding are described; and the solutions adopted are discussed in detail. The characteristics and choice of coolant are also discussed. (N.W.R.)

27227 DESIGN AND CONSTRUCTION OF THE DFR HEAT-TRANSFER CIRCUITS, STEAM-GENERATING PLANT AND REACTOR-CONTROL SYSTEM. R. R. Matthews (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.), W. G. Hutchinson, P. K. Richards, A. S. Fenimore, and M. H. Ewbank. *J. Brit. Nuclear Energy Conf.*, 6: 189-210 (July 1961).

The main problems encountered in the design, manufacture, and construction of the heat transfer circuits, steam generating plant, and control systems are described; and the solutions adopted are discussed in detail. An appendix is given containing data of the heat removal system for the 60 Mw(h) core output system. (N.W.R.)

27228 DFR METALLURGICAL PROBLEMS. C. R. Tottle (Univ. of Manchester, Eng.), D. S. Oliver, and K. Q. Bagley. *J. Brit. Nuclear Energy Conf.*, 6: 211-24 (July 1961).

Metallurgical problems encountered in fuel element development and liquid-metal coolant technology are presented and analyzed. Methods and materials are discussed for eliminating swelling, irradiation growth, and thermal cycling effects. The physical, tensile, and creep properties of various canning materials are presented and discussed in terms of their compatibility with the sodium-potassium alloy coolants. The effect of impurities and oxide removal are discussed for the sodium-potassium alloy coolants. (N.W.R.)

27229 DOUNREAY FAST REACTOR PHYSICS. R. D. Smith (United Kingdom Atomic Energy Authority, Winfrith, Dorset, Eng.) and D. C. G. Smith. *J. Brit. Nuclear Energy Conf.*, 6: 225-39 (July 1961).

Calculations and measurements are discussed for some of the reactor physics parameters. The method generally used is to calculate the desired quantity in ZEUS and compare the calculated value with the ZEUS experimental result. The calculations are of two kinds: those which are needed early in the design of the reactor to predict certain characteristics (such as the critical size) which could be checked when the reactor is in operation, and more detailed calculations of the behavior of the reactor which are inconvenient or impossible to confirm by measurements in the reactor itself. The calculation of detailed reaction rates in most parts of the core and breeder and the kinetic behavior of the reactor under fault conditions are in the latter category. (N.W.R.)

27230 TESTING, COMMISSIONING AND INITIAL OPERATION OF THE DFR. J. L. Phillips (Dounreay Experimental Reactor Establishment, Caithness, Scotland),

R. R. Matthews, and W. J. M. Davies. *J. Brit. Nuclear Energy Conf.*, 6: 240-52 (July 1961).

Testing, commissioning, and initial operation methods and procedures are described of some of the over 350 test schedules made. Major items are selected and described in fairly broad outline for a few of the test schedules. Some of the tests described are: sphere pressure test, primary liquid metal pipe weld tests, maintenance flask dry testing, and wet testing of the removal and replacement of equipment or charge and discharge operations. Other topics discussed are welding procedures, installation and calibration of outer and inner rotating shield and associated equipment including control rods, sodium-potassium filling, and critical assembly charging. A program involving operation of the reactor at low powers is also discussed. (N.W.R.)

28008 (ANL-6384) DEVELOPMENT OF A PROCESS FOR SODIUM BONDING OF EBR-II FUEL AND BLANKET ELEMENTS. Edmund S. Sowa and Edward L. Kimont (Argonne National Lab., Ill.). July 1961. Contract W-31-109-eng-38. 22p.

Procedures for assembling EBR-II fuel elements with annular sodium bonds between the uranium rods and the stainless steel claddings are outlined. The results of several meltdown and uranium-settling experiments are given. Bonding experiments were performed: furnace bonding, submerged canning, ultrasonic bonding, centrifuging, pressure pulsing, and vibratory bonding. Vibratory bonding was chosen for the production of the first EBR-II core. (D.L.C.)

28741 (NDA-2147-7) STABILITY STUDIES ON FAST REACTORS WITH DELAYED REACTIVITY COEFFICIENTS. J. Agresta (United Nuclear Corp., White Plains, N. Y.). Aug. 15, 1961. Contract AT(30-1)-2303(XIII). 51p.

A systematic study was made on the effects of power coefficients of the types usually found in fast reactors on reactor stability. Both finite and infinite resonances in the transfer function giving the power response to reactivity oscillations are considered. The following coefficient cases were treated: delayed negative coefficient of the conduction or convection type, either alone or in combination with a prompt coefficient, and a combination of both delayed-convection and delayed-conduction type coefficients. Formulas are given for testing unconditional stability of systems. (D.L.C.)

28814 (NP-10550) TRANSIENT AND OSCILLATOR ANALYSIS FOR HAZARDS EVALUATION COVERING TRIP LEVEL SETTINGS FOR THE ENRICO FERMI ATOMIC POWER PLANT POWER-LIMITING SYSTEM. C. E. Taulbee (Bendix Corp. Research Labs. Div., Southfield, Mich.). Nov. 1, 1960. For Atomic Power Development Associates. 32p. (BC/RDL-1621)

A transient and oscillator analysis for hazards evaluation covering trip level settings for the Enrico Fermi Atomic Power Plant power-limiting system is presented. Establishment of trip-level settings, review of trip-level settings from current simulation results, and description of simulator model are discussed. (M.C.G.)

28838 (LA-2569) NEUTRON TISSUE DOSE RATE SURVEY FOR THE GODIVA II CRITICAL ASSEMBLY. Morris J. Engelke, Edwin A. Bemis, Jr., and J. A. Sayeg (Los Alamos Scientific Lab., N. Mex.). May 1961. Contract W-7405-eng-36. 15p.

A neutron tissue dose rate survey was made for the Godiva II critical assembly using the Hurst neutron proportional counter. Measurements were made at 10 points from 40 to 350 meters. The data indicated values of ap-

proximately 25 mrad/hr per 100 watts operating power at 40 meters to 0.25 mrad/hr per 100 watts operating power at 350 meters. The results are consistent with the mean free path of 250 meters. The data are estimated to be uncertain by no more than 30% (auth)

29716 (ORNL-3160) METALLURGY DIVISION ANNUAL PROGRESS REPORT FOR PERIOD ENDING MAY 31, 1961. (Oak Ridge National Lab., Tenn.). Aug. 17, 1961. Contract W-7405-eng-26. 207p.

Fundamental Alloying. Studies of crystal structures, reactions at metal surfaces, spectroscopy of molten salts, mechanical deformation, and alloy theory are reported. Long-Range Applied Metallurgy. A thermal comparator is described and the characteristic temperature of UO_2 determined. Sintering studies were carried out on ThO_2 . The diffusion of fission products in fuel and of Al^{26} and Mn^{54} in Al and the reaction of Be with UC were studied. Transformation and oxidation data were obtained for a number of Zr alloys. Reactor Metallurgy. A large number of ceramic technology projects are described. Some corrosion data are given for metals exposed to impure He and molten fluorides. Studies were made of the fission-gas-retention properties of ceramic fuel bodies. A large number of materials compatibility studies are described. The mechanical properties of some reactor materials were studied. Fabrication work was conducted to develop materials for application in low-, medium-, and high-temperature reactors or systems. A large number of new metallographic and nondestructive testing techniques are reported. Studies were carried out on the oxidation, carburization, and stability of alloys. Equipment for postirradiation examination is described. Preparation of some alloys and dispersion fuels by powder metallurgy methods was studied. The development of welding and brazing techniques for reactor materials is described. (D.L.C.)

29788 (ANL-4983(Del.)) PHYSICS DIVISION, SUPPLEMENT TO THE SUMMARY REPORT, JUNE THROUGH NOVEMBER 1952 (Argonne National Lab., Ill.). Mar. 1953. Decl. with deletions Feb. 23, 1960. Contract W-31-109-eng-38. 29p

Transmission measurements with the fast chopper were made on U^{233} , normal U, and Pu. A search was made for Pu^{244} . The crystal structures of $\text{Mg}(\text{UO}_2)_2$ and KU_2F_6 were investigated. The magnitude of the resonance flux in CP-3' was measured. Multigroup methods for concentrated fast assemblies were studied. Work on the ZPR-II Reactor was continued. (M.C.G.)

30193 (ANL-6337) CRITICAL STUDIES OF A DILUTE OXIDE FAST REACTOR CORE (ZPR-III ASSEMBLY 30). P. I. Amundson, A. L. Hess, W. P. Keeney, J. K. Long, and R. L. McVean (Argonne National Lab., Idaho Falls, Idaho). May 1961. Contract W-31-109-eng-38. 23p.

Critical studies of a fast reactor core containing a simulated oxide fuel having an oxygen-uranium atomic ratio of 1:1 are described. Calculated and experimental critical masses are compared. Experimental results are given for fission ratio, central reactivity coefficient, fuel bunching, and distributed worth measurements. (auth)

30195 (ANL-6382) A DESCRIPTION OF INTEGRAL PHYSICS DATA FOR FAST REACTOR DESIGN. W. B. Loewenstein and D. Meneghetti (Argonne National Lab., Ill.). July 1961. Contract W-31-109-eng-38. 29p.

Integral physics data for fast reactor design are discussed. The measurements needed include those of critical mass, shape factor, detector ratios, neutron spectra, material replacement experiments, reflector savings, neutron lifetime, $\text{Rossi-}\alpha$, and similar quantities. Topics

covered include Pu- and U^{233} -fueled systems, highly enriched U^{235} systems in optimum geometry, uranium cores of various enrichments and dilutions, extreme geometry critical experiments, specific reactor systems, core mockup inhomogeneities, spectral studies and detector ratios, uranium equilibrium spectrum data, material-replacement measurements, fast reactor dynamics, and suggested future experiments and experimental programs (M.C.G.)

30196 (ANL-6399) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT [FOR] JULY 1961. (Argonne National Lab., Ill.). Aug. 15, 1961. Contract W-31-109-eng-38. 75p

A summary is presented of activities in reactor and general engineering research programs. Discussions are included for developments in EBWR, BORAX-V, ZPR-III, ZPR-VI, ZPR-IX, EBR-I, and EBR-II. Reactor safety studies were performed for fast and thermal reactors. Nuclear technology developments are discussed for applied nuclear and reactor physics, reactor fuels and materials development, heat engineering studies, separations processes, and advanced reactor concepts. (B.O.G.)

30218 (NDA-2147-8) STUDIES RELATED TO PREDICTING COOLANT BEHAVIOR FOR FAST REACTOR SAFETY. R. P. Stein (United Nuclear Corp., White Plains, N.Y.). Aug. 31, 1961. Contract AT(30-1)-2303(XIII). 77p.

Various problems related to the prediction of the behavior of liquid sodium coolant in fast reactors during abnormal operating conditions are studied. Included are two-phase flow characteristics, transient forced-convection heat transfer, transient fuel-element heat transfer, and the prediction of coolant behavior in a generalized fast reactor coolant channel. (auth)

30254 (GEAP-3646) CALCULATION OF DOPPLER COEFFICIENT AND OTHER SAFETY PARAMETERS FOR A LARGE FAST OXIDE REACTOR. P. Greebler, B. A. Hutchins, and J. R. Sueoka (General Electric Co. Atomic Power Equipment Dept., Schenectady, N.Y.). Mar. 23, 1961. Contract AT(04-3)-189. 43p.

Several safety parameters are calculated for a large, fast, Na-cooled, oxide-fueled reactor. The Doppler coefficient is studied as regards its effect on the reactivity during a power excursion, and its dependence on the neutron spectrum, on the concentrations of U^{238} , Pu^{239} , and Pu^{240} , on spatial temperature and power distributions, and on temperature. Other parameters studied include the Na temperature coefficient of reactivity, and the reactivity insertion caused by total Na loss, reactivity coefficients caused by thermal expansion of fuel and steel cladding, the excess operating reactivity, and the reactivity caused by a fuel slump. The effects of a Be reflector on neutron lifetimes are determined. The safety aspects are balanced against economic considerations. (T.F.H.)

30282 ON THE RESONANCE INSTABILITY OF EBR-1. Yoshie Ebizuka (Tokyo Inst. of Tech.), Waichiro Kawakami, and Masuo Shindo. J. At. Energy Soc. Japan, 3: 522-30 (July 1961). (In Japanese)

The experimental Breeder Reactor, EBR-1, received considerable attention when an oscillatory instability under abnormal operating conditions was observed. Investigations of kinetic characteristics revealed that a prompt positive power coefficient of reactivity, which is generally believed due to the bowing of fuel elements, and a more delayed negative power coefficient exist. Controlled oscillator measurements were performed with the result that EBR-1 exhibited resonance instability. Though a number of theoretical models of the coefficients was presented to study

the instability, they do not fit the mechanism of the delayed negative coefficient. The models of feedback reactivity for Mark II and Mark III core loading are presented. For Mark II, the main point to be emphasized is the cause of time delay by heat transfer from the core to the structure. The delayed negative reactivity is treated by a transfer function that takes into account axial temperature distribution in the core. It is demonstrated that the delayed negative reactivity combined with the prompt positive reactivity is sufficient to permit the resonance instability. For Mark III, the frequency transfer functions of feedback reactivity were separated from the experiment. The model is compared with the result of measurements. (auth)

30284 AN ECONOMIC FAST REACTOR. L. R. Blake (Dounreay Experimental Reactor Establishment, Risley, Lancs, Eng.) Nuclear Eng., 6: 371-5 (Sept. 1961).

A fast reactor can be built for a capital cost as low as \$240/kw installed, with standards of safety significantly better than any existing fast reactor and adequate for a large-scale power program. Tests with unalloyed uranium to nearly 5% burn-up under conditions appreciably worse than in a fast reactor suggest that 6% average burn-up can be achieved with uranium/plutonium metal fuel at a power density of about 50 to 100 w/g, with a maximum center temperature of 700°C and with coolant average outlet temperature of 450°C. These conditions of coolant temperature and power density are conservative with respect to current fast reactor practice, but the center temperature and the burn-up are very much higher. Assuming \$240/kw net, and 5.5% interest charges and 20 years reactor life, giving 8 3% capital charges, the electric power costs of the fast reactor with a 0.7 load factor are approximately 3.7 mil/kwh. (N.W.R.)

30285 ENRICO FERMI. Nuclear Eng., 6: 377-82 (Sept. 1961).

The Enrico Fermi fast breeder reactor plant at Monroe, Michigan is the first to be constructed of this type for the production of power and for experimental study. The reactor will have an initial charge of uranium 10 wt % molybdenum alloy fuel elements. The reactor is cooled by three circuits in series. The coolant is liquid sodium. Vapor from the third circuit is passed to the turbines for the generation of power. A complete description is given of the installation. Descriptions are given of the core, the fuel handling equipment, the primary and secondary circuits, and the control and safety systems. Fuel element fabrication is also described. (N.W.R.)

30287 BREEDER REACTORS Alvin M. Weinberg Sci. American, 202: No. 1, 82-93 (Jan. 1960).

The breeding cycles, characteristics, and economics of fast and thermal breeder reactors are described. The descriptions cover all known breeder types. Schematic diagrams are presented for the EBR-2, HRE-2, and the Enrico Fermi Fast Breeder Reactor. (N.W.R.)

30292 (ANL-6383) THE PHYSICS DESIGN OF THE EBR-II. W. B. Loewenstein (Argonne National Lab., Ill.). July 1961. Contract W-31-109-eng-38. 70p.

The physics design problems of the EBR-II are summarized. These include analysis of the EBR-II engineering design as well as applicable zero-power critical experiments. Pertinent reactor safety problems are reviewed. Safety considerations bearing on normal plant operation and manipulations within the reactor are emphasized. The implication of controlled in-pile meltdown experiments is considered. Irradiation damage and metallurgical phase phenomena are summarized and related to reactivity. The nuclear performance of the system is considered in terms

of actual plant operation. The predicted shift of both power and reactivity from core to radial reflector is described. (auth)

31489 (ANL-6401) CRITICAL STUDIES OF DILUTE CARBIDE FAST REACTOR CORE. ZPR-III Assembly 34. R. J. Hubert, J. K. Long, R. L. McVean, and J. M. Gasidlo (Argonne National Lab., Idaho Falls, Idaho). May 1961. Contract W-31-109-eng-38. 21p.

Critical studies were made with a simulated, large, dilute power reactor having uranium carbide as fuel. The uranium in the core was 30.7% enriched, and the atomic ratio of uranium to carbon was 0.946. The critical mass was 503.01 kg U²³⁵ and the critical volume 574.47 liters. Central reactivity coefficients, effective fission cross-section ratios, heterogeneity effects, reactivity worth of distributed materials, foil irradiations, and the average prompt neutron lifetime were measured. Multigroup calculations using the Yiftah, Okrent, and Moldauer cross-section set overestimated k for the critical configuration by 4.7%. (auth)

31800 (NDA-2116-2) SPONGE FUEL EVALUATION. Quarterly Technical Report for the Period Ending June 30, 1959. J. M. McKee (Nuclear Development Corp. of America, White Plains, N. Y.). Aug. 15, 1959. Contract AT(30-1)-2303. 25p.

Uranium powder, packed in a tube and infiltrated with sodium, is proposed as a promising fuel for nuclear power reactors. A two-year program for the experimental evaluation of this fuel, termed "sponge," is described. Technical progress during the first quarter consisted mostly of capsule design calculations and drawings. Fabrication of two control consoles was started. Natural uranium shot was obtained for out-of-pile tests. Sponge fuel showed a higher reactivity than U-10% Mo in both thermal and fast sodium-cooled reactors and a slight gain in breeding ratio in the latter. (auth)

31801 (NDA-2116-3) SPONGE FUEL EVALUATION. Quarterly Technical Report for Period Ending September 30, 1959. J. M. McKee (Nuclear Development Corp. of America, White Plains, N. Y.). Dec. 4, 1959. Contract AT(30-1)-2303. 11p.

Fabrication and assembly of the first two irradiation test capsules is about 60% complete. These will test two sizes of unalloyed 3% enriched uranium shot at controlled central temperatures of 1100 and 1400°F to a burnup of 1.25 at.%. The two control consoles are 90% complete. Construction of a vacuum distillation box for use during examination of the irradiated specimens is in progress. Packing tests with unalloyed natural uranium shot showed that a mixture of particle sizes in the range of 20 to 200 mesh (33 to 3 mils diameter) can be packed to 76% of solid metal density. In these sizes the shot is not pyrophoric and can be handled in air. Equipment for making sponge fuel specimens was assembled. Filling and sealing techniques are being developed on natural uranium specimens. Fabrication of 3% enriched sponge fuel specimens in niobium containers has just begun. The basic design parameters of sponge fuel elements for the SGR, EFFBR, and EBR-II were calculated as a function of the fuel density, thermal conductivity and central temperature of the sponge. The resulting designs were used to estimate the fuel cycle costs in the SGR and EFFBR for sponge fuel in comparison with present fuels. It appears that sponge fuel will be economically attractive if the assumed thermal conductivity, central temperature, and burnup can be realized. Savings are expected mainly in fabrication costs, due to larger (and hence fewer) elements and to simpler fabrication operations. (auth)

31803 (NP-10233(p 1-34)) OPERATIONAL EXPERIENCE ON DOUNREAY FAST REACTOR. J. Allen ([United Kingdom Atomic Energy Authority. Development and Engineering Group, Dounreay, Caithness, Scotland]).

The Dounreay Fast Reactor is described in some detail and the steps involved in its commission discussed. Experience gained in the operation of the reactor is presented under the following headings liquid metal circuits, zero-energy physics experiments, and shut-down. The reactor instrumentation is discussed (D L C)

31804 (NP-10233(p 35-47)) THE INSTRUMENTATION OF THE DOUNREAY FAST REACTOR. R. Smart ([United Kingdom Atomic Energy Authority. Development and Engineering Group, Dounreay, Caithness, Scotland])

Instrumentation of the Dounreay Fast Reactor is discussed under the following headings control room, neutron flux measurements, control rod position, liquid metal flow, level, pressure, temperature, and oxidation measurements, fuel element damage detection, steam plant, safety and alarm systems, and transfer function analyzer. (D L C)

31807 (NP-10233(p 65-78)) KINETIC BEHAVIOUR OF FAST REACTORS. D C G Smith ([United Kingdom Atomic Energy Authority. Development and Engineering Group, Dounreay, Caithness, Scotland]).

The behavior of a reactor following a perturbation is due to nuclear constants and to reactivity feedback from thermal and mechanical property variations. The effects of nuclear constants on reactor kinetics and the components of reactivity feedback are discussed. The calculation and measurements of fast reactor reactivity coefficients are considered, and the effects of various feedback types on reactor performance are outlined (D L C)

31852 (ANL-6290) PREPARATION OF ALLOY FOR FIRST CORE LOADING OF EBR-II. Donald C. Hampson (Argonne National Lab., Ill.). Aug. 1961. Contract W-31-109-eng-38. 17p

The alloy used for the fabrication of the fuel pins for the first core loading of the second Experimental Breeder Reactor (EBR-II) was prepared in the prototype equipment developed for the melt-refining processing of the irradiated EBR-II fuel. One hundred and twenty-five 10-kg ingots were made, of which 40 were unenriched uranium-fissium alloy and 85 were enriched uranium-fissium alloy. In addition, nineteen 10-kg batches of unenriched uranium-fissium scrap and forty-seven 10-kg batches of enriched uranium-fissium alloy scrap were melted for consolidation into ingots. The average yield for the alloy preparation runs was 96.5% and for the scrap remelt runs was 93%. The chemical and isotopic compositions of the ingots produced were all within specifications (95 ± 1.0 wt % uranium, of which 48.1 ± 1.2 wt % is ^{235}U). (auth)

31869 (NP-10700) "TRANSIENT AND OSCILLATOR ANALYSIS FOR HAZARDS EVALUATION" AND "OSCILLATOR ANALYSIS FOR REACTOR STARTUP TESTS" FOR THE ENRICO FERMI ATOMIC POWER PLANT. Final Report. Report No. 1653. C. D. Taulbee, E. H. Lemon, E. B. Skuta, and C. F. Connolly (Bendix Corp. Research Labs., Div., Southfield, Mich.) Dec. 30, 1960. For Atomic Power Development Associates, Inc. and Power Reactor Development Co. 153p.

A discussion is given of the work done on the programs involved in describing the Enrico Fermi Reactor mathematically and in turn electrically on a large analog computer, then simulating various transient conditions and transfer functions to evaluate the kinetic behavior of the reactor. Many of the simulations, repetitions of similar simulations done previously, were re-evaluated because of changes in fuel-element design and full-power operating level. The simulation program is described in terms of

transient simulation runs, analysis of transient temperatures, trip-level settings; transfer function simulation results; and the simulation model. (B.O.G.)

31893 DOUNREAY EXPERIMENTAL REACTOR ESTABLISHMENT INFORMATION BOOKLET (United Kingdom Atomic Energy Authority. Industrial Group. Dounreay Experimental Reactor Establishment, Caithness, Scotland) May 1961. 59p. \$0.86(BIS)

Work of the five divisions which make up the Dounreay Experimental Reactor Establishment is discussed. The Dounreay Fast Reactor and the Dounreay Materials Testing Reactor are described. Experimental work on assemblies of fissile material in various shapes and sizes is outlined. The development, fabrication, and reprocessing of fuel elements is described. The responsibilities of the engineering, health and safety, and administration divisions are summarized (M C G)

32911 (ANL-6334) STUDIES OF FAST REACTOR FUEL ELEMENT BEHAVIOR UNDER TRANSIENT HEATING TO FAILURE. I. INITIAL EXPERIMENTS ON METALLIC SAMPLES IN THE ABSENCE OF COOLANT.

C. E. Dickerman, E. S. Sowa, D. Okrent, J. Monaweck, and L. B. Miller (Argonne National Lab., Ill.). Aug. 1961. Contract W-31-109-Eng-38. 96p.

Meltdown tests on single metallic unirradiated fuel elements in TREAT are described. The fuel elements (EBR-II Mark I fuel pins, EBR-II fuel pins with refractory Nb or Ta cladding, and Fermi-I fuel pins) are tested in an inert atmosphere, with no coolant. The fuel elements are exposed to reactor power bursts of 200 msec to 25 sec duration, under conditions simulating fast reactor operations. For these tests, the type of power burst, the integrated power, the fuel enrichment, the maximum cladding temperature, and the effects of the test on the fuel element are recorded (T F.H.)

32913 (ANL-6409) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, AUGUST 1961. (Argonne National Lab., Ill.). Sept. 15, 1961. Contract W-31-109-eng-38. 72p.

Progress is reviewed on the following reactors EBWR, Borax-V, ZPR-III, ZPR-VI, ZPR-IX, EBR-I, and EBR-II. An outline of fast and slow reactor safety studies in TREAT is presented. Progress is also reported in applied nuclear and reactor physics, development of reactor fuels, materials, and components, heat engineering technology, separation processes, and advanced reactor concepts. (T.F.H.)

32914 (ANL-6433) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, SEPTEMBER 1961. (Argonne National Lab., Ill.). Oct. 15, 1961. Contract W-31-109-eng-38. 65p.

Data from examination of blade-type control rods which were used in BORAX are discussed. Operation and maintenance of EBWR is outlined. In work on Borax V, modifications for easier installation of reactor and components is outlined followed by discussion of superheat fuel element development, and fabrication of various reactor components. Borax reactor design is also reported along with information on development and testing. In research on sodium-cooled reactors, activities are summarized in the ZPR III and ZPR IV programs along with developmental work on EBR I and II. Studies on reactor safety are reported and activities in a program of nuclear technology and general support are outlined. (J.R.D.)

32987 (NP-10766(p,32-5)) FAST NEUTRON REACTORS IN THE U.S.S.R. O. D. Kazachovskii (Kazachovskii) (U.S.S.R.)

A summary discussion is given of work conducted to-

ward developing fast neutron reactors in the U.S.S.R. The characteristics and the experience gained in the operation of the BR-5 reactor are described. The BR-5 reactor is

plutonium-oxide-fueled, liquid-sodium-cooled, nickel-reflected, and has a maximum output of 5 Mw. Intended uses of this research reactor are outlined. (B.O.G.)

SUBJECT INDEX

ARGONNE POWER BREEDER REACTOR, U.S.A.

coolant flow and reactivity of, 12 971(R) (ANL-5060(Del.))
 core design, effects of corrosion, high temperature, and thermal stresses on, 11 7839 (LWS-22534)
 criticality studies, 12 975(R) (ANL-5398(Del.))
 design, 12 991 (CF-3582)
 design, analysis, and feasibility, 11 13866(R) (ANL-5208(Del.))
 design and criticality evaluation, 11 13864(R) (ANL-5134)
 design development, 12 970(R) (ANL-5012)
 design development, 12 971(R) (ANL-5060(Del.))
 fuel element design, 11 13868(R) (ANL-5260(Del. 2))
 fuel process loss effects on breeding gain and converting time, 11 9788 (ANL-LB-SL-1053)
 mathematical analysis, 11 11745(R) (ANL-5371)
 plutonium quality in graphite-reflected blanket, 11 13864(R) (ANL-5134)
 scope of study, 12 2063 (CF-52-11-164)

BREEDER REACTORS

analysis of PuO_2 - UO_2 fueled, 14 18676 (ANL-6122(p 116-33))
 bibliography on fuel breeding in, 13 19740 (TID-3534)
 blanket calculations, 12 2089 (KAPL-M-JHS-2)
 blanket calculations for, 12 1045 (KAPL-M-HAB-2)
 blanket calculations for, 12 1047 (KAPL-M-HB-31)
 blanket design for fast, 15 15228
 breeder element design, 14 26495
 breeder element design for, 14 21900
 breeder elements, merits of depleted vs natural uranium for, 14 23224 (NUMEC-TM-P-11)
 breeding and conversion processes in, 13 7235
 breeding characteristics in, 14 18678 (ANL-6122(p 149-62))
 breeding elements for, 13 9395
 breeding elements for, 13 9397
 breeding in sodium-cooled fast, 14 18679
 breeding ratio in, sensitivity to cross-section uncertainties, 14 18674 (ANL-6122(p 93-100))
 characteristics of fast, 15 10554
 chemical processing economics for, 7 5854 (AECD-3550)
 comparison of sodium-cooled with high-pressure water, 6 5711 (KAPL-591)
 computational survey of idealized fast, 14 18673 (ANL-6122(p.77-92))
 computational survey of idealized fast, 15 7110
 control system for, 13 14915
 core materials for, 12 12781
 coupled fast-thermal, 12 6218
 description of fast sodium-cooled, 13 14066 (NP-7383)
 description of Soviet BN-50 and BN-250, 13 7175 (A/CONF 15/P/2129)
 description of Soviet BR-1, BR-2, BR-5 programs, 13 7175 (A/CONF.15/P/2129)
 description of Soviet BR-2, 14 17628
 description of Soviet BR-5, 14 1125
 description of uranium-233 fueled, 13 2582 (CF-58-10-37)
 design calculations for fast, 12 996 (CF-52-8-230)
 design, 12 5677
 design of, 14 9195
 design of, 15 7111
 design of, 15 17811
 design of breeding components in, 14 18675 (ANL-6122(p 101-15))

design of fast, 12 991 (CF-3582)
 design of fast, 12 17302 (TID-2501(Del.)(p 425-40))
 design of fast, 14 26494
 design of fast, 15 8297
 design of fast, 15 8375
 design of fast, 15 8378
 design of fast, 15 13961
 design of fast, 15 15261
 design of fast, 15 23145
 design of fast molten fueled, 11 13957 (LWS-22710)
 design of fast oxide, 15 2332 (GEAP-3347)
 design of fast, review of, 10 10528 (TID-7506(Pt 1))
 design of sodium-cooled fast, 13 983
 design of two-zone, 14 2239
 development, 10 8949 (ANL-4481)
 development of, 10 3650 (ANL-4076)
 development of, 10 5297 (ANL-4092)
 development of fast, 10 9598 (AECU-3195)
 development of fast, 14 14593
 development of sodium-cooled fast, 11 2121 (NP-5997)
 development program, 7 2137
 development program review, 6 1318 (TID-5031(p 34-57))
 development status of fast, 13 16576
 Doppler coefficient for large sodium-cooled fast, 15 30254 (GEAP-3646)
 economics and technical problems of, 9 1701
 economics of, 8 6304
 economics of, 14 22628
 economics of, 15 2357
 economics of fast, 13 7270 (A/CONF 15/P/2028)
 economics of fast sodium-cooled, 11 6102
 economics of sodium-cooled fast, 15 12662 (TID-8533)
 equations for breeding in, 12 1034 (KAPL-334)
 feasibility study of uranium-233-thorium system, 15 21887
 fuel alloys for fast, 15 1908
 fuel and blanket processing for, bibliography on, 14 5262 (TID-3529(1st Rev.))
 fuel and blanket reprocessing for fast, 14 13793
 fuel and blanket subassemblies, handling of spent, 13 10714 (AECU-4058)
 fuel development for fast, carbide, 14 15894 (NDA-2140-2)
 fuel element design, 15 15267
 fuel element design for, 13 4353
 fuel elements for, 15 19815
 fuel elements for fast, 15 6368
 fuel elements for fast sodium-cooled, oxide, 13 3432
 fuel element material for fast, 14 25873
 fuel processing, 12 5676
 fuel reprocessing for 500-MW fast, 14 13791
 fuel systems for fast, 13 18725 (BMI-APDA-636)
 fuel systems for fast, 13 18726 (BMI-APDA-641)
 fuel systems for fast, 13 18727 (BMI-APDA-643)
 literature review of fast, 13 4237 (APEX-442)
 multigroup analysis of PuO_2 fueled, 12 3238 (KAPL-M-JBS-9)
 need for, 14 8297
 oxide fuel design for fast, 12 655 (KAPL-1784)
 physics and economic aspects of, 2 21
 physics calculations for fast, 12 976 (ANL-5492(Del.))
 physics calculations of fast, 15 11912
 physics of, 11 2128
 physics of, 12 970 (ANL-5012)

BREEDER REACTORS (Continued)

physics of breeding in, 14 18671 (ANL-6122)
 production of enriched nuclear fuel by, 4 4359
 radiation escape from breeding blankets of, 10 10961 (KAPL-M-JBS-4)
 reactivity conditions for continual operation and startup of, 10 4438 (KAPL-494)
 reactivity effects of protactinium-233 buildup in, 15 20343
 review of, 10 11515
 review of Soviet BR-1, BR-2, and BR-3 programs, 13 3395
 safety features of fast, 11 4110
 safety studies of fast, 14 1112 (NRL-Memo-819)
 safety studies of fast, 14 1113 (NRL-Memo 975)
 study of, 12 2063 (CF-52-11-164)
 study of 300 MW(e) sodium-cooled fast, 15 13977 (GEAP-3287)
 technical data on, 12 15904
 technology of fast, 12 9468
 theory of fast sodium-cooled, 11 13138 (CF-55-6-157)
 welding processes for fast, 13 18140

CRITICAL ASSEMBLIES

calculation of parametric information for ZPR-III, 15 17760
 coupled aspects of ZPR-V, 12 3206 (ANL-5775)
 criticality studies of ZPR-III, 15 31489 (ANL-6401)
 criticality studies with ZPR-III, 15 30193 (ANL-6337)
 danger coefficient data for Godiva fast, 11 13955 (LA-1708)
 description and use of ZPR-V, 12 15024 (A/CONF 15/P/599)
 description of Fast Exponential Experiment, 10 3384 (ANL-3384)
 description of Godiva I and II, 12 14987 (A/CONF 15/P/419)
 description of ZPR-III, 10 3226 (ANL-5513)
 description of ZPR-III, 12 15023 (A/CONF 15/P/598)
 description of ZPR IV', 12 15068 (A/CONF 15/P/1547)
 design and development of ZPR-III, ZPR-VI, and ZPR-IX, 15 15209 (ANL-6328)
 design of Godiva II, 15 7044
 development and operation of ZPR III, ZPR-VI, and ZPR-IX, 15 4708 (ANL-6253)
 development of ZPR-II, 15 29788 (ANL-4983(Del.))
 development of ZPR-III, ZPR-VI, and ZPR-IX, 15 32913 (ANL-6409)
 development of ZPR-III, ZPR-VI, and ZPR-IX, 15 17738 (ANL-6343)
 development of ZPR-III, ZPR-IV, and ZPR-IX, 15 20375 (ANL-6355)
 dose and flux measurements on Godiva, 14 3851 (LA-2355)
 dose rates from Godiva, 15 6160
 dosimetry for Godiva II, 13 23034 (LA-2310)
 effects of heterogeneity effects in ZPR-III, 15 10368 (ANL-6218)
 fast reactor predictions with Godiva, 14 10952 (ANL-5321(Rev.))
 hazard evaluation of ZPR-III, 12 2055 (ANL-FWT-105)
 hazards of ZPR-V, 13 23030 (ANL-5547)
 hazards summary for ZPR-VII, 12 5731 (ANL-5715)
 impulse response measurements with, 15 16651 (TID-12489)
 kinetic behavior of ZEPHYR with air-spaced core, 10 4915 (AERE-RP/P-1778)
 neutron energy spectra in ZPR-V, 14 13462
 neutron flux and dose measurements for Godiva II, 14 26032 (LA-2432)
 neutron leakage from fast, 12 2047 (TID-10073)
 neutron pulses in Godiva, analysis of, 13 17971 (CF-59-2-78)
 neutron spectra in, ZPR-V studies of, 13 17524
 neutron spectrum of Godiva, 15 6161
 operation at LASL, 10 10534
 operation of Godiva fast, 11 13953 (LA-1614)
 operation of Godiva fast, 11 13954 (LA-1653)
 operation of ZPR-III, 13 16586
 prompt excursions in Godiva, 13 16585
 radiation effects testing with Godiva II, 13 13026 (SCR-76)
 reactivity effects of inhomogeneities in ZPR III, 11 6521 (ANL 5657) (ANL-5657)
 review of fast, 14 12330
 rod drop experiments in ZPR-I, 11 13864 (ANL-5134)
 studies with ZPR-III and ZPR-IV, 15 32914 (ANL-6433)
 theory for fast, 12 1073 (LA-1525)
 time behavior of fast, 12 1075 (LA-2029)
 use for analysis of fast reactors, 13 16587
 use of ZPR-III, in EBR-II studies, 15 15275 (ANL-6301)
 zero power transfer function of ZPR-III, 11 13488 (ANL-5782)

DOUNREAY FAST REACTOR (DFR), U.K.

chemical and metallurgical plants, 11 9126
 chemical facilities, 11 9483, 10306
 civil engineering, 11 7332, 9128
 commissioning and start-up tests, 14 4138
 commissioning and startup, 14 7139
 commissioning and testing methods, 15 27230
 construction, 11 9110, 9129
 control and instrumentation facilities, 14 4139
 control by transfer function analyzer, 13 14068 (RIC-2/6)
 control rod position indicator design, 13 9367 (IGR-TN/CA 958)
 control system, 11 9124
 control system, 12- 7055(P)
 cooling system and reactor vessel fabrication problems, 12 15889
 cooling system operation, 11 9124
 core components, fabrication, 13 2575
 description of, 10 10535
 description, 15 10562
 description, 15 31893
 design, 11 8180 12248
 design and construction, 15 27226
 design construction, and operation, 12 6851
 design problems, review, 13 7225 (A/CONF 15/P/274)
 development, 12 16745
 development status, 13 14931 (NP 7539)
 diagrams, 11 8181
 fuel element development, metallurgical problems, 15 27228
 fuel element fabrication, 13 7200 (A/CONF 15/P/44)
 fuel element fabrication by precision casting, 13 16998 (RDB(C)/TN-74)
 fuel element internal inspection, 13- 16565
 fuel element manufacture and inspection, 11 9483
 fuel elements, 11 11444
 fuel loading, 11 9129, 9483
 heat exchange system, manufacture and installation, 11 9125
 heat transfer circuit design and construction, 15 27227
 instrumentation, 15 31804 (NP-10233(p 35-47))
 instrumentation for coolant, flux, and temperature control, 15 19579
 instrumentation review, 12 10096
 instrumentation, transfer function analyzer, 13 14068 (RIC-2/6)
 operation and performance, 15 31803 (NP-10233(p 1-34))
 oscillator experiments on, 15 3598 (ANL-6205(p 201-13))
 parameter calculations and measurements, physics, 15 27229
 reactivity effects controlled by boron shutoff and boron oscillator, 12 2500 (FRDC/P-252)
 scale model of, 10 8678
 specifications and cutaway drawings, 12 15905
 waste disposal system, 11 7332, 9483

EXPERIMENTAL BREEDER REACTOR (EBR-1), U.S.A.

analysis of power feedback relations, 15 3602 (ANL-6205(p 251-68))
 analytical studies, design data, and research and development results, 13 481
 breeding and fuel element fabrication for Mark III core, 12 15046 (A/CONF.15/P/791)
 breeding element fabrication, 11 7720 (ANL 5009), 10921(R) (ANL-4238), 10922 (ANL 5132(Del.))
 burnout, EBR-I meltdown physical and metallurgical changes in core, 12 3205 (ANL-5731)
 component developing and testing, 15 32913(R) (ANL 6409)
 conference papers on, 12 5013 (TID-7548)
 constants of, 9 757
 construction of, 10 6359
 conversion ratio, physical determination, 11 9786 (ANL-5222)
 conversion ratio, radiochemical determination, 11 9784 (ANL-5095), 13517
 coolant flow in Mark III, parallel, 13. 8227(R) (AECU-4007)
 coolant flow studies in Mark III, 13- 8228(R) (AECU-4008)
 coolant temperatures, structural response to oscillating, 15 21735
 coolant transfer system, 11 11746(R) (ANL-5471)
 cooling system and pumps, 11 6521(R) (ANL-5657), 7718(R) (ANL-4274)
 core development and fabrication, 15 15209(R) (ANL-6328)
 core dissolution flowsheet, 15 30767(R) (IDO 14410(Del.))
 Core-III design, 12- 8713 (ANL-5836)
 core meltdown, 11 3575
 core meltdown, analysis, 12- 15098 (A/CONF.15/P/2156)

EXPERIMENTAL BREEDER REACTOR (EBR-1), U.S.A. (Continued)

core structure, 11 11746(R) (ANL-5471), 13868(R) (ANL-5260(Del 2))
cost, characteristics, and operating experience, 12 15061 (A/CONF 15/P/1075)
critical mock up of, temperature coefficient measurements on, 12 7458 (ANL 5843)
criticality calculations for plutonium-fueled, 15 3643 (ANL-6217)
criticality studies, 11 1342, 7707 (ANL-4284), 8562(R) (ANL-4798(Del)) , 11747(R) (ANL-5561), 13866(R) (ANL-5208(Del))
criticality studies, 12 975(R) (ANL-5398(Del))
criticality studies, 14 18796(R) (ANL-5668)
design, 11 681, 1123 (ANL 5548), 4664(R) (ANL 5601), 10923(R) (ANL 5297), 11742(R) (ANL 4554), 11743(R) (ANL-5023), 11744(R) (ANL 5313), 11745(R) (ANL 5371), 13869(R) (ANL 5461), 13870(R) (ANL 5594(Del)) , 13879 (APDA 105)
design and development, 12 3866(R) (ANL-5514(Del))
design data, 14 9228
design development, 12 974(R) (ANL 5345(Del 2))
design of, 9 757
development, 12 977(R) (ANL-5511)
development, 15 2296(R) (ANL-6215)
development, 15 2297(R) (ANL-6234)
development, 15 20375(R) (ANL-6355)
development, 15 23116(R) (ANL-6374)
development, 15 30196(R) (ANL-6399)
development and operation, 15 4708(R) (ANL-6253)
development and operation, 15 17738(R) (ANL-6343)
development of Mark II and Mark IV cores, 15 12452(R) (ANL-7307)
Doppler temperature effect in, 13 15811
exponential power changes, analysis of temperatures and expansions resulting from, 12 14270
fabrication of Mark IV core, 15 25621(R) (ANL-6387)
facilities for, 11 11742(R) (ANL-4554)
fission and activation ratios in, 10 11502
fission rate measurements in, 11 9780 (ANL-4811), 13513
for fuel cycle plant bearing material development, 13 20138 (ANL-5973)
fuel element and core-3 welding, 13 8982 (TID-7562(p 78-135))
fuel element cladding materials, 14 18106(R) (BMI-1409)
fuel element cooling, 11 3082
fuel element design for Mark III core, 13 17405
fuel element development, 12 9448 (TID-7546(p 751-77))
fuel element development, radiation effects on clad and unclad, 14 5637 (ANL-5918)
fuel element development and fabrication for Core IV, 15 22692(R) (ANL-6330)
fuel element dimensional stability, 14 15091 (CF-53-3-276(Pt 1)(Del.) (p 36-50))
fuel element dissolution 11 7488(R) (CF 53-6-105), 7825 (IDO 14331), 7856 (ORNL 1137), 8323 (ANL 4530), 9580(R) (ANL 5494(Del)) , 9581(R) (ANL 5633)
fuel element fabrication, 12 969 (ANL-4847(Rev))
fuel element fabrication, 12 978 (ANL-5568(Del.))
fuel element fabrication, 12 1976 (BMI-T-6)
fuel element fabrication, 12 3301, 4824(R) (ANL-5717)
fuel element fabrication for Mark III core, 13 1349(R) (ANL-5790)
fuel element fabrication, 13 11819(R) (ANL-5975)
fuel element fabrication, 14 19308(R) (ANL 5709)
fuel element fabrication, 14 24465(R) (ANL-6099)
fuel element fabrication and thermal cycling tests for Mark III core, 14 19962 (ANL-6028)
fuel element fabrication, Mark-II 14 24521 (ANL-5724)
fuel element fabrication, 15 23908
fuel element leak testing, 15 21794
fuel element loading equipment, 11 13868(R) (ANL-5260(Del 2))
fuel element materials, 13 2967 (ANL-5797)
fuel element processing flowsheet, 12 750 (ANL-LB-SL-1016)
fuel element radiation damage, 11 7629(R) (ANL 5489)
fuel element radiation effects, 13 416 (ANL-5855)
fuel element remote injection casting, equipment, 15 21792
fuel element testing 13 2967 (ANL-5797)
fuel element testing by eddy current and sonic techniques, 13 13514 (ANL-5893)
fuel element thermal cycling, 12 887(R) (ANL-4399(Del))
fuel elements, fabrication for core IV, 15 32914(R) (ANL 6433)
fuel loss, effects on breeding gain and converting time 11 9788 (ANL LB SL 1053)

fuel meltdown in, 10 6997
fuel pin fabrication, 13 19215(R) (ANL-5837)
fuel recovery, 12 749(R) (ANL-5529(Del.))
fuel recovery, 12 17424 (TID-2506(Del Xp.39-55))
fuel recovery, 13 22144(R) (ANL-6029)
fuel recovery, 14 18796(R) (ANL-5668)
fuel-rod fabrication for Mark III core loading, 15 3047 (NMI-4801)
fuel slug dissolution procedures for, 10 4392 (CF-51-8-151)
gas disposal, 12 17056 (WASH-170(Del)) (p 63-77)
heat transfer-reactivity correlation, 13 11371
heat transfer studies, 11 9833 (KAPL-M-JZ-1), 11746(R) (ANL-5471)
history and status of, 10 3943 (AECD-3712)
instability mechanism, 15 21770
instability studies with Mark III, 15 1-522 (ANL-6266)
instrumentation, 14 14863 (ORNL-2695(p 67-85))
internal feedback of Mark III, 14 19963 (ANL-6124)
kinetic studies on, 15 3603 (ANL-6205(p 269-77))
kinetics, capture-to-fission ratio, 15 25734(R) (ANL-6333)
multigroup calculations, 12 976 (ANL-5492(Del.))
neutron capture and fission patterns of uranium-238 in, 13 1825
neutron economy, capture-to-fission ratios for uranium-235 and uranium-238 vs. position, 14 19008(R) (ANL-5730)
neutron flux distribution and shielding, 12 971(R) (ANL-5060(Del.))
neutron flux distribution, calculation of thermal, 12 15991 (WASH-174 (Del.))
neutron flux in radial shield, 13 17346
neutron flux measurements by gold and phosphorus activation, 13 12255 (ANL LAT-86)
neutron leakage, estimation, 13 12255 (ANL-LAT 86)
non-linear kinetics and stability of Mark II, analog study, 14 5453
operating characteristics, 6 5712
operating experience, 15 27217 (TID-13305)
operation, 12 1072 (KAPL-M-WRK-3(Rev)) (Del))
operation 15 10369(R) (ANL-6295)
operation, 15 12451(R) (ANL-6269)
operation and performance, 15 15275 (ANL-6301)
performance review, 14 16446 (KAPL M WMC 3)
plutonium production and uranium consumption rates, 11 7690(R) (ANL-5175), 13579(R) (ANL-5039(Del))
power variation, infinities in transfer function for, 11 12591 (TID-5352)
reactivity, Doppler temperature effect, 12 17756 (ANL-5809)
reactivity oscillation measurements, 11 7863 (TID-5351), 13488 (ANL-5782)
reactivity oscillation in Mark II and III, 15 30282
safety, 12 11950
safety mechanisms, 11 10921(R) (ANL-4238), 13763(R) (ANL-4316 (Del))
safety, operational stability, 13 16583
safety, radiochemical analysis of fuel assembly meltdown, 12 15069 (A/CONF.15/P/1549)
shielding, 11 7718(R) (ANL-4274), 9107 (NAVORD 4470), 9782 (ANL 4974), 10923(R) (ANL-5297)
stability, nonlinear treatment of large perturbation, 13 19748
stability studies on EBR-I, 12 15079 (A/CONF 15/P/1845)
stability studies, transfer function measurements, 15 3601 (ANL-6205 (p 232-49))
technical data and general information, 12 15904

EXPERIMENTAL BREEDER REACTOR-2 (EBR-2), U.S.A.

blanket processing, 15 17983(R) (ANL-6287)
breeding blanket processing, 14 16576(R) (ANL-6101)
component development and testing, 15 32913(R) (ANL 6409)
conference papers on, 12 5013 (TID-7548)
construction, 15 12451(R) (ANL-6269)
construction and component development, 15 25621(R) (ANL-6387)
construction and development 15 10369(R) (ANL 6295)
construction and engineering, 15 4708(R) (ANL-6253)
construction, development, and kinetics, 15 12452(R) (ANL-6307)
control system, 12 5677
coolant flow characteristics, 11 6521(R) (ANL-5657)
cooling system corrosion studies, 15 13272 (NASA-TN-D-769(p 25))
cooling system design, natural convection emergency 11 11748(R) (ANL-5571)
cooling system, steady-state characteristics of three loop design, 14 17590 (ANL 6137)
core meltdown, 15 15209(R) (ANL-6328)

EXPERIMENTAL BREEDER REACTOR-2 (EBR-2), U.S.A. (Continued)

design and design modifications since 1955, 12 15073 (A/CONF.15/P/1782)

design data, 14 9228

design description, 13 16589

design engineering, 12 1682

design, power cycle, 13 16580

development, 14 24465(R) (ANL-6099)

development, 15 2296(R) (ANL-6215)

development, 15 2297(R) (ANL-6234)

development, 15 17738(R) (ANL-6343)

development, 15 20375(R) (ANL-6355)

development, 15 23116(R) (ANL-6374)

development, 15 30196(R) (ANL-6399)

development and fabrication programs, 15 22692(R) (ANL-6330)

development and operation, 15 32914(R) (ANL-6433)

development and testing, 15 15209(R) (ANL-6328)

experimental facilities for irradiating fuel materials, 13 5113 (ANL-5940)

fission gas release from spent fuel in melt refining, 15 8897

fission product spectrum of spent fuel elements in, 10 4916 (ANL-5334)

fuel alloy irradiation behavior, 13 23056 (TID 7559(Pt 1)(Paper 18))

fuel alloy physical properties, 13 16603

fuel composition and preparation, 15 31852 (ANL 6290)

fuel-element assembly, design of remote-control equipment for, 15 4846 (TID-7599(p 429-40))

fuel element assembly, sodium bonding, and bond testing, 15 21795

fuel element cladding removal, description of machines for, 14 9578 (ANL-6106)

fuel element cladding development, 15 18482 (BMI-1513)

fuel element demolding, sizing, and inspection, equipment for remote, 15 21793

fuel element design, 13 9040

fuel element design, 13 16591

fuel element design concepts, 13 16604

fuel element fabrication, 13 16593

fuel element fabrication facility, shielding window design, 13 14359 (TID-7556(p.160-7))

fuel element fabrication methods, 13 23053 (TID-7559(Pt 1)(Paper 15))

fuel element fabrication, 15 23908

fuel element fabrication, cladding bonding with sodium annuli, 15 28008 (ANL-6384)

fuel element processing, 14 4397

fuel element processing, 14 16576(R) (ANL-6101)

fuel element reclamation, pyrometallurgical process and plant for, 12 15014 (A/CONF 15/P/541)

fuel element remote refabrication, 15 21791

fuel element reprocessing, 14 21559(R) (ANL-6145)

fuel element testing, 15 16761

fuel element testing in TREAT facility, 15 32911 (ANL-6334)

fuel elements, flowsheets for Dow aluminum pyrometallurgical processing, 15 26115(R) (TID 11692)

fuel-elements radiation effects, prototype, 15 3147 (ANL-6010)

fuel loading facility, 15 27065 (ANL-6092)

fuel pin meltdown tests, 14 17622

fuel processing cycle, 13 16592

fuel processing, design of facilities, 15 25734(R) (ANL-6333)

fuel processing, facilities development, 15 12649 (ANL-6231)

fuel recovery, 15 17983(R) (ANL-6287)

fuel refabrication cycle, 13 16581

fuel reprocessing system, 15 5825 (NP-9340(Vol IV)(Sect II))

fuel reprocessing, 15 6043 (ANL-6211)

fuel reprocessing, 15 5851(R) (ANL-6183)

fuel reprocessing by melt-refining, plutonium behavior in, 15 8896

fuel reprocessing, fission product behavior during melt refining, 15 8902

fuel reprocessing iodine volatilization as metallic iodides in melt refining, 15 8901

fuel reprocessing sorption of cesium and sodium vapor from, melt refining, 15 8899

fuel reprocessing, zirconium separation in melt refining, 15 8900

fuel reprocessing behavior of barium, cesium, rare earths, strontium, and yttrium in melt refining, 15 8898

fuels, evaluation of carbides, 14 15894(R) (NDA-2140 2)

instrumentation, 14 14863 (ORNL-2695(p 67-85))

kinetic studies on, 15 3603 (ANL-6205(p 269-77))

kinetic studies, thermal feedback, 15 3604 (ANL-6205(p 278-9))

mockup in ZPR-III, 15 15275 (ANL-6301)

performance, review, 14 16446 (KAPL M-WMC 3)

physics design problems, 15 30292 (ANL-6383)

primary tank support structure, design, 12 11941

reactivity, predicted dynamic behavior, 12 15007 (A/CONF.15/P/458)

review of, 9 3294

safety, containment system description and evaluation, 12 15090 (A/CONF 15/P/1892)

safety hazards summary report, 12 4430 (ANL-5719)

sodium system, 5000-gpm electromagnetic and mechanical pumps, 12 5676

technical data and general information, 12 15904

FAST REACTORS

accidents in, dynamic simulation of, 11 6810

accidents in, study methods for high-hazard, low-probability, 11 6089 (NDA-14-107)

analysis by Monte Carlo calculations, 13 18222 (CWR-900-13-4)

bibliography of reports on, 14 17578 (NAA-SR-Memo-5054)

bibliography on fuel breeding in, 13 19740 (TID-3534)

blanket design for breeder, 15 15228

breeding ratio in, sensitivity to cross-section uncertainties, 14 18674 (ANL-6122(p 93-100))

calculations on, effects of flux depression in scattering resonances, 15 21743

characteristics of breeder, 15 10554

characteristics of, comparison with thermal reactors, 15 4772 (WADC-TN-59-415)

computational survey of idealized breeder, 14 18673 (ANL-6122 (p 77-92))

computational survey of idealized breeder, 15 7110

control of, 11 13094 (LWS-24898(Del))

control and safety of, 11 13879 (APDA-105)

control rod position indicator for, 13 9367 (IGR-TN/CA-958)

coolants for, comparison, 11 8682 (LWS-24701)

core design for, 15 21813

criticality calculations for, 15 17763

critical mass of, calculation of, 13 22213

critical mass of plutonium spherical, effects of alloying on, 13 130

cross sections for, estimation of one-group, 14 3140 (IGR-209 (RD/R))

description of sodium-cooled breeder, 13 14066 (NP-7383)

description of Soviet BR-1, BR-2, BR-3, BR-4, and BR-5, 15 7057

description of Soviet BR-1, BR-2, BR-5 programs, 13 7175 (A/CONF 15/P/2129)

description of Soviet BR-2, 14 17628

description of Soviet BR-5, 14 1125

design, 12 7026

design of breeder, 14 26494

design of breeder, 15 13961

design of breeder, 15 15261

design of breeder, 15 23145

design, 15 23146

design calculations for breeder, 12 996 (CF-52-8-230)

design of breeder, 12 991 (CF-3582)

design of molten fuel breeder, 11 13957 (LWS-22710)

design, numerical methods for, 15 1013 (NP-tr-482)

design of oxide-fueled breeder, 15 2332 (GEAP-3347)

design of, physics data for, 15 30195 (ANL-6382)

design of sodium-cooled breeder, 13 983

design of sodium-cooled breeder, 14 11351

design of sodium-cooled, 15 25644 (NAA-SR-Memo-5996)

design of sodium-cooled, 14 18637 (NAA-SR-Memo-5011)

design of, stability conditions in, 13 7151 (A/CONF 15/P/1462)

development in U S S R, 12 12767

development of breeder, 14 14593

development of plutonium breeder, 10 12042

development of Soviet BR-5, 15 32987 (NP-10766(p 32-5))

development status of breeder, 13 16576

Doppler coefficient for large sodium-cooled breeder, 15 30254 (GEAP 3646)

Doppler effect in, 14 23681 (APDA-139)

Doppler effect in, estimation of, 10 10531

Doppler effect in, estimation of, 10 11467 (AERE-T/M-137)

Doppler reactivity effect in, 12 4461 (NDA-14-127)

Doppler reactivity effect in oxide-fueled, 15 21740

Doppler temperature coefficient in, 12 15072 (A/CONF 15/P/1777)

FAST REACTORS (Continued)

economics of breeder, 12 2543
economics of breeder, 13 7270 (A/CONF 15/P/2028)
economics of sodium-cooled breeder, 11 6102
economics of sodium-cooled breeder, 15 12662 (TID-8533)
excursions in, AX-I program for calculating, 13 17410
fission product accumulation in, 13 10588 (RDB(C)/TN-48)
fuel alloys for breeder, 15 1908
fuel and blanket reprocessing for breeder, 14 13793
fuel element material for breeder, 14 25873
fuel elements for sodium-cooled breeder, oxide, 13 3432
fuel element stresses in, 13 16224 (RDB(C)/TN-66)
fuel handling and processing for, 13 16595
fuel reprocessing economics, 12 1685
fuel reprocessing for, 11 7864
fuel reprocessing for, 15 4824 (TID-7599(p 185-98))
fuel reprocessing for 500-MW breeder, 14 13791
fuel systems for breeder, 13 18725 (BMI-APDA-636)
fuel systems for breeder, 13 18726 (BMI-APDA-641)
fuel systems for breeder, 13 18727 (BMI-APDA-643)
kinetics and stability of, 13 10708
kinetic behavior of, 8 6306
literature review of, 13 4237 (APEX-442)
maintenance, 11 10280
materials for, 14 25999
meltdown studies of, 13 16584
multigroup albedo calculations for, 15 23068 (DEG-Report-96)
multigroup calculations for, HASSITT program for, 14 25004 (AERE-M-597)
multigroup calculations on, 14 14544
neutron calculations for reflectors of, 10 2507 (A-4257)
nuclear calculations for sodium-cooled, 12 2082 (KAPL-908(Del))
neutron spectra in, 11 12210 (AEC-4244)
oxide fuel design for breeder, 12 655 (KAPL-1784)
physics calculations for, 11 11797 (KAPL-M-RE-8)
physics calculations for, 13 10633 (IGR R/R-221)
physics calculations for breeder, 12 976 (ANL-5492(Del))
physics calculations for sodium cooled, 12 1031 (KAPL-131)
physics of, 11 1333
physics of, 13 6929 (A/CONF 15/P/2038)
physics of breeder, 10 1029 (ANL-5491)
physics calculations of breeder, 15 11912
physics of, status report on, 12 15037 (A/CONF 15/P/637)
poisoning by fission products, 11 7840 (LWS-24612)
reactivity coefficients and stability of, 15 17749 (NDA-2147-3)
review of programs for, 14 12330
review of Soviet BR-1, BR-2, and BR-3 programs, 13 3395
safety aspects of, 10 4917 (ANL 5577)
safety calculations for, AX-I program for, 13 17409
safety features of breeder, 11 4110
safety of sodium-cooled, 17 23766
safety studies for breeder, 14 1112 (NRL-Memo-819)
safety studies of breeder, 14 1113 (NRL-Memo-975)
safety studies of, mathematical analysis of equations in, 10 1560 (NDA-14-58)
stability studies of, with delayed reactivity coefficients, 15 28741 (NDA-2147-7)
status report on, 14 21147 (TID-8518(Bk I))
study of 300 MW(e) sodium-cooled breeder, 15 13977 (GEAP-3287)
technology lecture on, 10 5385 (M-3753)
test installation for, 14 25013 (LAMS-2433)
theory of, 11 6899
theory of, 15 7034 (CEA-tr-A-804)
theory of sodium-cooled breeder, 11 13138 (CF-55-6-157)
thermal fluxes in, 13 7189 (A/CONF 15/P/2411)
welding processes for breeder, 13 18140

FERMI FAST BREEDER REACTOR (EFFBR), U.S.A.

analysis of power feedback relations, 15 3602 (ANL-6205(p 251-68))
analytical studies, design data, and research and development results, 13 481
atmosphere monitoring, inert gas system hazards analysis, 12 12724 (KLX-1782)
auxiliaries reactor building ventilation, 13 12290 (AECU-4098 and Add 1)
auxiliaries strength of pressure vessel 13 8340
bibliography 14 13504 (CEA-BIB 5)

bibliography, 14 21144 (TID-3556)
bibliography, 15 24592 (TID-3556(Rev 1))
blanket and fuel element development, 15 23067 (APDA-143)
breeding blanket subassembly model, 12 12685 (AECU-3757(Pts 1 and 2))
breeding blankets and fuel element design, 12 8709 (AECU-3678)
breeding element fabrication, 12 7838 (BW-7065)
containment study, 12 5650 (NAVORD-5747)
containment-vessel design, 15 17788
control and reactivity studies, 13 10716 (AECU-4061)
control rods effects of movement on reactivity of, 13 12265 (NDA-014 189)
control-system simulation, 15 13999 (TID 11025)
coolant radioactivity radiation levels in equipment compartment, 13 10717 (AECU-4063)
core design and fuel element development, 13 16641(R) (PRDC-TR-20)
core melt-down due to coolant loss, 12 12730 (NDA-14-170)
core meltdown, 15 15209(R) (ANL-6328)
critical experiments, 12 16676 (AECU-3815)
criticality calculation for uranium 235, 13 9479(R) (PRDC-TR 17)
criticality during hypothetical slow meltdown accident, calculated, 12 8735 (NDA-14-182)
criticality studies, enrichment and mass, 13 23101 (APDA-131)
danger coefficients, 12 13474 (NDA-014-167)
description, 15 30285
description of plant, 13 14054 (APDA-124)
design, 11 5608, 6103
design and construction, 12 15081 (A/CONF.15/P/1850)
design and development, 11 7727 (APDA-102)
design and development, 12 7461(R) (APDA-123), 8048(R) (AECU-3652, AEC-3653, and AECU-3654), 8077(R) (PRDC-TR-4, PRDC-TR-5, and PRDC-TR-6), 8078(R) (PRDC-TR-7), 8079(R) (PRDC-TR-8)
design and development, 11 11807(R) (TID-10097)
design and development, 13 477(R) (PRDC-TR-13)
design and safety features, 11 2104 (APDA-115)
design and site considerations, 11 6887 (APDA-108)
design data, 12 12780
design data, 14 9228
design description, 13 16590
design, hazards summary, and operational survey, 15 25647 (NP-10458)
design of reactor and turbine generator sections, 12 15822 (AECU-3816)
design of reactor and power plant, 13 16582
development, 12 12733(R) (PRDC-TR-10), 12734 (PRDC-TR-11), 15851 (PRDC-TR-12)
development, 12 15851 (PRDC-TR-12)
development, 11 13989 (TID-10077)
development, 13 9479(R) (PRDC-TR-17)
development, 13 17566(R) (PRDC-TR-21)
development, 13 20712(R) (PRDC-TR-23)
development, 13 20713(R) (PRDC-TR-24)
development, 13 21783(R) (PRDC-TR-22)
development, 14 2222(R) (PRDC-TR-25)
development, 14 2223(R) (PRDC-TR-26)
development, 14 3236(R) (PRDC-TR-27)
development, 14 22611(R) (PRDC-TR-33)
development, 15 24590(R) (PRDC-TR-44)
development and environs monitoring, 14 12356(R) (PRDC-TR-28)
development and environs monitoring, 14 12357(R) (PRDC-TR-29)
development and environs monitoring, 14 16458(R) (PRDC TR-30)
development and environs monitoring, 14 23758(R) (PRDC TR-34)
development and environs monitoring, 15 7091(R) (PRDC-TR-38)
development and environs monitoring, 15 15291(R) (PRDC-TR-39)
development and environs monitoring, 15 15292(R) (PRDC-TR-40)
development and environs monitoring, 15 15293(R) (PRDC-TR-41)
development and status of construction, 13 5958(R) (PRDC TR-16)
development and status of construction, 13 14067(R) (PRDC-TR-19)
development, December 1960, 15 17836(R) (PRDC-TR-42)
development for August 1958, 13 4274(R) (PRDC TR-14)
development for Dec , 1958 13 9480(R) (PRDC TR-18)
development for Feb 1959, 13 16641(R) (PRDC-TR-20)
development for January 1961, 15 20428(R) (PRDC-TR-43)
development for June 1960, 15 1064(R) (PRDC-TR-36)
development for May 1960, 15 1063(R) (PRDC TR 35)
development highlights to Nov 1957, 13 16579
development report for September 1958, 13 4275(R) (PRDC-TR-15)
developments from Aug. 1957 to March 1959, 14 2191(R) (AECU-4384)

FERMI FAST BREEDER REACTOR (EFFBR), U.S.A. (Continued)

electrical features, 14 6053
 environmental survey of site, 14 12806
 environs monitoring, 13 5958(R) (PRDC-TR-16)
 environs monitoring, 13 9479(R) (PRDC-TR-17)
 environs monitoring, 13 14067(R) (PRDC-TR-19)
 environs monitoring, 13 17566(R) (PRDC-TR-21)
 environs monitoring, 1958, 13 9879 (AECU-4017)
 environs monitoring, preoperational, 13 4275(R) (PRDC-TR-15)
 environs monitoring, 14 3236(R) (PRDC-TR-27)
 flat slab freezing with radioactive materials having internal heat generations, 13 8285 (NDA 14-178)
 fuel, economic feasibility and nuclear characteristics of sponge, 14 14009 (NDA-2116-4)
 fuel element cooling during transfer to storage, 13 10714 (AECU-4058)
 fuel element design, 13 16591
 fuel element design concepts, 13 16604
 fuel element dissolution for reprocessing, 14 25538(R) (DP-479)
 fuel element fabrication, 12 15047 (A/CONF 15/P/792)
 fuel element fabrication for core B, 15 29716(R) (ORNL-3160)
 fuel element flow characteristics, model studies, 12 11946
 fuel element handling and storage facility, design, 13 10718 (AECU 4064)
 fuel element, processing, 14 8511
 fuel element radiation stability, 12 14816 (A/CONF 15/P/622)
 fuel element reclamation, 13 13361 (BNL-549)
 fuel element testing, subassembly nozzle test in hot sodium, 12 7456 (AECU-3658)
 fuel element testing, 15 16761
 fuel element testing in TREAT facility, 15 32911 (ANL-6334)
 fuel elements, fabrication of simulated Core B, 15 27067 (CF-61-4-71)
 fuel elements, problem of low-melting-point eutectics at fuel-cladding interface, 12 12894 (NDA-Memo-14-65)
 fuel handling instrumentation 13 9479(R) (PRDC TR 17)
 fuel-pin end-capping, 13 9055
 fuel pin melt-down tests, 12 14252 (NMI-4402)
 fuel pin radiation stability, 15 18567
 fuel pin thermal cycle test, 12 7454 (AECU-3648)
 fuel pins, development and testing, 14 19496 (APDA-130)
 fuel plate cladding, survey of potential materials, 14 19314 (CF-60-4-118)
 fuel plates, fabrication of uranium dioxide-stainless steel composite, 15 17278 (ORNL-3077)
 fuel recovery, 12 16999(R) (BNL-511)
 fuel recovery, 13 19983(R) (DP-393)
 fuel reprocessing at Savannah River Plant, 14 6334 (TID-7583(p 174-86))
 fuel reprocessing at Savannah River Plant, 14 6336 (TID-7583(p 195-205))
 fuel reprocessing plant, 14 13791
 fuel subassemblies thermal bowing 13 10721 (APDA 128)
 fuel subassembly evaluation for Core II, 13 18725 (BMI-APDA-636)
 fuel subassembly evaluation for Core II, 13 18726 (BMI-APDA-641)
 fuel subassembly evaluation for Core II, 13 18727 (BMI-APDA-643)
 fuels, evaluation of carbides, 14 15894(R) (NDA 2140 2)
 gas supply system description of inert, 13 12288 (AECU 4096)
 gas system, basic concept for the inert, 13 12286 (AECU-4094)
 group theory, comparison of one- and two-dimensional calculations, 13 473 (NDA-14-163)
 hazard summary, 11 6089 (NDA-14-107)
 hazards summary, 12 7460 (APDA-120)
 hearings on construction permit, 13 16140
 heat exchange system, 13 2589
 heat exchanger design, 15 19523
 instrumentation, testing specification for non-nuclear, 14 3235 (NP-8052)
 kinetics of fuel freezing after core meltdown, 13 8286 (NDA 14 181)
 loading nuclear aspects of fuel and blanket, 13 10716 (AECU 4061)
 neutron fluxes and adjoints as functions of position and energy, calculations, 12 12729 (NDA-14-153)
 neutron-source requirement for subcritical reactor, 15 3615 (NDA-Memo-14 114)
 power limiting system design, 13 5945 (BAC/RL-1052)
 power-limiting system, analog study of conceptual design, 14 5450
 power limiting system design analysis, 15 31868 (NP-10699)
 radiation levels in containment vessel, 14 16468 (TID-6032)
 radiation monitoring system, in plant 14 11328 (TID 5686)

reactivity, Doppler temperature coefficient 14 23681 (APDA 139)
 safety, 11 6887 (APDA-108)
 safety features, 11 4110
 safety instrumentation for the power plant, 12 11032
 safety, internal temperatures during coolant flow reversal, 13 5953 (NDA-14-179)
 safety, low probability-high hazard accidents, 11 6089 (NDA-14-107)
 shield design, primary, 14 18649 (TID 6033)
 shielding materials selection and properties 13 3403
 shielding, primary 12 12040 (AECU 3701), 12863 (BMI APDA 622)
 sodium-air reaction 13 14924 (AECU-4161)
 sodium removal from core subassemblies, 15 17825 (APDA-142)
 technical data and general information, 12 15904
 transient behavior, simulator study, 11 2103 (APDA-111)
 transient conditions and transfer functions, analog computer simulation of, 15 31869 (NP-10700)
 trip-level settings analysis for power limiting system, 15 28814 (NP-10550)
 vapor formation effects 12 10968 (AECU 3697)
 waste handling, 15 21904

KAPL PLUTONIUM POWER BREEDER REACTOR, U.S.A.

design and cost factors, 12 2058 (CEPS-1121(Del))
 design of chemical processing plant for oxide fuel elements 11 13005 (KAPL M JKD-8)
 development, 14 1135 (KAPL 1701)
 fuel cycle and cost, 13 5183
 fuel cycle, cost factors and estimates, 12 14247 (KAPL-1757)
 fuel element assembly, 12 944 (KAPL-1302)
 fuel element assembly, 11 12556 (KAPL-M-EAL-16)
 fuel element design, effect of fission gases on internal pressure for 12 1065 (KAPL M-RSM 3)
 fuel element design concepts, 13 16604
 fuel element fabrication, 12 5639 (KAPL-1837)
 fuel element processing, 14 17644(R) (KAPL-1536)
 fuel recovery in Purex Process radiation factors 12 780 (KAPL-M SSJ 3)
 fuel recovery, radiation effects in processing, 12 4118 (KAPL 1758)
 fuel-rod design, stress considerations in, 15 2332 (GEAP 3347)
 mathematical analysis, 12 4441 (KAPL-1756(Pts 1&2))
 multi-group analysis of, 12 3238 (KAPL M JBS-9)
 multigroup calculations and criticality studies, 14 10873(R) (KAPL 1210)
 program, review of, 14 16446 (KAPL M WMC 3)
 safety, maximum nuclear explosion estimate, 12 2081 (KAPL 294)

LIQUID METAL COOLED REACTORS

control and dynamics performance of sodium-cooled, 14 14555
 economics of sodium-cooled fast breeder, 11 6102
 evaluation of breeder, 14 14556
 description of sodium fast breeder, 13 14066 (NP 7383)
 design of sodium fast breeder, 13 983
 development of sodium, 12 10083
 fuel-cycle costs for sodium-cooled fast, 15 21839 (NAA SR 5190)
 fuel element cladding for, corrosion and heat transfer of globeiron for 9 2776
 fuel elements for sodium fast breeder, oxide 13 3432
 fuel subassembly nozzle test in hot sodium, 12 7456 (AECU-3658)
 heat exchanger design for 15 12453 (APAE 78(Vol IV))
 heat exchanger design for sodium-cooled, 14 15443 (APAE-41 (Vol I))
 heat exchanger design for sodium-cooled, 14 15444 (APAE-41 (Vol II))
 heat exchanger design for sodium-cooled, 14 15445 (APAE-41 (Vol III))
 heat exchanger design for sodium-cooled, 15 11119 (APAE 78 (Vol II))
 heat exchanger design for sodium cooled, 15 11120 (APAE 78 (Vol III))
 heat exchangers and steam generators for sodium-cooled, 15 19515 (TID-6881)
 heat transfer in, 14 10235
 literature review of fast breeder, 13 4237 (APEX-442)
 review of Soviet BR-1, BR 2, and BR 3 programs, 13 3395
 sodium-air accident study for, 13 14924 (AECU-4161)
 sodium cooling of fuel assemblies in, 12 15897
 sodium-water reactions in, 13 4274 (PRDC-TR 14)

LIQUID METAL COOLED REACTORS (Continued)

test facility for sodium components 14 10545 (APDA 134)
 testing of sodium components for, 12 12733 (PRDC TR 10)
 testing of sodium components for 12 12734 (PRDC TR 11)
 theory of sodium fast breeder 11 13138 (CI 55-6 157)

LIQUID METAL PUMPS

design of isothermal graphite, 10 4262 (AECD 3831)
 development for EBR II, 11 6521 (ANL 5657)
 evaluation for sodium cooled fast breeder reactors, 13 8863

LOS ALAMOS FAST REACTOR (CLEMENTINE), U.S.A

design, control operation, and nuclear characteristics, 11 12598
 (TID 10048)
 disassembly, 8 6551
 disassembly, 11 12570 (LA-1575)
 effects of temperature and reactivity changes in operation of, 12 1076
 (LAMS 733(Del))
 fuel processing, 12 801 (ORNL-1941(Del))
 gamma spectra, 5 5362 (LADC 961)
 gamma spectra, 6 998 (LADC 1060)
 neutron beam energy spectra, 12 2042 (LA-1234)
 neutron energy spectrum 13 13708 (LA 2185)
 neutron flux distribution, Monte Carlo calculation 11 7707 (ANL-4284)
 neutron flux distribution, 11 13952 (LA 1192)
 processing of core and blanket, aqueous, 12 1889 (LWS-22679(Del))
 reflectors failure by rupture 14 4499 (ANL-6048)

ZEPHYR REACTOR, U.K.

breeding with thorium envelope 11 7321 (AERE R R 2151)
 criticality of, 9 2445 (AERE RP/M-46)
 delayed neutron fraction in, measurement, 12 17753 (AERE-R/M-174)
 description, 11 6930
 design and performance, 11 2130
 design and uses, 11 2135
 neutron effectiveness as function of energy in natural uranium envelope
 11 9465 (AERE R/R-2150)
 neutron flux distribution in core II, 11 635 (AERE-R/R-1949)
 neutron flux measurements, 11 4661 (AERE-RP/M-49)
 neutron lifetime measurements, 14 3213
 nuclear parameters, measurement, 13 7082 (A/CONF 15/P/39)
 operational features of, 9 753
 operational features of, 9 1083
 oscillator experiments on 15 3598 (ANL-6205(p 201-13))
 pneumatic transfer system for delayed neutron experiments 11 4985
 reactivity changes from insertion of various samples, 11 4660 (AERE
 RP/M 48)

ZEUS, U K

criticality of, 10 10502 (AERE-RP/M-81)
 description of, 10 9615
 design and uses, 11 2135
 neutron lifetime measurements 14 3213
 nuclear parameters, measurement, 13 7082 (A/CONF 15/P/39)
 oscillator experiments on 15 3598 (ANL 6205(p 201-13))
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119

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LA 2185	.60	2.39	NP 5997	.30	.91
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121

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RDBC/TN 74	.30	1.37	TID 10097	.89	5.47
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