**BIBLIOGRAPHY of REPORTS
on
RESEARCH
sponsored by the NRC
Office of Nuclear Regulatory Research
July - December 1977**

J. R. Buchanan

*Report of the Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, D.C. August 1978. NRC-145-1*

NUCLEAR SAFETY INFORMATION CENTER

NSIC

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FOREWORD

The Nuclear Safety Information Center (NSIC), which was established in March 1963 at Oak Ridge National Laboratory, is principally supported by the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research. Support is also provided by the Division of Reactor Development and Demonstration of the Department of Energy. NSIC is a focal point for the collection, storage, evaluation, and dissemination of safety information to aid those concerned with the analysis, design, and operation of nuclear facilities. Although the most widely known product of NSIC is the technical progress review *Nuclear Safety*, the Center prepares reports and bibliographies as listed on the inside covers of this document. The Center has also developed a system of key words to index the information which it catalogs. The title, author, installation, abstract, and key words for each document reviewed are recorded at the central computing facility in Oak Ridge. The references are cataloged according to the following categories:

1. General Safety Criteria
2. Siting of Nuclear Facilities
3. Transportation and Handling of Radioactive Materials
4. Aerospace Safety (inactive ~1970)
5. Heat Transfer and Thermal Hydraulics
6. Reactor Transients, Kinetics, and Stability
7. Fission Product Release, Transport, and Removal
8. Sources of Energy Release under Accident Conditions
9. Nuclear Instrumentation, Control, and Safety Systems
10. Electrical Power Systems
11. Containment of Nuclear Facilities
12. Plant Safety Features - Reactor
13. Plant Safety Features - Nonreactor
14. Radionuclide Release, Disposal, Treatment, and Management (inactive September 1973)
15. Environmental Surveys, Monitoring, and Radiation Dose Measurements (inactive September 1973)
16. Meteorological Considerations

17. Operational Safety and Experience
18. Design, Construction and Licensing
19. Internal Exposure Effects on Humans Due to Radioactivity
in the Environment (inactive September 1973)
20. Effects of Thermal Modifications on Ecological Systems
(inactive September 1973)
21. Radiation Effects on Ecological Systems
(inactive September 1973)
22. Safeguards of Nuclear Materials

Computer programs have been developed that enable NSIC to (1) operate a program of selective dissemination of information (SDI) to individuals according to their particular profile of interest, (2) make retrospective searches of the stored references, and (3) produce topical indexed bibliographies. In addition, the Center Staff is available for consultation, and the document literature at NSIC offices is available for examination. NSIC reports (i.e., those with the ORNL/NSIC and ORNL/NUREG/NSIC numbers) may be purchased from the National Technical Information Service (see inside front cover). All of the above services are free to NRC and DOE personnel as well as their direct contractors. They are available to all others at a nominal cost as determined by the DOE Cost Recovery Policy. Persons interested in any of the services offered by NSIC should address inquiries to:

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Telephone 615-483-8611, Ext. 3-7253
FTS number is 850-7253

PARTS AND METHOD OF INDEXING ABSTRACTS

BIBLIOGRAPHY OF REPORTS ON RESEARCH SPONSORED BY
THE NRC OFFICE OF NUCLEAR REGULATORY RESEARCH
JULY-DECEMBER 1977

J. R. Buchanan

ABSTRACT

A bibliography of 198 reports published by contractors of the NRC Office of Nuclear Regulatory Research during the period July through December 1977 is presented along with abstracts from the Nuclear Safety Information Center computer file. The bibliography has been sorted into the subject categories used by NRC to organize the research program. Within the subject categories, the reports are arranged first by contractor organization and then chronologically. A brief description of the NRC research program precedes the bibliography.

INTRODUCTION

The Energy Reorganization Act of 1974 provided for an Office of Nuclear Regulatory Research within the Nuclear Regulatory Commission (NRC) to perform research, characterized as "confirmatory assessment," relating specifically to regulatory decisions for the safe and environmentally compatible operation of nuclear facilities and materials as well as their protection. In implementing these research responsibilities, NRC continued the two primary programs of reactor safety research already being carried out by the former U.S. Atomic Energy Commission — light-water reactor safety research and advanced reactor safety research — and initiated the planning and coordination needed to accomplish regulatory environmental, fuel cycle, and safeguards research programs.

The goal of the Reactor Safety Research Program is to develop an independent basis and means to reliably and credibly analyze the course of events in hypothetical nuclear reactor accidents and to estimate the consequences of such accidents. Sufficient safety data exist to permit establishment of conservative requirements and safety margins for licensing nuclear power plants. NRC reactor safety research is directed toward refining and reducing the allowable uncertainties in the data, in order

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to better define and quantify the conservative design and safety margins that must be used because of these uncertainties.

This bibliography of NRC safety research reports has been sorted into the various categories of research being performed by NRC. A brief description of these categories is given below, followed by the bibliography.

WATER REACTOR SAFETY RESEARCH

Water reactor safety research is directed toward providing a capability for independent confirmatory assessment of the safety of the current generation of nuclear plants under postulated accident conditions. The research data and analytic methods applied to the assessment of hypothetical nuclear plant accidents will result in a greater measure of confidence that the margins of safety identified in the licensing review are well defined and quantified. The program is divided into five subject categories: systems engineering, fuel behavior, analysis development, metallurgy and materials, and site safety research.

Basic (R1, formerly NRC-1)

This category is included since it is used for the distribution of reports that pertain to all the water reactor safety research categories. However, no research per se is sponsored under this heading, although certain general-interest reports are distributed under this category.

Systems Engineering (R2, formerly NRC-2)

Safety research in systems engineering is addressed primarily to the study of postulated loss-of-coolant accidents (LOCAs) in reactors and the effectiveness of emergency core-cooling systems (ECCSs). Reports distributed under this category cover such research topics as the hydrodynamics of two-phase flow during the postulated LOCA, blowdown heat transfer, emergency core cooling (including alternate ECCSs) and integral LOCA/ECCS tests.

Fuel Behavior (R3, formerly NRC-3)

Fuel behavior research includes basic studies on the constituents of the fuel rod (fuel, gap, and cladding), studies on integral fuel rods,

and definitions of fuel failure limits and consequences. Reports issued under this category include studies on cladding properties, gap conductance, fuel-stored energy, decay heat, fuel pellet properties, steady-state and transient fuel rod performance, fission-product release, fuel meltdown phenomena, and fuel behavior computer codes.

Analysis Development (R4, formerly NRC-4)

Analytical development research includes improvement of existing reactor safety computer codes and development of codes for advanced reactor systems and for reactor components. Reports distributed under this category include information on intermediate and advanced system computer codes for studying the response of a nuclear power plant to postulated accidents and information on component code development in which the various single components of a reactor are modelled in greater detail. Reports on containment, hydroelastic, and fuel codes are also distributed under this category.

Metallurgy and Materials (R5, formerly NRC-5)

The objective of confirmatory safety research in reactor metallurgy and materials is to confirm the safe design of reactor vessels and piping and to establish ways to reduce the failure probabilities, if required. Reports in this category include those dealing with fracture mechanics, welding, irradiation embrittlement, stress corrosion cracking, crack growth and arrest, and nondestructive examination techniques.

Site Safety Research (R6, formerly NRC-6)

The site safety research program provides information to assist in the confirmation that nuclear power plant sites (including alternate sites) and the associated engineering design methodology have been properly characterized with regard to the effects of earthquakes, tornadoes, floods, and other natural phenomena. Reports in this category include information on severe regional environmental phenomena; understanding of seismic, hydrologic, and meteorologic events; methodology for geotechnical, hydrological, and meteorological site evaluations; assessment of engineering design methods and practices; and evaluation of alternate siting concepts.

ADVANCED REACTOR SAFETY RESEARCH

High-priority experimental and analytical safety research programs were initiated in 1974 on advanced reactors - liquid-metal-cooled (LMFBR) and gas-cooled (GCFBR) fast breeder reactors and high-temperature gas-cooled reactors (HTGR).

Fast Reactors (R7, formerly NRC-7)

The fast reactor program, which principally covers LMFBRs, gives strong consideration to the methods and data needed to assess the safety of fast reactors under a range of postulated accidents, from the anticipated loss of flow with scram to the hypothetical case of a core-disruptive accident such as might occur following a loss of flow without scram, including the challenge to the containment of postaccident environmental conditions.

The radiological source used in LMFBR site assessment is based on the potential leakage of coagglomerated aerosols of sodium oxide and uranium-plutonium oxide from the containment. Consequently, there is an experimental program to verify the source and mode of transport of aerosols generated in postulated LMFBR accidents.

Gas-Cooled Reactors (R8, formerly NRC-8)

In the case of the HTGR, the range of accidents includes the sudden depressurization (analogous to a pipe break in a light-water reactor) of the primary system, steam ingress from leaks in the steam generator, loss of forced circulation, and combinations of these events. Because of the unique structure of HTGRs, attention is also devoted to seismic response.

Studies of fission-product chemistry and graphite oxidation in laboratory loops are under way. These studies will lead to increased accuracy in the assessment of fission-product transport in HTGRs and of safety margins associated with vital structures such as the core support posts.

Those components and processes in the GCFBR which are similar to the HTGR (i.e., ex-core phenomena) are included in this category.

OTHER RESEARCH

Other important areas of research are generally relevant to any type of reactor system and to various parts of the nuclear fuel cycle.

Health Safety Research (RH, formerly NRC-9)

This category includes information on (1) studies on the technical aspects of NRC health policies and programs; (2) development of improved methods and procedures for licensing review, inspection, and enforcement; and (3) studies leading to improved regulations and guides to ensure implementation of effective health policies at licensed facilities.

Environmental Research (RE, formerly NRC-10)

This category includes information on (1) studies on the technical aspects of NRC environmental policies and programs; (2) development of improved methods and procedures for licensing review, inspection, and enforcement; and (3) studies leading to improved regulations and guides to ensure effective environmental implementation at licensed facilities.

Transportation Safety Research (RT, formerly NRC-12)

This category includes information on (1) general transportation operations and studies, (2) analyses which define the transportation environments, (3) tests to confirm the capabilities of shipping containers and their response to both normal and abnormal transport environments, (4) evaluations of the effectiveness of emergency planning and response to transport accidents, (5) studies which estimate and evaluate the risks involved with radioactive material shipments, and (6) evaluations which compare the degree of safety achieved by alternate transport methods and procedures.

Safeguards Research (RS, formerly NRC-13)

This category includes information on methods and data, as well as devices and techniques, which support the NRC functional capabilities in safeguards rule-making, licensing, and inspection. In particular, there is information on (1) the effectiveness of evaluation methods which aid

in the assessment of physical protection and material control subsystems at fixed sites and in transport; (2) analysis of safeguards information requirements; (3) physical protection equipment and its evaluation; (4) topics of broad safeguards interest relevant to future directions, including estimation of consequences of malevolent nuclear acts, safeguards vulnerability to white-collar crime, and impacts of performance-oriented regulations, among others.

Effluent and Radiation Field Source Terms in Operating Reactors (RR)

This category includes information on all testing in operating nuclear power plants to determine (1) the radioisotope content of liquid and gaseous streams within the plant and the performance of plant equipment for evaluation of analysis methods used for determining nuclear plant effluents; and (2) source terms and radiation fields associated with plant components and the cubicles within which they are contained for evaluation of current analysis approaches.

Criticality (RC)

This category includes information on all analyses and testing related to nuclear criticality. These include (1) experiments and analyses related to storage and shipment of reactor fuels, both as-built fuel and spent fuel; (2) experiments and analyses related to all aspects of the fuel cycle, such as fuel fabrication and reprocessing plants; (3) development of computer programs for analysis of nuclear criticality and comparison of results of these codes with experimental data; and (4) development of cross-section data useful to nuclear criticality analyses.

ORGANIZATION OF BIBLIOGRAPHY

The bibliography which follows contains all reports published from July through December 1977 by NRC contractors that are sponsored in the program described above. Reports that were published earlier but issued during this period are also included. The bibliography is organized by the category regime used for NRC report distribution. If a document falls into more than one category because of its subject matter, the complete abstract is included in each applicable category. Within each subject category, the reports are listed alphabetically by contractor name and then chronologically under each contractor. The bibliography is sorted into the following categories:

1. R1 - Water Reactor Safety Research, Basic
2. R2 - Water Reactor Safety Research, Systems Engineering
3. R3 - Water Reactor Safety Research, Fuel Behavior
4. R4 - Water Reactor Safety Research, Analysis Development
5. R5 - Water Reactor Safety Research, Metallurgy and Materials
6. R6 - Water Reactor Safety Research, Site Safety Research
7. R7 - Advanced Reactor Safety Research, Fast Reactors
8. R8 - Advanced Reactor Safety Research, Gas-Cooled Reactors
9. RH - Health Safety Research
10. RE - Environmental Research
11. RT - Transportation Safety Research
12. RS - Safeguards Research
13. RR - Effluent and Radiation Field Source Terms in Operating Reactors
14. RC - Criticality

**PRICES FOR DOCUMENTS ABSTRACTED
IN THIS REPORT**

The prices of the documents abstracted in this report depend upon the number of pages in the individual documents. The page count found in the line preceding the abstract determines the price according to the following schedule:

<u>Page Range</u>	<u>Domestic^{a,b} Price</u>
001-025	\$ 4.00
026-050	4.50
051-075	5.25
076-100	6.00
101-125	6.50
126-150	7.25
151-175	8.00
176-200	9.00
201-225	9.25
226-250	9.50
251-275	10.75
276-300	11.00
301-325	11.75
326-350	12.00
351-375	12.50
376-400	13.00
401-425	13.25
426-450	14.00
451-475	14.50
476-500	15.00
501-525	15.25
526-550	15.50
551-575	16.25
576-600	16.50
601-up	^c

^a Double the cost per copy for foreign price.

^b These prices are subject to change by NTIS.

^c Add \$2.50 for each additional 100-page increment over 600 pages.

Microfiche copies, regardless of document size, sell for \$3.00 per copy (domestic) and \$4.50 per copy (foreign), with requests from Mexico and Canada being counted as domestic. All documents may be ordered from

The National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22161

BIBLIOGRAPHY

1. IN - WATER REACTOR SAFETY RESEARCH, BASIC

120000

WEISLY HE • GILBERG PV
 PRATIC - A COMPUTER CODE FOR TIME DEPENDENT UNAVAILABILITY ANALYSIS
 U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON
 NUREG-0490 • 60 PPS, 45 FIGS, 3 REFS, OCT. 1977

PRATIC WAS DEVELOPED TO CALCULATE BOTH THE INSTANTANEOUS UNAVAILABILITY AND THE AVERAGE UNAVAILABILITY OF A SYSTEM AND TO GIVE A BREAKDOWN OF THE UNAVAILABILITY CONTRIBUTIONS FROM FAILURES, TESTING AND REPAIR. ACCIDENT SEQUENCES, SUCH AS CONSTRUCTED FROM EVENT TREES, CAN ALSO BE EVALUATED FOR THEIR INSTANTANEOUS AND AVERAGE PROBABILITY BEHAVIOR. THE PRATIC CODE REPRESENTS ONE EXTENSION OF THE PROBABILISTIC METHODOLOGY DESCRIBED IN NUREG-1400.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM • UNAVAILABILITY • FAULT TREE ANALYSIS • FAILURE • TESTING • PROBABILITY • MONITOR • COMPONENTS • RELIABILITY, COMPONENT • ACCIDENT, PROBABILITY OF • NRC-1

120000

BUCKMAN • HEDLHOLTZ GN
 LIQUID SAFETY 2. REVIEW OF CURRENT ISSUES AND BIBLIOGRAPHY OF LITERATURE, 1975-1976
 OAK RIDGE NATIONAL LAB., TENN.
 ORNL/NUREG/NSIC-125 • 5 OF ORNL/NUREG/NSIC-125 • 575 PPS, 2 TABS, 36 REFS, JULY 1977

DISCUSSES THE CURRENT STATUS OF LIQUID-METAL FAST BREEDER REACTOR (LMFR) DEVELOPMENT AND ONE OF THE PRINCIPAL SAFETY ISSUES, A HYPOTHETICAL CORE-DISRUPTIVE ACCIDENT (HCDA). BIBLIOGRAPHIC INFORMATION ON WORLDWIDE LMFRS RELATIVE TO THE DEVELOPMENT AND SAFETY OF THE BREEDER REACTOR IS PRESENTED FOR THE PERIOD 1975 THROUGH 1976. THE BIBLIOGRAPHY CONSISTS OF APPROXIMATELY 1600 ABSTRACTS COVERING EARLY RESEARCH AND DEVELOPMENT AND OPERATING EXPERIENCES LEADING UP TO THE PRESENT DESIGN PRACTICES THAT ARE NECESSARY FOR THE LICENSING OF BREEDER REACTORS. KEY-WORD, AUTHOR, AND PERMUTED-TITLE INDEXES ARE INCLUDED FOR COMPLETENESS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA. 22161
 (615-75/PULL-SIZE, 93/MF)

REACTOR, LMFR • PREVIEW • BIBLIOGRAPHY • SITING • REACTOR, BREEDER • REACTOR, FAST • R AND D PROGRAM • QUALITY ASSURANCE • OPERATING EXPERIENCE • DESIGN STUDY • ACCIDENT, CORE DISRUPTIVE • NRC-1 • NRC-7 • JACOBS

120000

HEDLHOLTZ GN
 STRUCTURAL INTEGRITY OF MATERIALS IN NUCLEAR SERVICE - A BIBLIOGRAPHY
 OAK RIDGE NATIONAL LAB., TENN.
 ORNL/NUREG/NSIC-100 • 205 PPS, JULY 1977

THIS REPORT CONTAINS 879 ABSTRACTS FROM THE NUCLEAR SAFETY INFORMATION CENTER (NSIC) COMPUTER FILE DATED 1973 THROUGH 1976 COVERING MATERIAL PROPERTIES WITH RESPECT TO THE STRUCTURAL INTEGRITY. ALL MATERIALS IMPORTANT TO THE NUCLEAR INDUSTRY (EXCEPT CONCRETE) ARE COVERED FOR MECHANICAL PROPERTIES, CHEMICAL PROPERTIES, CORROSION, FRACTURE OR FAILURE, RADIATION DAMAGE, CREEP, CRACKING, AND SWELLING. KEYWORD, AUTHOR, AND PERMUTED-TITLE INDEXES ARE INCLUDED FOR THE CONVENIENCE OF THE USER.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA. 22161
 (100-25/PULL-SIZE, 93/MF)

POWER PLANT, NUCLEAR • MATERIAL • STRUCTURAL INTEGRITY • BIBLIOGRAPHY • NRC-1 • NRC-5 • JACOBS

120300

SCOTT RL • GALLAMER RB
 SAFETY RELATED OCCURRENCES IN PRESSURIZED-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1976
 OAK RIDGE NATIONAL LAB., TENN.
 ORNL/NUREG/NSIC-130 • 415 PPS, 8 TABS, 23 REFS, AUG. 1977

THIS BIBLIOGRAPHY CONTAINS 100-WORD ABSTRACTS OF REPORTS TO THE U.S. NUCLEAR REGULATORY COMMISSION CONCERNING OPERATIONAL EVENTS THAT OCCURRED AT PRESSURIZED-WATER REACTOR NUCLEAR POWER PLANTS IN 1976. THE REPORT INCLUDES 1264 ABSTRACTS THAT DESCRIBE INCIDENTS, FAILURES, AND DESIGN OR CONSTRUCTION DEFICIENCIES EXPERIENCED AT THE FACILITIES. THEY ARE ARRANGED ALPHABETICALLY BY REACTOR NAME AND THEN CHRONOLOGICALLY FOR EACH REACTOR. THE INFORMATION LISTED IN THE TABLES INCLUDES INSTANTANEOUS FAILURES, EQUIPMENT FAILURES, SYSTEM FAILURES, CAUSES OF FAILURES, DEFICIENCIES NOTED, AND THE TIME OF OCCURRENCE DURING REFUELING, OPERATION, TESTING, OR CONSTRUCTION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA. 22161
 (1012/PULL-SIZE, U.S. ORDERS, 824 FOREIGN ORDERS; 93/MF, U.S. ORDERS, 94-50 FOREIGN ORDERS)

REACTOR, PWR • INCIDENT COMPILATION • OPERATING EXPERIENCE • BIBLIOGRAPHY • POWER PLANT, NUCLEAR • REVIEW • JACOBS

120701

SCOTT RL • GALLAMER RB

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PAGE -2

120701 *CONTINUED*
ABSTRACT BIBLIOGRAPHY OF SAFETY RELATED OCCURRENCES IN BOILING WATER NUCLEAR POWER PLANTS AS REPORTED IN 1976
ORNL/RNRC/NALAB., TENN.
ORNL/RNRC/NSIC-137 +. 400 PPS, 9 TABS, 21 REFS, SEPT. 1977

THIS BIBLIOGRAPHY CONTAINS 100-WORD ABSTRACTS OF REPORTS TO THE U.S. NUCLEAR REGULATORY COMMISSION CONCERNING OPERATIONAL EVENTS THAT OCCURRED AT BWR POWER PLANTS IN 1976. THE REPORT INCLUDES 1,83 ABSTRACTS THAT DESCRIBE INCIDENTS, FAILURES, AND DESIGN OR CONSTRUCTION DEFICIENCIES THAT WERE EXPERIENCED AT THE FACILITIES. THEY ARE ARRANGED ALPHABETICALLY BY REACTOR NAME AND THEN CHRONOLOGICALLY FOR EACH REACTOR. KEY-WORD AND PERMUTED-TITLE INDEXES ARE PROVIDED TO FACILITATE LOCATION OF THE SUBJECTS OF INTEREST, AND TABLES THAT SUMMARIZE THE INFORMATION CONTAINED IN THE BIBLIOGRAPHY ARE PROVIDED. THE INFORMATION LISTED IN THE TABLES INCLUDES INCIDENT FAILURES, EQUIPMENT FAILURES, SYSTEM FAILURES, CAUSES OF FAILURES, DEFICIENCIES NOTED, AND THE TIME OF OCCURRENCE (I.E., DURING REPELTING, OPERATION, TESTING, OR CONSTRUCTION).

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

REACTOR, BWR + INCIDENT COMPILATION + BIBLIOGRAPHY + OPERATING EXPERIENCE + DATA COLLECTION + NRC-1 + REACTOR, POWER + REACTOR POWER + JACOBS

120805
SHARP 85 + CONTROLLING
REPORTS DISTRIBUTED UNDER THE NRC LIGHT-WATER REACTOR SAFETY TECHNICAL EXCHANGE VOL. III (JANUARY-JUNE 1977)
ORNL/RNRC/NALAB., TENN.
ORNL/RNRC/NSIC-142VOL. 3 OF TID-3362 +. 20 PPS, 2 TABS, OCT. 1977

LISTS OF DOCUMENTS EXCHANGED DURING THE FIRST HALF OF 1977 UNDER AGREEMENTS BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION'S OFFICE OF NUCLEAR REGULATORY RESEARCH AND THE GOVERNMENTS OF FRANCE, FEDERAL REPUBLIC OF GERMANY, AND JAPAN ARE PRESENTED. DURING THIS PERIOD, THE NRC RECEIVED 41 REPORTS FROM FRANCE, 20 FROM F.R.G. GERMANY, AND 20 FROM JAPAN, AND IN RETURN SENT 107 U.S. REPORT TO EACH OF THESE THREE COUNTRIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

REACTOR, LWR + R AND D PROGRAM + BIBLIOGRAPHY + JAPAN + GERMANY + FRANCE + UNITED STATES + AGENCY, NRC + NRC-1 + JACOBS

120905
SUCHMAN JR
BIBLIOGRAPHY OF REPORTS ON RESEARCH SPONSORED BY THE NRC OFFICE OF NUCLEAR REGULATORY RESEARCH JANUARY-JUNE 1977
ORNL/RNRC/NALAB., TENN.
ORNL/RNRC/NSIC-143 +. 90 PPS, OCT. 1977

A BIBLIOGRAPHY OF 195 REPORTS PUBLISHED BY CONTRACTORS OF THE NRC OFFICE OF NUCLEAR REGULATORY RESEARCH DURING THE PERIOD JANUARY THROUGH JUNE 1977 IS PRESENTED ALONG WITH ABSTRACTS FROM THE NUCLEAR SAFETY INFORMATION CENTER COMPUTER FILE. THE BIBLIOGRAPHY HAS BEEN SORTED INTO THE SUBJECT CATEGORIES USED BY NRC TO ORGANIZE THE RESEARCH PROGRAM. WITHIN THE SUBJECT CATEGORIES, THE REPORTS ARE ARRANGED FIRST BY CONTRACTOR ORGANIZATION AND THEN CHRONOLOGICALLY. A BRIEF DESCRIPTION OF THE NRC RESEARCH PROGRAM PRECEDES THE BIBLIOGRAPHY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA. 22104
(\$6.50/FULL-SIZE DOMESTIC, \$11 FOREIGN ORDERS; \$3.75/FULL DOMESTIC, \$4.50 FOREIGN ORDERS)

REACTOR, LWR + REACTOR, HTGR + REACTOR, LIFER + R AND D PROGRAM + SAFETY PROGRAM + AGENCY, NRC + BIBLIOGRAPHY + NRC-1 + JACOBS

127009
KRYTER RC
INSIDE DIAGNOSTICS FOR SAFETY ASSESSMENT QUARTERLY PROGRESS REPORT FOR APRIL-JUNE 1977
ORNL/RNRC/NALAB., TENN.
ORNL/RNRC/TH-133 +. 11 PPS, SEPT. 1977

THE INFORMATION GATHERING TOURS OF VENDORS AND UTILITY USERS OF LOOSE-PARTS MONITORING SYSTEMS ARE NEARLY COMPLETE. THE CONCLUSIONS, RECOMMENDATIONS, AND PROPOSED SCOPE FOR FURTHER INVESTIGATIONS ARE PRESENTED IN THIS REPORT AND DETAILED PLANS FOR FUTURE WORK ARE BEING FORMULATED. OTHER ACTIVITIES DESCRIBED, ARE PROCEEDING ON SCHEDULE. A CONVERGENCE PROBLEM WITH A STOCHASTIC COMPUTER MODEL USED IN THE IN-CORE MONITORING PROGRAM, IS RECEIVING INTENSIVE STUDY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

INSIDE ANALYSIS + SAFETY EVALUATION + INSTRUMENT, IN CORE + INSTRUMENT, POSITION + INSTRUMENT, SURVEILLANCE + VIBRATION + CORE COMPONENTS + R AND D PROGRAM + NRC-1 + JACOBS

126006
KRYTER RC + PIETZ JR
INSIDE DIAGNOSTICS FOR SAFETY ASSESSMENT QUARTERLY PROGRESS REPORT FOR JULY-SEPTEMBER 1977
ORNL/RNRC/NALAB., TENN.
ORNL/RNRC/TH-161 +. 31 PPS, 1 TAB, 9 FIGS, 18 REFS, NOV. 1977

131056 *CONTINUED*

THE DEVELOPMENT OF ADVANCED CALCULATIONAL METHODS FOR UNDERSTANDING AND QUANTIFYING COMMONLY ENCOUNTERED NEUTRON NOISE SOURCES (E.G., COOLANT BOILING AND VIBRATIONS OF PUEL. CONTROL RODS. AND OTHER REACTOR INTERNALS) IN LWRs IS PROCEEDING ON SCHEDULE. WITH COMPLETION OF THE INITIAL NOISE-EQUIVALENT SOURCE FORMULATION AND GENERATION OF A WORKING CROSS SECTION LIBRARY BEING THE MAJOR ACCOMPLISHMENTS THIS QUARTER. A WORK PLAN FOR INVESTIGATING EXPERIMENTALLY THE PROPAGATION OF ACOUSTIC DISBURSANCES IN COMPLEX, MASSIVE, METALLIC STRUCTURES IS BEING FORMULATED AND A PORTION OF THE INSTRUMENTATION AND APPARATUS NECESSARY TO PURSUE THIS LOOSE-PARTS-ORIENTED WORK HAS OBTAINED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

NOISE ANALYSIS * OR AND D PROGRAM * NEUTRON * NOISE * REACTOR, LWR * NRC-1

130870

GENCO LA

RADIATION SIGNATURE FOLLOWING THE HYPOTHESIZED LOCA

SANDIA LABS., ALBUQUERQUE, N.M.

SAND-76-0760 * NUREG-76-0521 * 127 PPS. 17 TA'S. 36 FIGS. 15 REFS. SEPT. 1977

THIS STUDY ESTABLISHES THE RADIATION SOURCE PROFILE FOLLOWING THE HYPOTHESIZED LOSS OF COOLANT ACCIDENT (LCA) AS SUGGESTED BY THE APPLICABLE REGULATORY GUIDES. THE SOURCE IS SPECIFIED AS TIME-DEPENDENT GAMMA AND BETA ENERGY RELEASE RATES AND ENERGY SPECTRA WITH BURN AND CORE FATE VALUES PRESENTED FOR A GENERIC CONTAINMENT STRUCTURE. THE RESULTS OF THIS STUDY WILL PROVIDE A BASIS FOR A COMPARISON OF RADIATION SIMULATORS USED IN (RADIATION) QUALIFICATION TESTING OF CLASS I COMPONENTS AND AN EVALUATION OF SIMULATOR "ADEQUACY" IN DUPLICATING THE LOCA RADIATION ENVIRONMENTS AND RESULTANT COMPONENT DAMAGE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

JACOBS * QUALIFICATION * TESTING * COMPARISON * SIMULATION * COMPONENTS * RADIATION EXPOSURE * RADIATION DAMAGE * NRC-1

2. RE - WATER REACTOR SAFETY RESEARCH. SYSTEMS ENGINEERING

128704

CORRECK RA • PLAMERAN LJ • WOOTEN RB • GERRING RS
PENETRATION GENERATED IN A 1/15-SCALE MODEL OF A FOUR-LOOP PRESSURIZED WATER REACTOR

DAYVILLE COLUMBUS LAB., OHIO

ONS-NUREG-1973 •. 100 PPS. 12 TABS. 66 FIGS. 10 REFS. JUNE 1977

EXPERIMENTAL STUDIES OF ECC PENETRATION GENERATED IN A 1/15-SCALE MODEL OF A 4 LOOP PWR ARE DESCRIBED AND THE EFFECTS OF VARIOUS PUMP AND PUMP PARAMETERS, OPERATING CONDITIONS, AND GEOMETRIC PARAMETERS ON ECC PENETRATION ARE IDENTIFIED. A CORRELATION OF THE PENETRATION DATA HAS BEEN OBTAINED WHICH ACCOUNTS FOR THE EFFECTS OF THE ECC FLOW RATE AND SUBCOOLING, AND SYSTEM PRESSURE LEVEL ON ECC PENETRATION. THE CORRELATION PROVIDES A REASONABLE REPRESENTATION OF THE DATA, INCLUDING DATA OBTAINED WITH FILLED COLD LEG STEAM AND A COMMERCIAL COLD LEG STEAM SUPPLY. COMPARISONS HAVE BEEN MADE WITH THE CREANE PENETRATION DATA CORRELATION WHICH INDICATE THAT THE CORRELATIONS ARE IN SUBSTANTIAL AGREEMENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

MODEL, PHYSICAL • REACTOR, PWR • EMERGENCY COOLING • FLOW THEORY AND EXPERIMENTS • HYDRAULIC ANALYSIS • DATA PROCESSING • PUMP DISTRIBUTION • CORRELATION

128886

CARDENER RA • CORRECK RA • OHLSEN RC • GERRING RS • PLAMERAN LJ • CARR JA
STEAM-WATER MIXING AND SYSTEM HYDRODYNAMICS PROGRAM TASK • QUARTERLY PROGRESS REPORT APRIL 1, 1977-JUNE 30, 1977

DAYVILLE COLUMBUS LAB., OHIO

ONS-NUREG-1979 •. 100 PPS. SEPT. 1977

EXPERIMENTAL STUDIES WERE CONDUCTED IN A 2/15-SCALE MODEL OF A 4-LOOP PWR AT PRESSURES TO 75 PSIA TO EXTEND THE UNDERSTANDING OF STEAM-WATER INTERACTION PHENOMENA AND PROCESSES ASSOCIATED WITH A LOSS-OF-COOLANT ACCIDENT. THE INITIAL TEST SERIES CONSISTED OF QUASI-STEADY PENETRATION TESTS, ALSO, PLENUM FILLING STUDIES WITH AND WITHOUT HOT WALLS AND NO COLD LEG STEAM WERE COMPLETED. PRELIMINARY ANALYSES INDICATE THAT RESULTS OBTAINED FROM BOTH SERIES OF TESTING, QUASI-STEADY PENETRATION WITH LOWER PLENUM DRAINING AND PLENUM FILLING, PROVIDE SIMILAR RESULTS WHICH CAN BE USED TO DEVELOP CONDENSED PENETRATION CORRELATIONS AND TO COMPARE RESULTS FROM TESTS CONDUCTED IN EITHER OPERATIONAL MODE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

MODEL, PHYSICAL • REACTOR, PWR • REACTOR TEST FACILITY • ACCIDENT, LOSS OF COOLANT • STEAM • WATER • HYDRODYNAMIC ANALYSIS • NRC-2 • JACOBS

128704

CARDENER RA • CORRECK RA • OHLSEN RC • LDU JS
STEAM-WATER MIXING AND SYSTEM HYDRODYNAMICS PROGRAM, TASK 9. QUARTERLY PROGRESS REPORT JULY 1, 1977-SEPTEMBER 30, 1977

DAYVILLE COLUMBUS LAB., OHIO

ONS-NUREG-1987 •. 121 PPS. TABS. FIGS. REFS. DEC. 1977

EXPERIMENTAL STUDIES WERE CONDUCTED IN A 2/15-SCALE MODEL OF A FOUR-LOOP PRESSURIZED WATER REACTOR AT 75 PSIA TO EXTEND THE UNDERSTANDING OF STEAM-WATER INTERACTION AND PROCESSES ASSOCIATED WITH A LOSS-OF-COOLANT ACCIDENT. PLENUM FILLING TESTS WERE CONDUCTED WITH RAMPED CORE STEAM, HOT WALLS, AND NO COLD LEG STEAM. QUASI-STEADY STATE TESTS WERE CONDUCTED WITH COLD LEG STEAM, AND WITH HYDRAULIC COMMUNICATION BETWEEN THE CORE AND THE COLD LEG STEAM, WITH NO HOT WALLS. THE 1/15 STEEL AND TRANSPARENT MODELS WERE RELOCATED AND REPURCHASED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • NRC-2 • MODEL TESTING • REACTOR, PWR • ACCIDENT, LOSS OF COOLANT • FLOW THEORY AND EXPERIMENTS • EMERGENCY COOLING SYSTEM • STEAM • WATER • PLENUM FILLING • PLENUM, TWO PHASE

128430

WEISMAN J

EVALUATION OF PRESSURE DROP ACROSS AREA CHANGES DURING BLOWDOWN QUARTERLY PROGRESS REPORT JANUARY 1, 1977-MARCH 31, 1977

UNIV. OF CINCINNATI, OHIO

NUREG-0007-0 •. 3 PPS. JULY 1977

TRANSIENT PRESSURE DROPS ACROSS AREA CHANGES WERE DETERMINED IN A SERIES OF BLOWDOWN EXPERIMENTS. THESE TESTS WERE CONDUCTED WITH PWR 113 AS THE TEST FLUID IN A WELL INSTRUMENTED APPARATUS. DURING THIS PERIOD, SOME ADDITIONAL CHECK RUNS WERE MADE WITH THE 3/4" x 1 1/2" PIPE EXPANSION IN PLACE. THESE RUNS MADE THE CONCLUSION OF THE EXPERIMENTAL PROGRAM. THE DATA FOR ALL RUNS HAVE BEEN REDUCED TO ENGINEERING UNITS AND RECORDED ON A SERIES OF 6 MAGNETIC TAPES. IN ADDITION, A FINAL REPORT, CONTAINING GRAPHICAL DISPLAYS OF ALL DATA AND INSTRUCTIONS FOR USE OF THE MAGNETIC TAPES, WAS PREPARED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, LWR • PRESSURE DROP • PART OF PIPE EXPERIMENT • BLOWDOWN • PRESSURE TRANSIENT • CALCULATION • PLENUM, TWO PHASE • NRC-2

120915

WEISSMAN J • ELJAS V • KAMRAN A • TENTNER A
EXPERIMENTAL DATA ON TWO-PHASE PRESSURE DROPS ACROSS AREA CHANGES DURING FLOW TRANSIENTS
UNIV. OF CINCINNATI, OHIO
NREG-0306 VOL. 13 • 695 PGS. 2 FIGS. JULY 1977

A SERIES OF BLOWDOWN-TYPE EXPERIMENTS WERE CONDUCTED, USING PRESSURE 113 AS THE TEST FLUID, TO OBTAIN DATA ON PRESSURE DROP ACROSS ABRUPT AREA CHANGES DURING FLOW TRANSIENTS. A RAPIDLY OPENING BALL VALVE ALLOWED PRESSURE CONTAINED IN A PRESSURIZED RESERVOIR TO BE RAPIDLY DISCHARGED ACROSS AN ORIFICE WHERE FLASHING OCCURRED. THE TWO-PHASE MIXTURE PRODUCED THEN FLOWED THROUGH AN INSTRUMENTED TEST SECTION CONTAINING THE ABRUPT AREA CHANGE. DATA OBTAINED DURING THE TRANSIENT INCLUDED FLOW RATE OUT OF THE UPSTREAM RESERVOIR, VOID FRACTION AT THE INLET AND OUTLET OF THE TEST SECTION, ABSOLUTE PRESSURE, PRESSURE DROPS AT SEVERAL LOCATIONS ALONG THE PIPING AND IN THE VICINITY OF THE AREA CHANGE, AND THE TOTAL QUANTITY OF FLUID DISCHARGED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104
ACCIDENT, LOSS OF COOLANT • BLOWDOWN • PRESSURE DROP • TRANSIENT • FLOW, TWO PHASE • OUT OF PILE EXPERIMENT • NRC-2

121013

WEISSMAN J • ELJAS V • KAMRAN A • TENTNER A
EXPERIMENTAL DATA ON TWO-PHASE PRESSURE DROPS ACROSS AREA CHANGES DURING FLOW TRANSIENTS
UNIV. OF CINCINNATI, OHIO
NREG-0306 VOL. 21 • 330 PGS. JULY 1977

PRESENTS THE DATA COLLECTED FROM TESTS WITH PIPE CONTRACTIONS FROM 1 1/2 IN. TO 1 IN. AND FROM 1 1/2 IN. TO 3/4 IN. ALSO PRESENTS A DESCRIPTION OF THE INSTRUMENTATION AND THE DATA TAPES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104
N AND D PROGRAM • SAFETY PROGRAM • TRANSIENT • FLOW, TWO PHASE • PRESSURE DROP • MEASUREMENT • FLOW THEORY AND EXPERIMENTS • EXPERIMENT • NRC-2

120052

CROWLEY CJ • BLOCK JA
LOWER PLENUM VOIDING WITH SCALED PUR PLENUM GEOMETRIES QUARTERLY PROGRESS REPORT JAN. 1, 1977-MARCH 31, 1977
CREARE INC., HANOVER, N.H.
CREARE-TN-250 • 60 PGS. 15 TABS. 10 FIGS. 7 REFS. MAY 1977

RECENT RESULTS IN A PROGRAM TO DEVELOP AN ANALYTICAL AND EMPIRICAL MODEL FOR PLENUM FILLING IN A 1/15-SCALE, CYLINDRICAL, ELEVATED PRESSURE PUR-MODEL VESSEL ARE DESCRIBED. WORK DURING THE QUARTER INCLUDED STUDIES OF LOWER PLENUM VOIDING BEHAVIOR WITH SCALED MODEL WESTINGHOUSE AND BABBCOCK AND WILCOX PUR LOWER PLENUM HARDWARE. THIS REPORT PRESENTS RESULTS AND DISCUSSION OF VOIDING BEHAVIOR WITH DIRECT PLENUM INJECTION AND WITH COLD LEG INJECTION. THE REPORT ALSO COMPARES THE SCALED GEOMETRY BEHAVIOR TO EARLIER BEHAVIOR WITH SIMPLIFIED CORE INLET GEOMETRIES (SIMPLE HOLE PLATES) AND WITH BATTELLE COLUMBUS LABORATORIES' RESULTS IN SCALED GEOMETRIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104
PLENUM • REACTOR, PUR • ACCIDENT, LOSS OF COOLANT • BLOWDOWN • EMERGENCY COOLING SYSTEM • CORE REFLOODING SYSTEM • OUT OF PILE EXPERIMENT • ANALYTICAL MODEL • THERMAL HYDRAULIC ANALYSIS • NRC-2

120303

CROWLEY CJ • ROTHE PH • BLOCK JA
PRELIMINARY RAMPED TRANSIENT ANALYSIS WITH SUPERHEATED WALLS QUARTERLY PROGRESS REPORT APRIL 1, 1977-JUNE 30, 1977
CREARE INC., HANOVER, N.H.
CREARE-TN-201 • 35 PGS. FIGS. REFS. AUG. 1977

RECENT RESULTS IN A PROGRAM TO DEVELOP AN ANALYTICAL AND EMPIRICAL MODEL FOR PLENUM FILLING IN A 1/15-SCALE, CYLINDRICAL, ELEVATED PRESSURE PUR-MODEL VESSEL ARE DESCRIBED. WORK DURING THE QUARTER INCLUDED MODEL-BUILDING EXPERIMENTS MINIMIZING THE EFFECT OF WALL SUPERHEAT IN RAMPED TRANSIENTS AND SCOPING THE EFFECTS OF HIGH WALL SUPERHEAT AND ELEVATED ECC TEMPERATURE ON DELIVERY DELAYS WITH A STEADY STEAM SUPPLY. THE TOPICAL SECTION OUTLINES A PRELIMINARY ANALYSIS FOR RAMPED TRANSIENTS WITH SUPERHEATED WALLS (INCLUDING LOWER PLENUM HEAT TRANSFER) BASED ON TIME-MARCHING APPLICATION OF STEADY-STATE CORRELATIONS AND EQUATIONS. THE PRELIMINARY ANALYSIS IS COMPARED WITH BASELINE RAMPED TRANSIENT AND STEADY STEAM SUPPLY DATA.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104
JACOBS • MODEL, PHYSICAL • REACTOR, PUR • REACTOR TEST FACILITY • PRESSURE VESSELS • PLENUM • SUPERHEAT • REACTIVITY EFFECT • REACTOR TRANSIENT • ANALYTICAL MODEL • ACCIDENT, LOSS OF COOLANT • NRC-2

122211

SCHWIEBERG D
OWN BLOWDOWN/EMERGENCY CORE COOLING FOURTH QUARTERLY PROGRESS REPORT - OCTOBER 1-DECEMBER 31, 1976

132211 *CONTINUOUS*
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
GEAP-21300-6 P. 11 PPS. 5 REFS. JAN. 1977

THE PREPARATION OF THE TEST PLAN FOR THE 600 BOWT TEST PHASE, THE ASSEMBLY OF THE 600 BOWT ROD BUNDLE TEST SECTION, AND THE DEVELOPMENT OF A SELF-STANDING TRANSIENT THERMAL-HYDRAULIC MODEL FOR PREDICTION OF LOCAL THERMOMECHANICAL PARAMETERS IN ROD BUNDLES DURING LOCA CONDITIONS WERE COMPLETED DURING THE REPORTING PERIOD. DOCUMENTATION OF THE THERMAL-HYDRAULIC MODEL IS IN PROGRESS. THE VTA HARDWARE IS CURRENTLY BEING PREPARED FOR 600 BOWT TESTING. THE PRELIMINARY FACILITY DESCRIPTION REPORT FOR SD/ECCIA TESTING IS IN PROGRESS. VTA COMPONENT COPL CHARACTERISTIC TESTS IN THE COPL TEST APPARATUS WERE STARTED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, BWR • BLOWDOWN • ACCIDENT, LOSS OF COOLANT • EMERGENCY COOLING SYSTEM • CORE REFLOODING SYSTEM • OUT OF PILE EXPERIMENT • GENERAL ELECTRIC • NRC-2 • JACOB

132222
FURNISHES UC
NATHAN - A METHOD TO EVALUATE TRANSIENT THERMAL HYDRAULIC CONDITIONS IN ROD BUNDLES
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
GEAP-22017 P. 110 PPS. 11 TABS. 10 FIGS. MARCH 1977

DESCRIBES THE TECHNICAL BASIS AND NUMERICAL CALCULATIONS PERFORMED BY THE CODE NATHAN. THE CODE ANALYZES ONE-DIMENSIONAL SINGLE CHANNEL HYDRAULIC AND HEAT TRANSFER TRANSIENTS IN ROD BUNDLES AND CALCULATED LOCAL HYDRAULIC CONDITIONS AND THEN HEATER OR FUEL ROD HEAT TRANSFER INFORMATION (I.E., TEMPERATURE AND HEAT TRANSFER COEFFICIENTS). THE BRIEF PLAN FORMULATION EMPLOYED IS PARTICULARLY USEFUL IN HANDLING PROBLEMS TYPICAL OF THE LOSS-OF-COOLANT ACCIDENT ANALYSIS. THE CONSTITUTION EQUATIONS ARE WRITTEN IN CANONICAL FORM AND ARE SOLVED USING THE METHOD OF CHARACTERISTICS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM • THERMAL TRANSIENT • FUEL ELEMENTS • ACCIDENT, LOSS OF COOLANT • FUEL ROD • THERMAL HYDRAULIC ANALYSIS • CORRELATION • HEAT TRANSFER COEFFICIENT • FLOW, TWO PHASE • NRC-2

132409
WOOD JC
BWR BLOWDOWN/EMERGENCY CORE COOLING FIFTH QUARTERLY PROGRESS REPORT JANUARY 1-MARCH 31, 1977
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
GEAP-21300-5 P. 31 PPS. 5 TABS. 9 FIGS. 3 REFS. APRIL 1977

INSTALLATION OF THE 60-400 BUNDLE, COMPLETION OF THE FIRST THREE 600 BOWT MATRIX TESTS, AND REPORTING OF ADDITIONAL COPL TEST RESULTS WERE THE HIGHLIGHTS OF WORK ACCOMPLISHED DURING THE PERIOD.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, BWR • BLOWDOWN • ACCIDENT, LOSS OF COOLANT • EMERGENCY COOLING SYSTEM • CORE REFLOODING SYSTEM • OUT OF PILE EXPERIMENT • NRC-2

132601
REVISION AP
BWR BLOWDOWN/EMERGENCY CORE COOLING PROGRAM-60-400 BUNDLE CORE SPRAY INTERACTION (SD/ECCIA) TEST PLAN
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
GEAP-21300-4 P. 47 PPS. REFS. JUNE 1977

A TEST PLAN FOR THE SD/ECCIA PHASE OF THE SD/ECC TEST PROGRAM IS PRESENTED. THE TEST PROGRAM WILL INVESTIGATE SYSTEM-CORE SPRAY INTERACTION UNDER LOCA CONDITIONS IN A SCALED 210 BWR/6 TEST APPARATUS. TEST PARAMETERS AND RANGES ARE SELECTED; INDIVIDUAL TESTS ARE DEFINED AND ARRANGED IN A TEST MATRIX; AND MEASUREMENT AND DATA UTILIZATION PLANS ARE PROPOSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOB • REACTOR, BWR • CORE SPRAY • EMERGENCY COOLING SYSTEM • GENERAL ELECTRIC • BLOWDOWN • NRC-2 • FUEL COOLANT INTERACTION

132700
NATHAN BROWMAN A
BWR BLOWDOWN/EMERGENCY CORE COOLING SIXTH QUARTERLY PROGRESS REPORT APRIL 1-JUNE 30, 1977
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
GEAP-21300-6 P. 16 PPS. 4 FIGS. 5 REFS. JULY 1977

SD/ECC WORK COMPLETED IN THE SECOND QUARTER OF 1977 IS SUMMARIZED. COMPLETION OF TWO 600 BOWT MATRIX TESTS, VERIFICATION AND COMPARISON OF DATA FROM THE FIRST THREE 600 BOWT MATRIX TESTS WITH CORRESPONDING 707 BOWT TESTS, AND THE DOCUMENTATION OF THE TRANSIENT THERMAL HYDRAULIC CODE (NATHAN) AND THE SD/ECC 1A PRELIMINARY TEST PLAN WERE THE HIGHLIGHTS OF WORK ACCOMPLISHED DURING THE PERIOD.

127006 (CONTINUED)
 AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND B PROGRAM • SAFETY PROGRAM • NRC-2 • REACTOR, BOR • BLOWDOWN • HEAT TRANSFER • COMPUTER PROGRAM •
 EMERGENCY COOLING SYSTEM • OUT OF PILE EXPERIMENT

127000
 PELHAM EN • SACKETT RE
 EXPERIMENT DATA REPORT FOR SENISCALE MOD-1 TESTS S-05-6 AND S-05-7 (ALTERNATE ECC INJECTION TESTS)
 IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
 TREE-NUREG-1096 • 325 PPS, 1 TABS, 596 FIGS, 1 REF, JUNE 1977

RECORDED TEST DATA ARE PRESENTED FOR TESTS S-05-6 AND S-05-7 OF THE SENISCALE MOD-1 ALTERNATE ECC INJECTION TEST SERIES, CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING A HYPOTHESIZED LOSS-OF-COOLANT ACCIDENT IN A PRESSURIZED WATER REACTOR (PWR) SYSTEM. THE SPECIFIC OBJECTIVE FOR THESE TESTS WAS TO INVESTIGATE THE EFFECTIVENESS OF LOW PRESSURE INJECTION SYSTEM (LPIIS) COOLANT INJECTION INTO THE UPPER PLENUM WHEN COMBINED WITH COLD LEG INTACT LOOP INJECTION. THE SENISCALE MOD-1 SYSTEM WAS MODIFIED TO REPRESENT MORE CLOSELY A TWO-LOOP PWR. HOWEVER, TESTS S-05-6 AND S-05-7 WERE NOT EXPECTED TO BE A DEMONSTRATION OF A TWO-LOOP PWR SYSTEM BECAUSE OF SCALING DISTORTIONS IN THE SENISCALE MOD-1 SYSTEM RELATIVE TO A TWO-LOOP PWR.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ACCIDENT, LOSS OF COOLANT • REACTOR, PWR • BLOWDOWN • EMERGENCY COOLING SYSTEM • CORE REFLOODING SYSTEM • OUT
 OF PILE EXPERIMENT • THERMAL EXPERIMENT • HYDRAULIC EXPERIMENT • FLOW, TWO PHASE • NRC-2

127056
 WELKINS SC
 EMBEDDED CLADDING SURFACE THERMOCOUPLES ON ZIRCALOY-SHEATHED HEATER RODS
 IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
 TREE-NUREG-1072 • 27 PPS, 2 TABS, 21 FIGS, 4 REFS, JUNE 1977

TITANIUM-SHEATHED TYPE K THERMOCOUPLES EMBEDDED IN THE CLADDING WALL OF ZIRCALOY-SHEATHED HEATER RODS ARE DESCRIBED WITHIN THE TEXT. THESE THERMOCOUPLES CONSTITUTE PART OF A PROGRAM INTENDED TO CHARACTERIZE THE UNCERTAINTY OF MEASUREMENTS MADE BY SURFACE-MOUNTED CLADDING THERMOCOUPLES ON NUCLEAR FUEL RODS. FABRICATION AND INSTALLATION DETAILS, AND LABORATORY TESTING OF SAMPLE THERMOCOUPLE INSTALLATIONS ARE INCLUDED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FUEL ROD • ZIRCALOY • SIMULATION • HEATERS • THERMOCOUPLE • NRC-2

129002
 ESPARZA V • SACKETT RE
 EXPERIMENT DATA REPORT FOR SENISCALE MOD-1 TEST S-06-5 (LOFT COUNTERPART TEST)
 IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
 TREE-NUREG-1125 • 213 PPS, 9 TABS, 350 FIGS, 1 REF, JUNE 1977

RECORDED TEST DATA ARE PRESENTED FOR TEST S-06-5 OF THE SENISCALE MOD-1 LOFT COUNTERPART TEST SERIES, CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING A LOCA IN A PWR. TEST S-06-5 WAS CONDUCTED FROM INITIAL CONDITIONS OF 2272 PSIA AND 536 F TO INVESTIGATE THE RES JUMP OF THE SENISCALE MOD-1 SYSTEM TO A DEPRESSURIZATION AND REFLOOD TRANSIENT FOLLOWING A SIMULATED DOUBLE-ENDS OFFSET SHEAR OF THE BROKEN LOOP COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEGS OF THE INTACT AND BROKEN LOOPS TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR. THE PURPOSE WAS TO ASSESS THE INFLUENCE OF THE BREAK NOZZLES GEOMETRY ON CORE THERMAL AND SYSTEM RESPONSE AND ON THE SUBCOOLED AND LOW QUALITY MASS FLOW RATES AT THE BREAK LOCATIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, PWR • ACCIDENT, LOSS OF COOLANT • BLOWDOWN • EMERGENCY COOLING SYSTEM • CORE REFLOODING • OUT OF PILE
 EXPERIMENT • FLOW, TWO PHASE • NRC-2

129103
 CARO ON
 QUARTERLY TECHNICAL PROGRESS REPORT ON WATER REACTOR SAFETY PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY
 COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH (JAN-MAR 1977)
 IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
 TREE-NUREG-1120 • 135 PPS, 17 TABS, 75 FIGS, REFS, JUNE 1977

LIGHT WATER REACTOR SAFETY RESEARCH PERFORMED JANUARY THROUGH MARCH 1977 IS DISCUSSED, INCLUDING THE FOLLOWING: A TEST SERIES TO EVALUATE THE EFFECTIVENESS OF DIFFERENT EMERGENCY CORE COOLANT INJECTION ARRANGEMENTS WAS COMPLETED. A LOFT TEST SERIES WAS INITIATED TO INVESTIGATE THE EFFECT OF BREAK NOZZLE GEOMETRY ON BREAK FLOW AND SYSTEM RESPONSE DURING A 2000 DOUBLE-ENDED COLD LEG BREAK TEST. TWO ANALYSES OF RESULTS OF LOFT NONNUCLEAR EXPERIMENTS WERE MADE. A REVIEW OF PLECHT AND SENISCALE FORCED FLOW AND GRAVITY FLOW REFLOOD HPAT TRANSFER EXPERIMENTS WAS PERFORMED. A MORE GENERAL VERSION OF THE R-LAPS PILOT CODE (VERSION 2) WAS PROGRAMMED. EXPERIMENT RESULTS FROM TESTS CONDUCTED AT THE HARVIREN CONTAINMENT TEST FACILITY IN SWEDEN AND THE BATTLE TEST FACILITY IN FRANKFURT, GERMANY, WERE USED FOR VERIFICATION OF COMPUTER

PAGE -8

120683 *CONTINUED*
CONTAINMENT CODES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, LWR + SAFETY ANALYSIS + OUT OF PILE EXPERIMENT + COMPUTER PROGRAM, DIGITAL + COMPARISON, THEORY AND EXPERIENCE + EMERGENCY COOLING SYSTEM + BLOWDOWN + ACCIDENT, LOSS OF COOLANT + NRC-2 + NRC-3 + NRC-4

120693
RUSSELL RL
LEFT FUEL MODULES DESIGN, CHARACTERIZATION, AND FABRICATION PROGRAM
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NURE-1131 P., 100 PPS, 10 TABS, 30 FIGS, JUNE 1977

THE LOSS-OF-FLUID TEST (LOFT) FUEL MODULES HAVE EVOLVED FROM A COMPREHENSIVE FIVE-YEAR DESIGN, CHARACTERIZATION, AND FABRICATION PROGRAM WHICH HAS RESULTED IN THE ACCOMPLISHMENT OF MANY TECHNICAL ACTIVITIES OF INTEREST IN PRESSURIZED WATER REACTOR FUEL DESIGN DEVELOPMENT AND SAFETY RESEARCH. THIS REPORT SUMMARIZES THE HIGHLIGHTS WHICH INCLUDE: (1) DETERMINATION OF FUNDAMENTAL HIGH-TEMPERATURE REACTOR MATERIAL PROPERTIES, (2) DESIGN INVENTION RELATED TO IN-CORE DISTRIBUTION ATTACHMENT, (3) IMPLEMENTATION OF ADVANCED AND/OR UNIQUE FUEL BUNDLE CHARACTERIZATION TECHNIQUES, (4) IMPLEMENTATION OF IMPROVED FUEL BUNDLE FABRICATION TECHNIQUES, AND (5) PLANNING AND EXECUTION OF A MULTIMILLION DOLLAR DESIGN, CHARACTERIZATION, AND FABRICATION PROGRAM FOR PRESSURIZED WATER REACTOR FUEL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, PWR + ACCIDENT, LOSS OF COOLANT + BLOWDOWN + LOFT (S-MR) + EMERGENCY COOLING SYSTEM + NRC-2

120697
CALLINE BL + PATTON RL + SACKETT KE
EXPERIMENT DATA REPORT FOR SEMISCALE MOD-1 TEST S-06-1 (LOFT COUNTERPART TEST)
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NURE-1121 P., 202 PPS, JULY 1977

RECORDED TEST DATA ARE PRESENTED FOR TEST S-06-1 OF THE SEMISCALE MOD-1 LOFT COUNTERPART TEST SERIES. THESE TESTS ARE AMONG SEVERAL SEMISCALE MOD-1 EXPERIMENTS CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING AN HYPOTHESIZED LOCA IN A PWR. TEST S-06-1 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15 500 KPA AND 364 X TO INVESTIGATE THE RESPONSE OF THE SEMISCALE MOD-1 SYSTEM TO A DEPRESSURIZATION AND REFLOOD TRANSIENT FOLLOWING A SIMULATED DOUBLE-ENDED OR SET SHEAR OF THE BROKEN LOOP COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INTACT LOOP TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS + REACTOR TEST FACILITY + ACCIDENT, LOSS OF COOLANT + REACTOR, PWR + SIMULATION + THERMAL HYDRAULIC ANALYSIS + EMERGENCY COOLING + SAFETY INJECTION + IDAHO NAT ENG LAB + DATA COLLECTION + DATA PROCESSING + CORE REPLEADING + PRESSURE TRANSIENT + NRC-2

120610
CARTWELL CE
ANALYSIS OF STANDARD PROBLEM SIX (SEMISCALE TEST S-02-6) DATA
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NURE-1096 P., 56 PPS, 1 TAB, 33 FIGS, 11 REFS, AUG. 1977

TEST S-02-6 OF THE SEMISCALE MOD-1 BLOWDOWN HEAT TRANSFER TEST SERIES WAS CONDUCTED TO SUPPLY DATA FOR NRC STANDARD PROBLEM SIX. TO DETERMINE THE CREDIBILITY OF THE DATA AND THUS ESTABLISH THE VALIDITY OF STANDARD PROBLEM SIX, AN ANALYSIS OF THE RESULTS OF TEST S-02-6 WAS PERFORMED AND IS PRESENTED. THIS ANALYSIS CONSISTED OF INVESTIGATIONS OF SYSTEM HYDRAULIC AND CORE THERMAL DATA. THE CREDIBILITY OF THE SYSTEM HYDRAULIC DATA WAS INVESTIGATED THROUGH COMPARISONS OF THE DATA WITH DATA AND CALCULATIONS FROM RELATED SOURCES (TEST S-02-4) AND THROUGH ASSESSMENT OF PHYSICAL EVENTS. THE CREDIBILITY OF THE CORE THERMAL DATA WAS BASED ON A THOROUGH ANALYSIS OF PHYSICAL EVENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS + BLOWDOWN + HEAT TRANSFER ANALYSIS + CORE + THERMAL HYDRAULIC ANALYSIS + ACCIDENT, LOSS OF COOLANT + REACTOR, PWR + ANALYTICAL MODEL + COMPARISON + AGENCY, NRC + IDAHO NAT ENG LAB + NRC-2

120906
HARRISON RB
EVALUATION OF THE EFFECTS OF BREAK NOZZLE CONFIGURATION IN THE SEMISCALE MOD-1 SYSTEM
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NURE-1110 P., 30 PPS, 4 TABS, 20 FIGS, AUG. 1977

THE SEMISCALE MOD-1 PROGRAM HAS UTILIZED TWO DIFFERENT BREAK NOZZLE CONFIGURATIONS IN THE TEST SYSTEM. AN EVALUATION HAS BEEN MADE TO DETERMINE THE EFFECT THESE BREAK NOZZLE CONFIGURATIONS HAVE ON SYSTEM THERMAL-HYDRAULIC RESPONSE DURING A 200S DOUBLE-ENDED COLD LEG BREAK LOSS-OF-COOLANT ACCIDENT SIMULATION. THE FIRST NOZZLE WAS A CONVERGENT-DIVERGENT NOZZLE (HENRY NOZZLE)

129300 *CONTINUED*

AND THE SECOND, AN ELONGATED CONSTANT AREA THROAT NOZZLE. ANALYSIS IS CONFINED PRIMARILY TO SYSTEM RESPONSE PHENOMENA OBSERVED TO BE AFFECTED BY THE NOZZLE CONFIGURATION AND CONCENTRATES ON THE FLUID RESPONSE AT THE BREAK AND THE RESULTING CORE BEHAVIOR DURING SUBCOOLED AND SATURATED BLOWDOWN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

JACOBS * ACCIDENT, LOSS OF COOLANT * SIMULATION * REACTOR TEST FACILITY * THERMAL HYDRAULIC ANALYSIS * BLOWDOWN * CORE * RESPONSE SPECTRUM * IDAHO NAT ENG LAB * NRC-2

129300

PATTON RL * COLLINS BL * SACCHETTI KE
EXPERIMENT DATA REPORT FOR SEMISCALE MOD-1 TEST S-06-2 (LOFT COUNTERPART TEST)
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NUREG-1122 * 202 PGS, FIGS, AUG. 1977

RECORDED TEST DATA ARE PRESENTED FOR TEST S-06-2 OF THE SEMISCALE MOD-1 LOFT COUNTERPART TEST SERIES. THESE TESTS ARE AMONG SEVERAL SEMISCALE MOD-1 EXPERIMENTS CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING AN HYPOTHESIZED LOCA IN A PWR. TEST S-06-2 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15.513 MPa AND 563 °C TO INVESTIGATE THE RESPONSE OF THE SEMISCALE MOD-1 SYSTEM TO A DEPRESSURIZATION AND REPLED TRANSIENT FOLLOWING A SIMULATED DC BLE-ENDED OFFSET SHEAR OF THE BROKEN LOOP COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INTACT LOOP TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

JACOBS * REACTOR TEST FACILITY * ACCIDENT, LOSS OF COOLANT * REACTOR, PWR * SIMULATION * THERMAL HYDRAULIC ANALYSIS * EMERGENCY COOLING * SAFETY INJECTION * DATA COLLECTION * DATA PROCESSING * CORE REPLEDING * PRESSURE TRANSIENT * IDAHO NAT ENG LAB * NRC-2

130500

QUARTERLY TECHNICAL PROGRESS REPORT ON WATER REACTOR SAFETY PROGRAM SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH APRIL-JUNE 1977
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NUREG-1147 * 92 PGS, AUG. 1977

RESULTS FROM THE PREVIOUSLY CONDUCTED SEMISCALE MOD-1 ECC INJECTION TEST SERIES WERE ANALYZED. TESTING IN THE LOFT COUNTERPART TEST SERIES WAS ESSENTIALLY COMPLETED, AND THE STEAM GENERATOR TUBE RUPTURE TEST SERIES WAS BEGUN. TWO TESTS IN THE ALTERNATE ECC INJECTION TEST SERIES WERE CONDUCTED WHICH INCLUDED INJECTION OF EMERGENCY CORE COOLANT INTO THE UPPER PLENUM THROUGH USE OF THE LOW PRESSURE INJECTION SYSTEM. THE LOSS-OF-FLUID TEST PROGRAM SUCCESSFULLY COMPLETED NONNUCLEAR LOSS-OF-COOLANT EXPERIMENT L1-0. A NUCLEAR TEST, CC 2-3, IN THE POWER BURST FACILITY REACTOR WAS PERFORMED TO EVALUATE THE POWER OSCILLATION METHOD OF DETERMINING GAP CONDUCTANCE TO DETERMINE THE EFFECTS OF INITIAL GAP SIZE, FILL GAS COMPOSITION, AND FUEL DENSITY ON THE THERMAL PERFORMANCE OF A LIGHT WATER REACTOR FUEL ROD.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

OR AND D PROGRAM * SAFETY PROGRAM * REACTOR, LWR * ACCIDENT, LOSS OF COOLANT * EMERGENCY COOLING SYSTEM * HEAT FLUX, CRITICAL * DNB * THERMOCOUPLE * HEAT CONDUCTANCE, FUEL TO CLAD * COMPUTER PROGRAM * CREEP BEHAVIOR * THERMAL HYDRAULIC ANALYSIS * COMPARISON, THEORY AND EXPERIENCE * NRC-2 * NRC-3 * NRC-4 * JACOBS

131100

SILVERMAN S
PRINCIPLES OF OPERATION AND DATA REDUCTION TECHNIQUES FOR THE LOFT DRAG DISC TURBINE TRANSDUCER
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NUREG-1109 * 163 PGS, 21 TABS, 62 FIGS, REFS, SEPT. 1977

FOR ALL-WATER FLOW IN PIPES UP TO AND INCLUDING 11-IN. ID, THE PRESENCE OF THE DRAG DISC TURBINE TRANSDUCER DISRUPTS THE VELOCITY PROFILE TO THE EXTENT THAT THE TURBINE METER OUTPUT IS PROPORTIONAL TO AVERAGE VELOCITY. FOR THE TEST DATA ANALYZED THE THREE TWO-FLUID MODELS REDUCE THE SCATTER AS COMPARED TO A SIMPLE HOMOGENEOUS MODEL. FOR ANY GIVEN MODEL, PREDICTED STEAM AND LIQUID VELOCITIES FOR FORWARD FLOW AGREE CLOSELY WITH THOSE FOR REVERSE FLOW. THE TURBINE 10 TO 90% RISE TIME IS ESTIMATED TO BE ABOUT 0.29 SEC IN WATER AND 11.9 SEC IN STEAM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

R AND D PROGRAM * SAFETY PROGRAM * REACTOR, LWR * LOFT (S-06) * TURBINE * INSTRUMENT, FLOW * MEASUREMENT * TESTING * FLOW, TWO PHASE * STEAM * WATER * NRC-2 * JACOBS

131107

MARVEGO EA * LOOMIS GG * PETERSON AC
INVESTIGATION OF ALTERNATE ECC INJECTION CONCEPTS IN THE SEMISCALE MOD-1 SYSTEM
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NUREG-1117 * 179 PGS, 5 TABS, 151 FIGS, REFS, SEPT. 1977

THE POTENTIAL BENEFITS AND RELATIVE MERITS OF FOUR ALTERNATE EMERGENCY CORE COOLANT (ECC)

131107 *CONTINUED*

INJECTION CONCEPTS ARE INVESTIGATED IN THE SEMISCALE MOD-1 SYSTEM. THE MAJOR (CONROLLING THERMAL-HYDRAULIC PHENOMENA ASSOCIATED WITH THE EFFECTIVENESS OF EACH ALTERNATE ECC INJECTION CONCEPT ARE ALSO EXAMINED. THE PRINCIPAL MEANS USED TO EVALUATE THE VARIOUS CONCEPTS IS A COMPARATIVE ANALYSIS OF EXPERIMENTAL DATA FOR A GIVEN CONCEPT (FROM TEST SERIES 5) WITH DATA FROM A BASELINE COLD LEG ECC INJECTION TEST (TEST 5-06-6). RUN AS PART OF A PICHON TEST SERIES, AND WITH DATA FROM OTHER TESTS IN THE ALTERNATE ECC INJECTION TEST SERIES. AS A RESULT OF THIS INVESTIGATION CONCLUSIONS PERTAINING TO THE EFFECTIVENESS OF THE DIFFERENT CONCEPTS IN THE SEMISCALE MOD-1 SYSTEM ARE REACHED, AND THE RELATIVE MERITS OF THE CONCEPTS ARE IDENTIFIED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + REACTOR, LWR + ACCIDENT, LOSS OF COOLANT + BLOWDOWN + EMERGENCY COOLING SYSTEM + THERMAL HYDRAULIC ANALYSIS + INJURY OF PIPE EXPERIMENT + IDAHO FALLS + SAFETY INJECTION + NRC-2 + JACOBS

130639

ESPARZA V + SACKETT RE
EXPERIMENT DATA REPORT FOR SEMISCALE MOD-1 TEST 5-06-6 (LEFT COUNTERPART TEST)
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NUREG-1126 +, 284 PPS, 12 TABS, 395 FIGS, REFS, SEPT. 1977

RECORDED TEST DATA ARE PRESENTED FOR TEST 5-06-6 OF THE SEMISCALE MOD-1 LEFT COUNTERPART TEST SERIES. THESE TESTS ARE AMONG SEVERAL SEMISCALE MOD-1 EXPERIMENTS CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING A HYPOTHESIZED LOSS-OF-COOLANT ACCIDENT IN A PWR. TEST 5-06-6 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15 792 KPA AND 563 °C TO INVESTIGATE THE RESPONSE OF THE SEMISCALE MOD-1 SYSTEM TO A DEPRESSURIZATION AND REFLUDD TRANSIENT FOLLOWING A SIMULATED DOUBLE-ENDED OFFSET SHEAR OF THE BROKEN LOOP COLD LEG PIPING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, PWR + ACCIDENT, LOSS OF COOLANT + REACTOR TEST FACILITY + IDAHO NAT ENG LAB + DATA COLLECTION + CORE REFLUDDING + PRESSURE TRANSIENT + THERMAL HYDRAULIC ANALYSIS + JACOBS + NRC-2 + EXPERIMENT

130632

DAGBJARTSSON SJ + MURDOCK BA + MACDONALD PE + OWEN DE
AXIAL GAS FLOW IN IRRADIATED PWR FUEL RODS
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NUREG-1126 +, 140 PPS, 1 FIGS, REFS, SEPT. 1977

TRANSIENT AND STEADY STATE AXIAL GAS FLOW EXPERIMENTS WERE PERFORMED ON 6, IRRADIATED, COMMERCIAL PWR FUEL RODS AT AMBIENT TEMPERATURE AND 323 K. LAMINAR FLOW EQUATIONS, AS USED IN THE PRAP-T2 AND SSYST FUEL BEHAVIOR CODES, WERE USED WITH THE GAS FLOW RESULTS TO CALCULATE EFFECTIVE FUEL ROD RADIAL GAPS. THE RESULTS OF THESE ANALYSES WERE COMPARED WITH MEASURED GAP SIZES OBTAINED FROM METALLOGRAPHIC EXAMINATION OF ONE FUEL ROD. USING MEASURED GAP SIZES AS INPUT, THE SSYST CODE WAS USED TO CALCULATE PRESSURE DROPS AND MASS FLUXES AND THE RESULTS WERE COMPARED WITH THE EXPERIMENTAL GAS FLOW DATA.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

STEADY STATE + EXPERIMENT + GAS + FLOW, AXIAL + FUEL ROD + IRRADIATION TESTING + REACTOR, PWR + COMPARISON, THEORY AND EXPERIENCE + IDAHO NAT ENG LAB + JACOBS + NRC-2

131944

WHITE JR + HOLMSTROM MLO
POSTTEST RELAP5 ANALYSIS OF LOFT EXPERIMENT L1-3A
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NUREG-1137 +, 190 PPS, 68 FIGS, 10 REFS, OCT. 1977

PRESENTS SELECTED RESULTS OF POSTTEST RELAP5 MODELING OF LOFT LOSS-OF-COOLANT EXPERIMENT L1-3A, A DOUBLE-ENDED ISOTHERMAL COLD LEG BREAK WITH LOWER PLENUM EMERGENCY CORE COOLANT INJECTION. COMPARISONS ARE PRESENTED BETWEEN THE PRETEST PREDICTION, THE POSTTEST ANALYSIS, AND THE EXPERIMENTAL DATA. IT IS CONCLUDED THAT PRESSURIZER MODELING IS IMPORTANT FOR ACCURATELY PREDICTING SYSTEM BEHAVIOR DURING THE INITIAL PORTION OF SATURATED BLOWDOWN. USING MEASURED INITIAL CONDITIONS RATHER THAN NOMINAL SPECIFIED INITIAL CONDITIONS DID NOT INFLUENCE THE SYSTEM MODEL RESULTS SIGNIFICANTLY. USING FINER MODELIZATION IN THE REACTOR VESSEL IMPROVED THE PREDICTION OF THE SYSTEM PRESSURE HISTORY BY MINIMIZING STEAM CONDENSATION EFFECTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + ACCIDENT, LOSS OF COOLANT + SAFETY INJECTION + CORE SPRAY + MODEL + COMPARISON, THEORY AND EXPERIENCE + BLOWDOWN + EMERGENCY COOLING CONSIDERATIONS + COMPUTER PROGRAM + NRC-2 + JACOBS

131110

COLLINS BL + COPPIN CE + SACKETT RE
EXPERIMENT DATA REPORT FOR SEMISCALE MOD-1 TEST 5-20-1 (STEAM GENERATOR TUBE RUPTURE TEST)
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NUREG-1140 +, 289 PPS, 12 TABS, 211 FIGS, REFS, OCT. 1977

131110 *CONTINUED*

RECORDED TEST DATA ARE PRESENTED FOR TEST S-28-1 OF THE SEMISCALE MOD-1 STEAM GENERATOR TUBE RUPTURE TEST SERIES. THESE TESTS ARE AMONG SEVERAL SEMISCALE MOD-1 EXPERIMENTS CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING A HYPOTHESIZED LOSS-OF-COOLANT ACCIDENT IN A PWR. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INTACT AND BROKEN LOOPS TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR. SIXTY STEAM GENERATOR TUBE RUPTURES WERE SIMULATED BY A CONTROLLED INJECTION FROM A HEATED ACCUMULATOR INTO THE INTACT LOOP HOT LEG.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM - SAFETY PROGRAM - REACTOR, LWR - SIMULATION - OUT OF PILE EXPERIMENT - THERMAL HYDRAULIC ANALYSIS - ACCIDENT, LOSS OF COOLANT - FAILURE, TUBING - BLOWDOWN - STEAM GENERATOR - SAFETY INJECTION - MOCKUP - NRC-2 - JACOBS

131109

PATTON WL - SACKETT KE
EXPERIMENT DATA REPORT FOR SEMISCALE MOD-1 TEST S-28-2 (STEAM GENERATOR TUBE RUPTURE TEST)
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-MUREG-1149 - 262 PPS, 12 TABS, 411 FIGS, REFS, OCT. 1977

RECORDED TEST DATA ARE PRESENTED FOR TEST S-28-2 OF THE SEMISCALE MOD-1 STEAM GENERATOR TUBE RUPTURE TEST SERIES. THESE TESTS ARE AMONG SEVERAL SEMISCALE MOD-1 EXPERIMENTS CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING A HYPOTHESIZED LOSS-OF-COOLANT ACCIDENT IN A PWR. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INTACT AND BROKEN LOOPS TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR. FOR TEST S-28-2, ACCUMULATOR INJECTION INTO THE INTACT LOOP HOT LEG WAS PROVIDED TO SIMULATE THE RUPTURE OF SIX STEAM GENERATOR TUBES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM - SAFETY PROGRAM - REACTOR, LWR - SIMULATION - OUT OF PILE EXPERIMENT - THERMAL HYDRAULIC ANALYSIS - ACCIDENT, LOSS OF COOLANT - FAILURE, TUBING - BLOWDOWN - STEAM GENERATOR - SAFETY INJECTION - MOCKUP - NRC-2 - JACOBS

131106

GILLINS RL - SACKETT KE
EXPERIMENT DATA REPORT FOR SEMISCALE MOD-1 TEST S-28-3 (STEAM GENERATOR TUBE RUPTURE TEST)
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-MUREG-1150 - 251 PPS, 12 TABS, 391 FIGS, REFS, OCT. 1977

RECORDED TEST DATA ARE PRESENTED FOR TEST S-28-3 OF THE SEMISCALE MOD-1 STEAM GENERATOR TUBE RUPTURE TEST SERIES. THESE TESTS ARE AMONG SEVERAL SEMISCALE MOD-1 EXPERIMENTS CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING A HYPOTHESIZED LOSS-OF-COOLANT ACCIDENT IN A PWR. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INTACT AND BROKEN LOOPS TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR. TWELVE STEAM GENERATOR TUBE RUPTURES WERE SIMULATED BY A CONTROLLED INJECTION FROM A HEATED ACCUMULATOR INTO THE INTACT LOOP HOT LEG.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM - SAFETY PROGRAM - REACTOR, LWR - SIMULATION - OUT OF PILE EXPERIMENT - THERMAL HYDRAULIC ANALYSIS - ACCIDENT, LOSS OF COOLANT - FAILURE, TUBING - BLOWDOWN - STEAM GENERATOR - SAFETY INJECTION - MOCKUP - NRC-2 - JACOBS

132705

ESPARZA V - SACKETT KE
EXPERIMENT DATA REPORT FOR SEMISCALE MOD-1 TEST S-28-4 (STEAM GENERATOR TUBE RUPTURE TEST)
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-MUREG-1151 - 256 PPS, 12 TABS, 403 FIGS, NOV. 1977

TEST S-28-4 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15 646 KPA AND 557 K TO INVESTIGATE THE RESPONSE OF THE SEMISCALE MOD-1 SYSTEM TO DEPRESSURIZATION AND REFLOOD TRANSIENT FOLLOWING A SIMULATED DOUBLE-END OF OFFSET SHEAR OF THE BROKEN LOOP COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INTACT AND BROKEN LOOPS TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR. THIRTY STEAM GENERATOR TUBE RUPTURES WERE SIMULATED BY A CONTROLLED INJECTION FROM A HEATED ACCUMULATOR INTO THE INTACT LOOP HOT LEG.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM - SAFETY PROGRAM - NRC-2 - STEAM GENERATOR - FAILURE, TUBING - ACCIDENT, LOSS OF COOLANT - REACTOR, PWR - THERMAL EXPERIMENT - BLOWDOWN - EMERGENCY COOLING - PRESSURE TRANSIENT - THERMAL TRANSIENT

132807

COLLINS BL - SACKETT KE - COPPIN CE
EXPERIMENT DATA REPORT FOR SEMISCALE MOD TEST S-28-5 (STEAM GENERATOR TUBE RUPTURE TEST)
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-MUREG-1152 - 250 PPS, 12 TABS, 386 FIGS, REFS, NOV. 1977

128007 *CONTINUED*

TEST S-20-5 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15 700 KPA AND 5% K TO INVESTIGATE THE RESPONSE OF THE SEMISCALE RED-1 SYSTEM TO A DEPRESSURIZATION AND REFLOOD TRANSIENT FOLLOWING A SIMULATED DOUBLE-ENDED OFFSET SHEAR OF THE BROKEN LOOP COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INTACT AND BROKEN LOOPS TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR. FOR TEST S-20-5, ACCUMULATOR INJECTION INTO THE INTACT LOOP HOT LEG WAS PROVIDED TO SIMULATE THE RUPTURE OF 20 STEAM GENERATOR TUBES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND S PROGRAM • SAFETY PROGRAM • NRC-2 • STEAM GENERATOR • ACCIDENT, LOSS OF COOLANT • BLOWDOWN • REACTOR, PWR • ACCUMULATORS • EMERGENCY COOLING SYSTEM • PRESSURE TRANSIENT • THERMAL TRANSIENT • FAILURE, TUBING • HYDRAULIC EXPERIMENT • OUT OF PILE EXPERIMENT

128009

PHYTON RL • SACRETT RE • COPPIN CE
DISPERSED DATA REPORT FOR SEMISCALE RED-1 TEST S-20-6 (STEAM GENERATOR TUBE RUPTURE TEST)
IDAND NATIONAL ENGINEERING LAB., IDAND FALLS
THERM-NUREG-1101 •, 200 PPS, 7 FIGS, 8 REFS, NOV. 1977

TEST S-20-6 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15 770 KPA AND K TO INVESTIGATE THE RESPONSE OF THE SEMISCALE RED-1 SYSTEM TO A DEPRESSURIZATION AND REFLOOD TRANSIENT FOLLOWING A SIMULATED DOUBLE-ENDED OFFSET SHEAR OF THE BROKEN LOOP COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INTACT AND BROKEN LOOPS TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR. SIXTEEN STEAM GENERATOR TUBE RUPTURES WERE SIMULATED BY A CONTROLLED INJECTION FROM A HEATED ACCUMULATOR INTO THE INTACT LOOP HOT LEG.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND S PROGRAM • SAFETY PROGRAM • NRC-2 • STEAM GENERATOR • ACCIDENT, LOSS OF COOLANT • BLOWDOWN • REACTOR, PWR • ACCUMULATORS • EMERGENCY COOLING SYSTEM • PRESSURE TRANSIENT • THERMAL TRANSIENT • FAILURE, TUBING • HYDRAULIC EXPERIMENT • OUT OF PILE EXPERIMENT

128000

CHAPPELL JR
LEFT BROKEN LOOP MASS FLOW MEASUREMENTS UNCERTAINTY ANALYSIS
IDAND NATIONAL ENGINEERING LAB., IDAND FALLS
THERM-NUREG-1101 •, 32 PPS, 3 TABS, 33 FIGS, 8 REFS, NOV. 1977

THE LOSS-OF-FLUID (LOFT) MEASUREMENTS RELATING TO MASS FLOW RATE FROM THE BROKEN LOOP ARE EXAMINED. DATA FROM EXPERIMENTS LI-1, LI-2, LI-3 AND LI-3A ARE USED TO DETERMINE THE MASS FLOW RATES AND THEIR CORRESPONDING UNCERTAINTIES. THE 2 SIGMA UNCERTAINTIES ARE DETERMINED FOR EACH MEASUREMENT LOCATION, EXPERIMENT AND METHOD OF CALCULATION FOR THE TIME INTERVALS 0 TO 10, 10 TO 30, 30 TO 60, 60 TO 90, AND 90 TO 120 SECONDS AFTER RUPTURE. THE UNCERTAINTIES ARE EXPRESSED AS PERCENTAGES OF THE MAXIMUM MEAN FLOW RATE OBSERVED DURING THE FIRST 60 SECONDS FOR THE PARTICULAR LOCATION. THE UNCERTAINTIES ARE EXCESSIVELY LARGE - ON THE ORDER OF 12.5% AFTER BEING CORRECTED WITH INTEGRAL CORRECTION FACTORS. RECOMMENDATIONS FOR IMPROVING THE MEASUREMENT ACCURACY ARE GIVEN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND S PROGRAM • SAFETY PROGRAM • NRC-2 • LOFT (S-R) • BLOWDOWN • MEASUREMENT • ACCIDENT, LOSS OF COOLANT • FLOW THEORY AND EXPERIMENTS • INSTRUMENT, FLOW • ERROR ANALYSIS

127116

ESTRADA H • SHEPPARD JO
SOME ASPECTS OF INTERPRETING TWO-PHASE FLOW MEASUREMENTS IN INSTRUMENTED PIPING SPOOL PIECES
OAK RIDGE NATIONAL LAB., TENN. • NRC ASSOC., WASHINGTON, D.C.
NUREG-0200 •, 56 PPS, 2 TABS, 9 FIGS, 13 REFS, JUNE 1977

A DISCUSSION IS PRESENTED OF FACTORS AFFECTING THE DETERMINATION OF TWO-PHASE MASS FLOW RATE AND QUALITY UTILIZING DATA FROM INSTRUMENTED PIPING SPOOL PIECES SUCH AS THOSE OF THE OAK THERMAL HYDRAULICS TEST FACILITY. STEADY-STATE AIR-WATER DATA WERE CORRELATED WITH SEPARATED FLOW MODELS WHICH INCLUDED THREE DIFFERENT TURBINE MODELS. RESULTS SHOWED LITTLE DIFFERENCE IN THE USE OF EITHER THE AIA OR ROUGHNESS TURBINE MODELS FOR QUALITY UP TO APPROX. 30%; A MODEL WHICH WEIGHTED PHASE VELOCITIES ACCORDING TO VOID FRACTION WAS NOT REPRESENTATIVE OF THE DATA. ALSO PRESENTED IS A MODEL FOR THE DYNAMICS OF A TURBINE ROTOR IN TRANSIENT TWO-PHASE FLOW.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FLOW, TWO PHASE • OUT OF PILE EXPERIMENT • INSTRUMENT, FLOW • TURBINE • ANALYTICAL MODEL • OAK • NRC-2 • FLOW THEORY AND EXPERIMENTS

127836

SHEPPARD JO • LEAVELL WM • SHAMROKH P • BYRN MC
QUARTERLY PROGRESS REPORT ON THE ADVANCED TWO-PHASE INSTRUMENTATION PROGRAM FOR JANUARY-MARCH 1977
OAK RIDGE NATIONAL LAB., TENN.
OAK/NUREG/TH-119 •, 13 PPS, 3 FIGS, 6 REFS, JULY 1977

127935 *CONTINUED*

AN INSTRUMENTED PIPE SPOOL PIECE HAS BEEN DESIGNED TO IMPROVE THE ACCURACY AND PRECISION OF TWO-PHASE FLOW MEASUREMENTS. THE ALGORITHM FOR CALCULATING NON-STATIONARY VELOCITIES BY INSTRUMENT SIGNAL CORRELATION WAS MODIFIED TO USE THE NORMALIZED CROSS-COVARIANCE FUNCTION RATHER THAN THE CROSS-CORRELATION FUNCTION TO DETERMINE TRANSIT TIME. A PRELIMINARY FEASIBILITY STUDY OF INNOVATIVE TWO-PHASE FLOW INSTRUMENTS HAS BEEN CONDUCTED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

FLOW, TWO PHASE * INSTRUMENT, FLOW * MEASUREMENT * FLOW * EQUIPMENT DEVELOPMENT * NRC-2

131100

THOMAS DG * BANCUM WE * BOHANNAN RE * EADS BG

OAK RIDGE NATIONAL LAB., TENN.

OAK RIDGE NATIONAL LAB., TENN.

ORNL/NUREG/TR-100 * 86 PPS, TABS, FIGS, REFS, AUG, 1977

DEPRESSURIZATION RATES AND MASS FLOWS WERE QUALITATIVELY SIMILAR TO THOSE PREDICTED FOR PRESSURIZED-WATER REACTORS IN SAFETY ANALYSIS REPORTS. TESTS WERE MADE FROM AN INITIAL HEATER ROD POWER OF 80 KW/MOD WITH OUTLET SUBCOOLING AS THE PRINCIPAL INDEPENDENT VARIABLE. MEAN TIME TO CRITICAL HEAT FLUX (CHF) IN THE THREE TESTS WAS 1.2 TO 2.8 SEC. WITH A RANGE FROM 0.8 TO 3.6 SEC. TESTS WERE INITIATED WITH A TRIPLE-BEAM GAMMA DENSITOMETER ON THE AIR-WATER LOOP, WITH FLOW IN THE VERTICAL DOWNWARD DIRECTION. PRELIMINARY MEASUREMENTS INDICATED THAT WITH FLOW DISPENSERS LOCATED UPSTREAM OF THE DENSITOMETER, THE MEAN DENSITY OBTAINED BY AVERAGING THE THREE-BEAM DATA WAS IN CLOSE AGREEMENT WITH DENSITY MEASUREMENTS MADE BY THE DIAMETRAL BEAM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

R AND D PROGRAM * SAFETY PROGRAM * REACTOR, PWR * ACCIDENT, LOSS OF COOLANT * BLOWDOWN * OUT OF PILE EXPERIMENT * EMERGENCY COOLING SYSTEM * SAFETY INJECTION * COMPUTER PROGRAM * HEAT TRANSFER EXPERIMENT * ORNL * HEATERS * FLOW, TWO PHASE * FLOW THEORY AND EXPERIMENTS * THERMAL-HYDRAULIC ANALYSIS * NRC-2 * JACOBS

127900

WHITE RD * MEDRICK RA

PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM - THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT

FOR TEST 100

OAK RIDGE NATIONAL LAB., TENN.

ORNL/NUREG/TR-110 * 25 PPS, 5 TABS, 4 FIGS, AUG, 1977

REDUCED INSTRUMENT RESPONSES ARE PRESENTED FOR THERMAL-HYDRAULIC TEST FACILITY (THTF) TEST 100, WHICH IS PART OF THE ORNL PRESSURIZED-WATER REACTOR (PWR) BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM. THE OBJECTIVE OF THE PROGRAM IS TO INVESTIGATE THE THERMAL-HYDRAULIC PHENOMENON GOVERNING THE ENERGY TRANSFER AND TRANSPORT PROCESSES THAT OCCUR DURING A LOSS-OF-COOLANT ACCIDENT IN A PWR SYSTEM. TEST 100 WAS CONDUCTED TO INVESTIGATE THE RESPONSE OF HEATER ROD BUNDLE 1 AND INSTRUMENTED SPOOL PIECES WITH FLOW HOMOGENIZING SCREENS TO A DOUBLE-ENDED RUPTURE WITH EQUAL BREAK AREAS AT THE TEST SECTION INLET AND OUTLET.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

REACTOR, PWR * ACCIDENT, LOSS OF COOLANT * BLOWDOWN * OUT OF PILE EXPERIMENT * THERMAL EXPERIMENT * HYDRAULIC EXPERIMENT * ORNL * FLOW, TWO PHASE * NRC-2

127600

WHITE RD * MEDRICK RA

PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM - THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT

FOR TEST 101

OAK RIDGE NATIONAL LAB., TENN.

ORNL/NUREG/TR-129 * 28 PPS, 4 TABS, 5 FIGS, SEPT, 1977

REDUCED INSTRUMENT RESPONSES ARE PRESENTED FOR THERMAL-HYDRAULIC TEST FACILITY (THTF) TEST 101, WHICH IS PART OF THE ORNL PRESSURIZED-WATER REACTOR (PWR) BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM. THE OBJECTIVE OF THE PROGRAM IS TO INVESTIGATE THE THERMAL-HYDRAULIC PHENOMENON GOVERNING THE ENERGY TRANSFER AND TRANSPORT PROCESSES THAT OCCUR DURING A LOSS-OF-COOLANT ACCIDENT IN A PWR SYSTEM. TEST 101 WAS CONDUCTED TO INVESTIGATE THE THERMAL-HYDRAULIC RESPONSE OF BUNDLE 1 AND THE MAIN HEAT EXCHANGERS TO POWER OPERATION AND A SINGLE-ENDED RUPTURE AT THE TEST SECTION OUTLET.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

REACTOR, PWR * BLOWDOWN * OUT OF PILE EXPERIMENT * EMERGENCY COOLING SYSTEM * CORE REPLEADING SYSTEM * ACCIDENT, LOSS OF COOLANT * THERMAL EXPERIMENT * HYDRAULIC EXPERIMENT * NRC-2 * JACOBS

131101

MEDRICK RA * CRAODICK VG * HYAM CR * TURNAGE RG

PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM DATA EVALUATION REPORT--SYSTEM RESPONSE FOR THERMAL-

HYDRAULIC TEST FACILITY TEST SERIES 100

OAK RIDGE NATIONAL LAB., TENN.

ORNL/NUREG-19 * 132 PPS, FIGS, 15 REFS, NOV, 1977

131101 *CONTINUED*

THE EXPERIMENTAL DATA PRESENTED IN THIS REPORT REPRESENT THE MOST SIGNIFICANT ASPECTS OF THE OVERALL SYSTEM RESPONSE FOR TESTS 100 THROUGH 105. THESE TESTS ENCOMPASS THE EXPERIMENTAL CHECKOUT OF THE FACILITY AND ITS ABILITY TO PRODUCE THE TRANSIENT DATA REQUIRED. THEY ALSO INCLUDE BASELINE TESTS FOR DETERMINING THE INPUT PARAMETERS NECESSARY FOR THE VERIFICATION OF LARGE DIGITAL SIMULATORS SUCH AS RELAP/WHOS (UPDATE 2). TESTS 103, 104, AND 105 PROVIDE A SIGNIFICANT DATA BASE FOR USE IN INVESTIGATING THERMAL-HYDRAULIC PHENOMENA ASSOCIATED WITH LOSS-OF-COOLANT ACCIDENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, LWR • ACCIDENT, LOSS OF COOLANT • BLOWDOWN • OUT OF PILE EXPERIMENT • HEAT TRANSFER EXPERIMENT • ORNL • HEATERS • NRC-2 • THERMAL HYDRAULIC ANALYSIS

130904

OTT LJ • KERRICK RA
ORNL - A ONE DIMENSIONAL IMPLICIT APPROACH TO THE INVERSE HEAT CONDUCTION PROBLEM
ORNL RISE NATIONAL LAB., TENN.
ORNL/NUMES-23 • 300 PPS, TABS, FIGS, NOV. 1977

THIS REPORT DEVELOPS AN IMPLICIT SOLUTION TECHNIQUE TO DETERMINE BOTH THE TRANSIENT SURFACE TEMPERATURE AND THE TRANSIENT SURFACE HEAT FLUX OF ELECTRICALLY HEATED RODS GIVEN THE POWER INPUT AND AN "INDICATED" INTERNAL TEMPERATURE DURING A SIMULATED LOSS-OF-COOLANT ACCIDENT. A DIGITAL COMPUTER PROGRAM ORNLIC IS DEVELOPED WHICH SOLVES A ONE-DIMENSIONAL, TRANSIENT, LUMPED PARAMETER, IMPLICIT FORMULATION OF THE CONDUCTION EQUATION AT EACH BUNDLE THERMOCOUPLE POSITION IN THE THERMAL-HYDRAULIC TEST FACILITY (THTF).

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

THERMAL TRANSIENT • HEAT TRANSFER, CONDUCTION • ACCIDENT, LOSS OF COOLANT • FUEL ROD • NRC-2 • COMPUTER PROGRAM

131105

THOMAS BF • JOHNSON RF • HADPHERSON RE • SHEPPARD JO
QUARTERLY PROGRESS REPORT ON BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM FOR APRIL-JUNE 1977
ORNL RISE NATIONAL LAB., TENN.
ORNL/NUMES/TH-130 • 62 PPS, TABS, FIGS, REFS, NOV. 1977

THE THERMAL-HYDRAULIC TEST FACILITY HAS COMPLETED 16 POWERED ROD BLOWDOWNS THROUGH JUNE 16, 1977. OF THESE 5 WERE COMPLETED WITH ALL 49 RODS POWERED, 2 WERE COMPLETED WITH 2 INACTIVE RODS, AND 9 WERE COMPLETED WITH 1 INACTIVE RODS. INITIAL SYSTEM PRESSURE IN THESE TESTS WAS APPROXIMATELY 15.5 MPa (APPROXIMATELY 2250 PSIG). TEST SECTION INLET TEMPERATURE WAS APPROXIMATELY 200 DEG C (APPROXIMATELY 367 DEG F), AND BREAK AREA WAS EQUIVALENT TO A 2005 BREAK WITH THE TOTAL AREA USUALLY SPLIT BETWEEN INLET AND OUTLET IN THE RATIO 0.40:0.60. HEATER ROD POWER IN THESE TESTS WAS 90, 100, OR 122 KW/ROD. MEAN TIME TO CRITICAL HEAT FLUX (CHF) VARIED FROM 0.7 TO 1.4 SEC. WITH DELAYED CHF OF APPROXIMATELY 2.3 SEC IN THE UPPER HALF OF THE BUNDLE IN SOME TESTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, PWR • ACCIDENT, LOSS OF COOLANT • BLOWDOWN • OUT OF PILE EXPERIMENT • EMERGENCY COOLING SYSTEM • SAFETY INJECTION • COMPUTER PROGRAM • HEAT TRANSFER EXPERIMENT • ORNL • HEATERS • FLOW, TWO PHASE • FLOW THEORY AND EXPERIMENTS • THERMAL HYDRAULIC ANALYSIS • NRC-2 • JACOBS

130816

LEVALL BN • HATES PH • KENED CD • JALLOM PA • SHANPOKHI P
ADVANCED TWO-PHASE INSTRUMENTATION PROGRAM QUARTERLY PROGRESS REPORT FOR APRIL-JUNE 1977
ORNL RISE NATIONAL LAB., TENN.
ORNL/NUMES/TH-148 • 15 PPS, 2 TABS, 9 FIGS, 5 REFS, DEC. 1977

THE OBJECTIVE OF THE ADVANCED TWO-PHASE INSTRUMENTATION PROGRAM IS THE APPLICATION OF ADVANCED INSTRUMENTATION SCIENCE TO IMPROVE THE ACCURACY AND PRECISION OF THE TRANSIENT TWO-PHASE MEASUREMENTS REQUIRED IN REACTOR SAFETY RESEARCH. A CONDUCTIVITY PROBE SYSTEM CONSISTING OF TWO IDENTICAL ELECTRODE ASSEMBLIES 25 MM (1 IN.) APART AND ROTATED 90 DEGREES WITH RESPECT TO EACH OTHER WAS TESTED IN AN AIR-WATER LOOP AT VOID FRACTIONS OF 3 AND 40%, RELATIVELY HIGH CONDENSANCE WAS OBTAINED BETWEEN THE TWO ELECTRODE ASSEMBLIES, AND THE VELOCITY MEASUREMENTS OBTAINED FROM THE EXPERIMENTALLY DETERMINED PHASE SHIFT CURVES WERE IN REASONABLE AGREEMENT WITH THOSE CALCULATED FROM GENERAL TWO-PHASE FLOW PRINCIPLES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

INSTRUMENT, COMPONENT • FLOW, TWO PHASE • MEASUREMENT • FLOW, TWO PHASE • INSTRUMENT, FLOW • INSTRUMENT, PROCESS • NRC-2 • EQUIPMENT DEVELOPMENT

131130

RODAL BN • COMBAT CE • KREPIVICH MC
PLECHY LOW FLOWING RATE SKEWED TEST SERIES DATA REPORT
WESTINGHOUSE ELECTRIC CORP., PITTSBURGH, PA
WCAP-9500 • APPROX. 1100 PPS, TABS, 165, REFS, MAY 1977

THE TESTS WERE CONDUCTED IN AN IMPROVED ORIGINAL PLECHY TEST FACILITY TO PROVIDE HEAT TRANSFER

131130 *CONTINUED*

COEFFICIENT AND ENTRAINMENT DATA AT FORCED FLOODING RATES OF 1 IN/SEC. AND WITH ELECTRICALLY HEATED ROD BUNDLES WHICH HAD COSINE AND TOP SKEWED AXIAL POWER PROFILES. THE COSINE AXIAL POWER PROFILE TEST SERIES WERE COMPLETED IN 1975 AND THE TEST RESULTS WERE REPORTED IN WCAP-0051. THESE TESTS EXAMINED THE EFFECTS OF INITIAL CLAD TEMPERATURE, VARIABLE STEPPED AND CONTINUOUSLY VARIABLE FLOODING RATES, HOUSING HEAT RELEASE, NO. PEAK POWER, CONSTANT LOW FLOODING RATES, COOLANT SUBCOOLING, HOT AND COLD CHANNEL ENTRAINMENT, AND BUNDLE STORED AND GENERATED POWER.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545 (60 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

R AND D PROGRAM • SAFETY PROGRAM • FLOODING COEFFICIENT • HEAT TRANSFER COEFFICIENT • TESTING • REACTOR, LOW • SUBCOOLING • MOCKUP • POWER DISTRIBUTION • WRC-2

3. R3 - WATER REACTOR SAFETY RESEARCH, FUEL BEHAVIOR

129177
LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM: QUARTERLY PROGRESS REPORT JANUARY-MARCH 1977
ARGONNE NATIONAL LAB., ILL.
ANL-77-34 +. 120 PPS, TABS, FIGS, REFS, JUNE 1977

THIS PROGRESS REPORT SUMMARIZES THE ARGONNE NATIONAL LABORATORY WORK PERFORMED DURING JANUARY, FEBRUARY, AND MARCH 1977 ON WATER-REACTOR-REACTOR SAFETY PROBLEMS. THE FOLLOWING RESEARCH AND DEVELOPMENT AREAS ARE COVERED: (1) LOSS-OF-COOLANT ACCIDENT RESEARCH: HEAT TRANSFER AND FLUID DYNAMICS; (2) TRANSIENT FUEL RESPONSE AND FISSION-PRODUCT RELEASE PROGRAM; (3) MECHANICAL PROPERTIES OF ZIRCALOY CONTAINING OXYGEN; AND (4) STEAM-EXPLOSION STUDIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
REACTOR, LWR + R AND D PROGRAM + ANAL + SAFETY ANALYSIS + ACCIDENT, LOSS OF COOLANT + HEAT TRANSFER ANALYSIS +
FLOW STABILITY + FISSION PRODUCT RELEASE + PROPERTY, MECHANICAL + ZIRCALOY + CLADDING + STEAM + EXPLOSION +
NRC-3 + NRC-6

129183
LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM: QUARTERLY PROGRESS REPORT, APRIL-JUNE 1977
ARGONNE NATIONAL LAB., ILL.
ANL-77-35 +. 120 PPS, TABS, FIGS, REFS, DEC. 1977

THIS PROGRESS REPORT SUMMARIZES THE ARGONNE NATIONAL LABORATORY WORK PERFORMED ON WATER-REACTOR-REACTOR SAFETY PROBLEMS. THE FOLLOWING RESEARCH AND DEVELOPMENT AREAS ARE COVERED: (1) LOSS-OF-COOLANT ACCIDENT RESEARCH: HEAT TRANSFER AND FLUID DYNAMICS; (2) TRANSIENT FUEL RESPONSE AND FISSION-PRODUCT RELEASE PROGRAM; (3) MECHANICAL PROPERTIES OF ZIRCALOY CONTAINING OXYGEN; AND (4) STEAM-EXPLOSION STUDIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
REACTOR, LWR + R AND D PROGRAM + ANAL + SAFETY ANALYSIS + ACCIDENT, LOSS OF COOLANT + HEAT TRANSFER ANALYSIS +
FLOW STABILITY + FISSION PRODUCT RELEASE + PROPERTY, MECHANICAL + ZIRCALOY + CLADDING + STEAM + EXPLOSION +
NRC-3 + NRC-6

127846
DAUER AA + LOWRY LH + PERRIN JS
EVALUATING STRENGTH AND DUCTILITY OF IRRADIATED ZIRCALOY. QUARTERLY PROGRESS REPORT JANUARY THROUGH MARCH, 1977
BATTILLE COLUMBUS LAB., OHIO
OBI-MR66-1976 +. 30 PPS, 3 TABS, 13 FIGS, APRIL 1977

THE CHARACTERIZATION OF 10 OCONEE FUEL RODS HAS PROCEEDED THROUGH VISUAL EXAMINATION, PROFILOMETRY, AND EDDY-CURRENT EXAMINATION OF THE RODS. MARKING AND SECTIONING TO PROVIDE SPECIMENS FOR MECHANICAL PROPERTY TESTS HAVE BEGUN. STRENGTH PROPERTIES ARE OBSERVED TO DECREASE WITH INCREASE IN TEMPERATURE AND DECREASE IN STRAIN RATE, WHILE DUCTILITY PROPERTIES EXHIBIT THE OPPOSITE TREND. ANNEALING OF IRRADIATION DAMAGE IS ACCOMPANIED BY A STRENGTH DECREASE THAT PRECEDES RECOVERY OF DUCTILITY PROPERTIES; DUCTILITY PROPERTIES FIRST DECREASE AND THEN INCREASE AS RECOVERY OCCURS. THE TUBE BURST PROPERTIES SHOW AN ANNEALING DEPENDENCE SIMILAR TO THAT DETERMINED IN TENSILE TESTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
ZIRCALOY + IRRADIATION TESTING + DUCTILITY + EXAMINATION + FUEL ROD + OCONEE 1 (PWR) + NRC-3

126636
DAUER AA + LOWRY LH + PERRIN JS
EVALUATING STRENGTH AND DUCTILITY OF IRRADIATED ZIRCONIUM. QUARTERLY PROGRESS REPORT, APRIL-JUNE 1977
BATTILLE COLUMBUS LAB., OHIO
OBI-MR66-1976 +. 40 PPS, JULY 1977

THREE M. B. ROBINSON FUEL RODS WERE CHARACTERIZED BY VISUAL EXAMINATION, GAMMA SCANNING, PROFILOMETRY, EDDY CURRENT INSPECTION, A C PLENUM-GAS MEASUREMENT. METALLOGRAPHIC EXAMINATION OF POSSIBLE DEFECT AREAS AS INDICATED BY EDDY-CURRENT INSPECTION REVEALED NO EVIDENCE OF TUBE-WALL DEFECTS. TENSILE TESTS OF OCONEE FUEL-ROD CLADDING SHOW THAT THE ULTIMATE AND YIELD STRENGTHS DECREASE LINEARLY WITH TEMPERATURE FROM ROOM TEMPERATURE TO 800 F. TUBE WALL-THICKNESS AND DIAMETER MEASUREMENTS AS A FUNCTION OF DISTANCE FROM THE POINT OF FRACTURE IN A TENSILE SPECIMEN HAVE BEEN MADE ON IRRADIATED-AND-ANNEALED M. B. ROBINSON CLADDING AND AN UNIRRADIATED SANDVICH TUBE. THE BEHAVIOR OF THE RODS HIGHLY ANNEALED MATERIAL IS SIMILAR TO THAT OF THE UNIRRADIATED TUBING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
ZIRCONIUM + DUCTILITY + STRUCTURAL INTEGRITY + EXAMINATION + TENSILE PROPERTY + FUEL ROD + OCONEE 1 (PWR) +
ROBINSON 2 (PWR) + JACOBS + NRC-3

129085
GIESBRE JA + JORDAN H + BAYBUTT P + EOSTON RO + DENNING RS
FISSION PRODUCT TRANSPORT ANALYSIS - QUARTERLY PROGRESS REPORT FOR APRIL THROUGH JUNE 1977

120501 CONTINUED
 BATTTELLE COLUMBUS LABS., OHIO
 DBI-MUREG-1901 +. 17 PPS. SEPT. 30, 1977

THIS PROGRESS REPORT COVERS CONTINUING ACTIVITIES ASSOCIATED WITH MODELING THE TRANSPORT AND DEPOSITION OF FISSION PRODUCTS WITHIN PWR AND BWR PRIMARY SYSTEMS DURING THE REFLOOD TIME PERIOD FOLLOWING A TERMINATED LOCA. THE ORIGINAL SCOPE OF THE OVERALL PROJECT HAS BEEN EXPANDED TO INCLUDE CONSIDERATION OF CONDITIONS CONSISTENT WITH THOSE LEADING TO POSTULATED CORE MELTDOWN SITUATIONS. THUS, MAJOR EFFORTS ARE TO BE CONCERNED WITH THE ANALYSIS OF POSTULATED MELTDOWN CONDITIONS AND THE EXTENSION AND REVISION OF THE TRAP CODES FOR APPLICABILITY TO THESE CONDITIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS + FISSION PRODUCT TRANSPORT + ANALYTICAL MODEL + CORE REFLOODING + CORE MELTDOWN + COMPUTER PROGRAM + DEPOSITION + NRC-3 + NRC-4

120700
 BAKER AA + GALLAGHER WJ + LOVRY LM + MARKWORTH AJ
 EVALUATING STRENGTH AND DUCTILITY OF IRRADIATED ZIRCALOY. QUARTERLY PROGRESS REPORT JULY THROUGH SEPTEMBER 1977
 BATTTELLE COLUMBUS LABS., OHIO
 DBI-MUREG-1905 +. 32 PPS. 3 TABS. 16 FIGS. OCT. 1977

ISOTHERMAL TUBE-BURST TEST RESULTS OBTAINED ON H. B. ROBINSON SPENT-FUEL CLADDING ARE REPORTED. ANALYSIS OF ISOTHERMAL-ANNEALING, TENSILE-TEST DATA INDICATE A FIRST-ORDER TIME DEPENDENCE FOR THE ANNEALING KINETICS WITH AN ACTIVATION ENERGY SIMILAR TO THAT FOR ZIRCONIUM SELF-DIFFUSION. TRANSIENT TUBE-BURST TESTS OF THE CLADDING CONDUCTED AT INITIAL PRESSURES RANGING FROM 2100 TO 10,500 KPA, PRODUCED BURST TEMPERATURE AND BURST DUCTILITY MEASUREMENTS SIMILAR TO THOSE RECORDED FOR UNIRRADIATED ZIRCALOY TUBING. DIFFERENCES IN BURST TEMPERATURE ARE ASSOCIATED WITH AXIAL AND ANGULAR TUBE TEMPERATURE VARIATIONS. NO INFLUENCE OF AXIAL RESTRAINT DURING THE TUBE BALLOONING AND FAILURE STAGES HAS BEEN OBSERVED ON TUBE BURST BEHAVIOR.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + IRRADIATION TESTING + FUEL ROD + CLADDING + REACTOR, PWR + ROBINSON 2 (PWR) + FAILURE, CLADDING + THERMAL TRANSIENT + NRC-3 + ZIRCALOY

122033
 CIESERE JA + BATHWELL P + JORDAN M + DENNING RS
 FISSION PRODUCT TRANSPORT ANALYSIS: QUARTERLY PROGRESS REPORT FOR JULY THROUGH SEPTEMBER, 1977
 BATTTELLE COLUMBUS LABS., OHIO
 DBI-MUREG-1906 +. 27 PPS. 4 TABS. 5 FIGS. 12 REFS. NOV. 30, 1977

TECHNICAL PROGRESS DURING THIS QUARTER CONSISTED OF COMPLETING AND CHECKING THE MODEL FOR EMERGENCY CORE COOLING SYSTEM (ECCS) WATER SCRUBBING OF FISSION PRODUCTS FOR INCLUSION IN THE TRAP CODE, IDENTIFYING FISSION PRODUCT SOURCE TERMS DATA AND INFORMATION FOR INITIAL USE UNDER ASSUMED MELTDOWN CONDITIONS, SPECIFYING THE THERMAL-HYDRAULIC CONDITIONS TO BE CONSIDERED FOR FISSION PRODUCT TRANSPORT IN POSTULATED MELTDOWN ACCIDENTS, AND CHOOSING A CONTROL VOLUME FRAMEWORK FOR INITIAL USE IN DEVELOPMENT OF THE TRANSPORT MODEL FOR MELTDOWN CONDITIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + FISSION PRODUCT TRANSPORT + ANALYTICAL MODEL + CORE REFLOODING + CORE MELTDOWN + COMPUTER PROGRAM + DEPOSITION + NRC-3 + NRC-4

120370
 BAILEY WJ
 QUARTERLY PROGRESS REPORT TO NUCLEAR REGULATORY COMMISSION DIVISION OF REACTOR SAFETY RESEARCH FUEL BEHAVIOR RESEARCH BRANCH PROGRAMS FOR PERIOD ENDING MARCH 1977
 BATTTELLE PACIFIC NORTHWEST LABS., RICHLAND, WASH.
 BNWL-2270 +. 100 PPS. 27 TABS. 79 FIGS. 65 REFS. APRIL 1977

THIS IS THE FIRST QUARTERLY REPORT ISSUED ON THREE PROGRAMS AT BATTTELLE, PACIFIC NORTHWEST LABORATORIES (PNL) SPONSORED BY THE FUEL BEHAVIOR RESEARCH BRANCH OF THE DIVISION OF REACTOR SAFETY RESEARCH (DSR) OF THE NUCLEAR REGULATORY COMMISSION (NRC). THIS REPORT COVERS THE REPORTING PERIOD THROUGH MARCH 1977 AND INCLUDES BACKGROUND INFORMATION ON WORK PERFORMED PRIOR TO JANUARY 1977. THE 3 PROGRAMS ARE EXPERIMENTAL VERIFICATION OF STEADY-STATE CODES PROGRAM, FUEL SUBASSEMBLY PROCUREMENT AND IRRADIATION TEST PROGRAM, AND PRESSURE TRANSDUCER DEVELOPMENT AND TEST TRAIN INSTRUMENTATION QUALIFICATION PROGRAM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + BATTTELLE NORTHWEST + AGENCY, NRC + STEADY STATE + COMPUTER PROGRAM + FUEL ELEMENTS + IRRADIATION TESTING + INSTRUMENT, PRESSURE + QUALIFICATION + NRC-3 + JACOBS

120401
 HARRIS CR + MARSHALL RE
 COMPARATIVE ANALYSIS OF PELLETT-CLADDING INTERACTION FROM IFA-831 AND IFA-832 HALDEN REACTOR TESTS
 BATTTELLE PACIFIC NORTHWEST LABS., RICHLAND, WASH.

129401 *CONTINUED*

ORNL-2240 +. 48 PPS. NEVS. JULY 1977

TWO ESSENTIALLY IDENTICAL TEST ASSEMBLIES, EACH CONTAINING SIX INSTRUMENTED FUEL RODS, WERE IRRADIATED UNDER SIMILAR CONDITIONS IN THE HWRP. PARAMETERS IN THE TEST INCLUDE PELLET-CLADDING GAP SIZE AND/OR GAP ECCENTRICITY, FILL GAS COMPOSITION, FUEL DENSITY AND STABILITY, LINEAR HEAT RATING, AND BURNUP. THE CONCLUSIONS ARE AS FOLLOWS: THE UNSTABLE FUEL CAUSED LESS CLADDING ELONGATION THAN THE REFERENCE (STABLE) FUEL. FUEL HAVING A SIGNIFICANTLY LOWER THERMAL CONDUCTIVITY THAN THE REFERENCE FUEL CAUSED GREATER CLADDING ELONGATIONS THAN THE REFERENCE FUEL. AT A BURNUP OF APPROXIMATELY 3000 MWG/WTM, PCI OCCURRED IN ALL RODS REGARDLESS OF INITIAL GAP SIZE OR FUEL TYPE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS + FUEL CLAD INTERACTION + FUEL, PELLET TYPE + COMPUTER PROGRAM + IRRADIATION TESTING + NRC-3

129163

CARD ON

QUARTERLY TECHNICAL PROGRESS REPORT ON BWR REACTOR SAFETY PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH (JAN-MAR 1977)
IDAND NATIONAL ENGINEERING LAB., IDAND FALLS
TRNS-NURES-1180 +. 138 PPS. 17 TABS. 75 FIGS. NEVS. JUNE 1977

LIGHT WATER REACTOR SAFETY RESEARCH PERFORMED JANUARY THROUGH MARCH 1977 IS DISCUSSED, INCLUDING THE FOLLOWING: A TEST SERIES TO EVALUATE THE EFFECTIVENESS OF DIFFERENT EMERGENCY CORE COOLANT INJECTION ARRANGEMENTS WAS COMPLETED. A LOFT TEST SERIES WAS INITIATED TO INVESTIGATE THE EFFECT OF BREAK NOZZLE GEOMETRY ON BREAK FLOW AND SYSTEM RESPONSE DURING A 200S DOUBLE-ENDED COLD LEG BREAK TEST. THE ANALYSIS OF RESULTS OF LOFT NONNUCLEAR EXPERIMENTS WERE MADE. A REVIEW OF FLEHT AND SERISCALE FORCED FLOW AND GRAVITY FLOW REPLED HEAT TRANSFER EXPERIMENTS WAS PERFORMED. A MORE GENERAL VERSION OF THE RELAPS PILOT CODE (VERSION 2) WAS PROGRAMMED. EXPERIMENT RESULTS FROM TESTS CONDUCTED AT THE MARVIRN CONTAINMENT TEST FACILITY IN SWEDEN AND THE BATTLE TEST FACILITY IN FRANKFURT, GERMANY, WERE USED FOR VERIFICATION OF COMPUTER CONTAINMENT CODES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, LWR + SAFETY ANALYSIS + OUT OF PILE EXPERIMENT + COMPUTER PROGRAM, DIGITAL + COMPARISON, THEORY AND EXPERIENCE + EMERGENCY COOLING SYSTEM + BLOWDOWN + ACCIDENT, LOSS OF COOLANT + NRC-2 + NRC-3 + NRC-4

130620

ALLISON CH + FLOER SA + CROUCHER DV + HENNER AS
IRRADIATION EFFECTS TEST SERIES TEST IE-2 TEST RESULTS REPORT
IDAND NATIONAL ENGINEERING LAB., IDAND FALLS
TRNS-NURES-1074 +. 189 PPS. AUG. 1977

DESCRIBES THE RESULTS OF THE FOURTH TEST, TEST IE-2, USING FOUR, 0.97- .006, PUR-TYPE FUEL RODS WITH DIFFERENCES IN DIAMETRAL GAP AND CLADDING IRRADIATION. THE OBJECTIVE WAS TO PROVIDE INFORMATION ABOUT THE EFFECTS OF THESE DIFFERENCES ON FUEL AND BEHAVIOR DURING QUASE-EQUILIBRIUM AND FILM BOILING OPERATION. DATA ARE PRESENTED ON THE BEHAVIOR OF FUEL RODS DURING QUASE-EQUILIBRIUM AND DURING FILM BOILING OPERATION. THE EFFECTS OF INITIAL GAP SIZE, CLADDING IRRADIATION, ROD POWER CYCLING, A RAPID POWER INCREASE, AND SUSTAINED FILM BOILING ARE DISCUSSED. THESE DISCUSSIONS ARE BASED ON MEASURED TEST DATA, PRELIMINARY POSTIRRADIATION EXAMINATION RESULTS, AND COMPARISONS OF RESULTS WITH PRAP-V3 COMPUTER MODEL CALCULATIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS + R AND D PROGRAM + FUEL ROD + REACTOR, PUR + IRRADIATION TESTING + FILM BOILING + NRC-3

130660

QUARTERLY TECHNICAL PROGRESS REPORT ON BWR REACTOR SAFETY PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH APRIL-JUNE 1977
IDAND NATIONAL ENGINEERING LAB., IDAND FALLS
TRNS-NURES-1147 +. 92 PPS. AUG. 1977

RESULTS FROM THE PREVIOUSLY CONDUCTED SERISCALE ROD-1 ECC INJECTION TEST SERIES WERE ANALYZED. TESTING IN THE LOFT COUNTERPART TEST SERIES WAS ESSENTIALLY COMPLETED, AND THE STEAM GENERATOR TUBE RUPTURE TEST SERIES WAS BEGUN. TWO TESTS IN THE ALTERNATE ECC INJECTION TEST SERIES WERE CONDUCTED WHICH INCLUDED INJECTION OF EMERGENCY CORE COOLANT INTO THE UPPER PLENUM THROUGH USE OF THE LOW PRESSURE INJECTION SYSTEM. THE LOSS-OF-FLUID TEST PROGRAM SUCCESSFULLY COMPLETED NONNUCLEAR LOSS-OF-COOLANT EXPERIMENT LI-6. A NUCLEAR TEST, GC 2-3, IN THE POWER BURST FACILITY REACTOR WAS PERFORMED TO EVALUATE THE POWER OSCILLATION METHOD OF DETERMINING GAP CONDUCTANCE TO DETERMINE THE EFFECTS OF INITIAL GAP SIZE, FILL GAS COMPOSITION, AND FUEL DENSITY ON THE THERMAL PERFORMANCE OF A LIGHT WATER REACTOR FUEL ROD.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + REACTOR, LWR + ACCIDENT, LOSS OF COOLANT + EMERGENCY COOLING SYSTEM + HWR PLUM, CRITICAL + DNB + THERMOCUPLE + HEAT CONDUCTANCE, FUEL TO CLAD + COMPUTER PROGRAM + CROSB BEHAVIOR + THERMAL HYDRAULIC ANALYSIS + COMPARISON, THEORY AND EXPERIENCE + NRC-2 + NRC-3 + NRC-4 + JACOBS

132782

QUAPP DJ + FARRAR LC + MENNER AS + ALLISON CH
IRRADIATION EFFECTS TEST SERIES SCOPING TEST 2 TEST RESULTS REPORT
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NUMBER-1000 ++ 155 PPS, 20 TABS, 102 FIGS, SEPT. 1977

THIS TEST, WHICH WAS CONDUCTED IN THE POWER BURST FACILITY, USED FOUR 0.97-M-LONG, PRESSURIZED WATER REACTOR TYPE FUEL RODS. THREE OF THESE RODS WERE FABRICATED FROM PREVIOUSLY IRRADIATED MATERIALS AND THE FOURTH WAS FABRICATED FROM UNIRRADIATED MATERIALS. THE OBJECTIVES OF THIS SCOPING TEST WERE TO EVALUATE FABRICATION PROCEDURES, HANDLING PLANS, AND REACTOR CAPABILITIES FOR IRRADIATED FUEL ROD MATERIALS AS WELL AS TO ACQUIRE FUEL ROD BEHAVIOR DATA. THIS REPORT PRESENTS THE FUEL BEHAVIOR DATA ACQUIRED DURING THE TEST. THE FUEL RODS WERE SUBJECTED TO A PRECONDITIONING PERIOD, FOLLOWED BY A POWER INCREASE TO A PEAK POWER OF 61 KW/M AND A FLOW REDUCTION TO INDUCE FILM BOILING. IT IS REPORT DESCRIBES THE DATA OBTAINED IN EACH PHASE OF THE TEST AND THE POSTTEST CONDITION OF THE FUEL RODS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + IRRADIATION TESTING + FUEL ROD + FILM BOILING + EXAMINATION + REACTOR, PWR + FLOW + THERMAL TRANSIENT + PDF (S-99) + NRC-3

131940

QUAPP DJ + ALLISON CH + FARRAR LC
IRRADIATION EFFECTS TEST SERIES SCOPING TEST 1 TEST RESULTS REPORT
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NUMBER-1000 ++ 106 PPS, FIGS, SEPT. 1977

DESCRIBES THE RESULTS OF THE FIRST SCOPING TEST IN THE IRRADIATION EFFECTS TEST SERIES. THIS TEST USED AN UNIRRADIATED, THREE-FOOT-LONG, PWR-TYPE FUEL ROD. THE OBJECTIVE OF THIS TEST WAS TO THOROUGHLY EVALUATE THE REMOTE FABRICATION PROCEDURES TO BE USED FOR IRRADIATED RODS IN FUTURE TESTS, HANDLING PLANS, AND REACTOR OPERATIONS. ADDITIONALLY, SELECTED FUEL BEHAVIOR DATA WERE OBTAINED. THE FUEL ROD WAS SUBJECTED TO A SERIES OF PRECONDITIONING POWER CYCLES FOLLOWED BY A POWER INCREASE WHICH BROUGHT THE FUEL ROD POWER TO ABOUT 20.0 KW/FT PEAK LINEAR HEAT RATING. FILM BOILING OCCURRED FOR A PERIOD OF 0.8 MINUTES. THE TEST FUEL ROD FAILED FOLLOWING REACTOR SHUTDOWN AS A RESULT OF HEAVY INTERNAL AND EXTERNAL CLADDING OXIDATION AND EMBRITTLEMENT WHICH OCCURRED DURING THE FILM BOILING OPERATION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + REACTOR, PWR + TESTING + FUEL ELEMENTS + FAILURE, FUEL ELEMENT + EMBRITTLEMENT + OXIDATION + FILM BOILING + STEEL, STAINLESS + NRC-3 + JACOBS + CLADDING

132783

HANDOOK BA
EXPERIMENT DATA REPORT GAP CONDUCTANCE TEST SERIES TEST GC 1-3 POSTIRRADIATION EXAMINATION
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NUMBER-1000 ++ 103 PPS, 20 TABS, 25 FIGS, REFS, SEPT. 1977

THE RESULTS OF THE POSTIRRADIATION EXAMINATION OF FOUR BOILING WATER REACTOR TYPE, ZIRCALOY-CLAD, UO₂-FUELED RODS TESTED AS PART OF THE THERMAL FUELS BEHAVIOR PROGRAM ARE DISCUSSED. THESE RODS WERE EMPLOYED IN GAP CONDUCTANCE TEST GC 1-3 WHICH WAS CONDUCTED TO OBTAIN EXPERIMENTAL DATA FROM WHICH BOTH GAP CONDUCTANCE VALUES COULD BE DETERMINED BY BOTH THE STEADY STATE INTEGRAL OF K DT AND THE POWER OSCILLATION METHODS. EVIDENCE IS PRESENTED SHOWING THAT SIGNIFICANT AMOUNTS OF WATER HAD BEEN PRESENT IN TWO OF THE FOUR FUEL RODS DURING TESTING. FOR THE TWO FUEL RODS THAT REMAINED INTACT DURING THE TEST, MEASUREMENTS OF FUEL PELLETT-TO-CLADDING GAP, AS WELL AS THE SURFACE AREA OF THE FUEL CRACKS AT SEVERAL AXIAL LOCATIONS ARE PRESENTED. A TOTAL EFFECTIVE RADIAL GAP IS CALCULATED AND THE FUEL STRUCTURE AND POROSITY ARE ANALYZED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + REACTOR, BWR + FUEL ROD + IRRADIATION TESTING + URANIUM DIOXIDE + ZIRCALOY + HEAT CONDUCTANCE, FUEL TO CLAD + PDF (S-99) + EXAMINATION + NRC-3

132784

FARRAR LC + CROUCHER ON + ALLISON CH + FLOREN SA
IRRADIATION EFFECTS TEST SERIES TEST IE-3 TEST RESULTS REPORT
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TREE-NUMBER-1106 ++ 137 PPS, 18 TABS, 45 FIGS, REFS, OCT. 1977

TEST IE-3 USED FOUR, 0.97-M LONG PRESSURIZED WATER REACTOR TYPE FUEL RODS FABRICATED FROM PREVIOUSLY IRRADIATED FUEL. THE FUEL RODS WERE SUBJECTED TO A PRECONDITIONING PERIOD, FOLLOWED BY A POWER RAMP TO 60 KW/M. ONE ROD FAILED APPROXIMATELY 45 SECONDS AFTER THE REACTOR WAS SHUT DOWN AS A RESULT OF CLADDING EMBRITTLEMENT DUE TO EXTENSIVE CLADDING OXIDATION. TEST DATA ARE COMPARED WITH FRAP-T3 COMPUTER MODEL CALCULATIONS AND DATA FROM A PREVIOUS IRRADIATION EFFECTS TEST IN WHICH FOUR IRRADIATED FUEL RODS OF A SIMILAR DESIGN WERE TESTED. TEST IE-3 RESULTS INDICATED THAT THE IRRADIATED STATE OF THE FUEL RODS DID NOT SIGNIFICANTLY AFFECT FUEL ROD BEHAVIOR DURING NORMAL, ABNORMAL (POWER RAMP OF 20 KW/M PER MINUTE), AND ACCIDENT (FILM BOILING) CONDITIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

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129704 CONTINUED

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, FUR • FUEL ROD • PWR (S-RR) • EXAMINATION • FUEL BOWLING • FAILURE, CLADDING • THERMAL TRANSIENT • ENRICHMENT • ORIGATION • IRRADIATION TESTING

129857

ENGLAND TH • STANLEYLATOR HS
MULTIGROUP BETA AND GAMMA SPECTRA OF INDIVIDUAL ENDF/B-IV FISSION-PRODUCT NUCLIDES
LOS ALAMOS SCIENTIFIC LAB., N.M.
LA-4485-0632-MS • 106 PPS. 3 REFS. DEC. 1976

DELAYED BETA AND GAMMA GROUP-ENERGY SPECTRA WERE CALCULATED BY 100 INDIVIDUAL FISSION-PRODUCT NUCLIDES HAVING SPECTRAL DATA IN ENDF/B-IV FISSION-PRODUCT FILES. TO BETA SPECTRA, IN UNIFORM-GRID 70-GROUP STRUCTURE, AND THE GAMMA SPECTRA, IN UNIFORM-GRID 150-GROUP STRUCTURE BETWEEN 0 AND 7.5 MEV, ARE SUFFICIENTLY DETAILED FOR COLLAPSING TO ANY DESIRED PRACTICAL GROUP STRUCTURE. THESE SPECTRA HAVE BEEN USED IN DECAY-HEAT SUMMATION CALCULATIONS FOR COMPARISONS WITH CORRESPONDING EXPERIMENTAL BETA AND GAMMA SPECTRA.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

DATA COLLECTION • COMPUTER PROGRAM • DECAY HEAT • SPECTROMETRY, BETA • SPECTROMETRY, GAMMA • NRC-3 • RADIOISOTOPE

129940

ENGLAND TH • STANLEYLATOR HS
COMPARISONS OF CALCULATED AND EXPERIMENTAL DELAYED FISSION-PRODUCT BETA AND GAMMA SPECTRA FROM 235U THERMAL FISSION
LOS ALAMOS SCIENTIFIC LAB., N.M.
LA-4485-0690-MS • 64 PPS. 3 TABS. 102 FIGS. 11 REFS. JULY 1977

DELAYED FISSION-PRODUCT BETA AND GAMMA SPECTRA FOLLOWING SHORT AND LONG 235U THERMAL NEUTRON IRRADIATIONS (0.015 S, 1 S, 10 S, 100 S, 5.50 M, AND 8 H) WERE CALCULATED FOR A NUMBER OF COOLING PERIODS. THE RESULTS OF THESE CALCULATIONS, BASED ON ENDF/B-IV FISSION-PRODUCT DATA, ARE COMPARED WITH CORRESPONDING EXPERIMENTAL DATA AVAILABLE RECENTLY FROM SEVERAL RESEARCH ESTABLISHMENTS (LOS ALAMOS SCIENTIFIC LABORATORY, OAK RIDGE NATIONAL LABORATORY, AND THE UNIVERSITY OF ILLINOIS). THESE COMPARISONS SHOW GENERALLY GOOD AGREEMENT BETWEEN CALCULATIONS AND EXPERIMENTS AND INDICATE THE ADEQUACY OF ENDF/B-IV DATA FOR SUCH CALCULATIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • SOURCE, RADIATION • FISSION PRODUCT ACTIVITY, GROSS • BETA EMITTER • GAMMA EMITTER • CODES AND STANDARDS • DECAY HEAT • NRC-3 • JACOBS

129987

ENGLAND TH • STANLEYLATOR HS • SCHENTER HE • SCHWITZTROTH P
FISSION-PRODUCT SOURCE TERMS FOR REACTOR APPLICATIONS
LOS ALAMOS SCIENTIFIC LAB., N.M.
LA-4485-0617-MS • 27 PPS. AUG. 1977

CALCULATED AND MEASURED FISSION-PRODUCT SOURCE TERMS ARE PRESENTED. THE EMPHASIS IS ON DECAY HEATING FOLLOWING FISSION FOR USE IN THE DEVELOPMENT OF A NEW AND DECAY HEAT STANDARD. COMPARISONS WITH EXPERIMENT ARE GIVEN FOR BETA AND GAMMA SPECTRA, AND ENERGY-INTEGRATED VALUES. CALCULATED RESULTS FOR GASEOUS CONTENT, ABSORPTION BUILDUP IN REACTORS, AND PHOTONEUTRON SPECTRA ARE SUMMARIZED. UNCERTAINTY ANALYSES BASED ON NUCLEIDE DATA (ENDF/B-IV) AND ON COMBINED DECAY-HEAT EXPERIMENTS ARE PROVIDED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • SOURCE, RADIATION • FISSION PRODUCT ACTIVITY, GROSS • BETA EMITTER • GAMMA EMITTER • CODES AND STANDARDS • DECAY HEAT • NRC-3

132200

TARNELL AL • BENDT PJ
DECAY HEAT FROM PRODUCTS OF 235U THERMAL FISSION BY FAST-RESPONSE BOIL-OFF CALORIMETRY
LOS ALAMOS SCIENTIFIC LAB., N.M.
LA-4485-0713 • 37 PPS. SEPT. 1977

A CRYOGENIC BOIL-OFF CALORIMETER WAS USED TO MEASURE THE DECAY HEAT FROM THE PRODUCTS OF THERMAL-NEUTRON-INDUCED FISSION OF 235U. DATA ARE PRESENTED FOR COOLING TIMES BETWEEN 10 AND 101 (EXP 5) SEC FOLLOWING A 2 X 10¹² (EXP 4) SEC IRRADIATION AT CONSTANT THERMAL-NEUTRON FLUX. THE DATA AGREE WITHIN THE COMBINED UNCERTAINTIES WITH SUMMATION CALCULATIONS USING THE ENDF/B-IV DATA BASE. THE EXPERIMENTAL DATA WERE COMBINED WITH SUMMATION CALCULATIONS TO GIVE THE DECAY HEAT FOR INFINITE (10 EXP 13 SEC) IRRADIATION. FOR SHORT COOLING TIMES, THE RESULTS ARE ABOUT 7% BELOW THE CURRENT AMERICAN NUCLEAR SOCIETY (ANS) STANDARD. THE UNCERTAINTY IN THESE RESULTS IS SIGNIFICANTLY SMALLER THAN THAT ASSIGNED TO THE STANDARD.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • DECAY HEAT • MEASUREMENT • THERMAL NEUTRON • CRYOGENICS • CODES AND STANDARDS • COMPARISON • NRC-3 • JACOBS

126619

HELEMAN JT

PREP2: A PROGRAM TO AID IN THE PREPARATION OF INPUT DATA FOR THE FABOUT TWO-DIMENSIONAL DISCRETE ELEMENT CODE
 OAK RIDGE NATIONAL LAB., TENN.
 ORNL/CSO/TN-19 +. 89 PPS, 1 TAB, 7 FIGS, 4 REFS, MAY 1977

A COMPUTER CODE PREP2 WAS WRITTEN TO AID IN THE PREPARATION OF DATA SETS FOR THE FABOUT TWO-DIMENSIONAL DISCRETE ELEMENT CODES USED FOR THE ASSESSMENT OF ENVIRONMENTAL IMPACTS IN POWER PLANT SITES. THESE LATTER HYDRAULIC TRANSPORT CODES USE TWO TYPES OF DISCRETE ELEMENTS, A RECTANGULAR "INTERNAL" ELEMENT AND A MORE COMPLICATED "BOUNDARY" ELEMENT WHICH INCORPORATES THE IRREGULAR SHAPE OF THE REGION BEING SIMULATED. THE DATA SET IS PREPARED FROM A MAP OF THE AREA TO BE SIMULATED AND A DESCRIPTION OF THE DISCRETE ELEMENT STRUCTURE TO BE USED. THIS REPORT DESCRIBES THE COMPUTER PROGRAM, THE REQUIREMENTS FOR ITS USE, AND ILLUSTRATES ITS USE AT AN ACTUAL PLANT SITE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM + TRANSPORT + DISPERSION + RIVER + ESTUARY + DATA PROCESSING + ENVIRONMENT + HYDRAULIC ANALYSIS + NRC-3 + PLUME BEHAVIOR

129280

DICKENS JA + EMERY JF + FREESTONE RH + LOVE TA

FISSION PRODUCT BETA AND GAMMA ENERGY RELEASE QUARTERLY PROGRESS REPORT FOR JANUARY-MARCH 1977
 OAK RIDGE NATIONAL LAB., TENN.
 ORNL/NUREG/TN-121 +. 22 PPS, 2 TABS, 2 FIGS, 16 REFS, JULY 1977

FISSION CHAMBERS CONTAINING FOILS OF 200 MICRO CM TO 800 MICRO CM 235U WERE EXPOSED TO AN EXTERNAL BEAM OF THERMAL NEUTRONS FOR APPROXIMATELY 30 HOURS. COUNTING RATES AND PULSE-HEIGHT SPECTRA WERE OBSERVED. FOLLOWING THE END OF THE EXPOSURE THE CHAMBERS WERE MOVED TO THE LOW-BACKGROUND COUNTING AREA, AND THE RESULTING FISSION-PRODUCT GAMMA RAYS WERE COUNTED. K-FACTORS WERE DETERMINED. THESE DATA WERE THEN USED TO DETERMINE MSUD P) FOR SELECTED SAMPLES RUN EARLIER IN THE PROGRAM, WITH THE RESULTING MSUD P) BEING APPROXIMATELY 1.58 LOWER THAN PREVIOUSLY MEASURED. SIMILAR CHAMBERS WERE MADE CONTAINING 100 MICRO CM TO 200 MICRO CM FOILS OF 239 PU.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FISSION RECOIL + CHAMBER, FISSION + SPECTROMETRY, GAMMA + SPECTROMETRY, BETA + GAMMA EMITTER + BETA EMITTER + NRC-3

127849

HOBSON OO

QUARTERLY PROGRESS REPORT ON THE CREEPDOWN AND COLLAPSE OF ZIRCALOY FUEL CLADDING FOR JULY 1976-MARCH 1977
 OAK RIDGE NATIONAL LAB., TENN.
 ORNL/NUREG/TN-125 +. 25 PPS, 2 TABS, 16 FIGS, JULY 1977

DESIGN CONSIDERATIONS AND PROGRESS ON THE JOINT U.S./DUTCH IN-REACTOR CREEPDOWN EXPERIMENT ARE DISCUSSED. FINAL DEVELOPMENT OF THE EDDY-CURRENT SURFACE DISPLACEMENT MONITORING SYSTEM IS DESCRIBED AND THE RESULTS OF INITIAL CREEPDOWN TESTS ARE SHOWN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ZIRCALOY + CLADDING + CREEP + FAILURE + NRC-3

127862

CATHCART JV + PAVEL RE + HREE RA + DRUSCHEL RE + TUPER GJ + CAMPBELL JJ + JURY SH

ZIRCONIUM METAL-WATER OXIDATION KINETICS IV. REACTION RATE STUDIES
 OAK RIDGE NATIONAL LAB., TENN.
 ORNL/NUREG-17 +. 100 PPS, 55 TABS, 62 FIGS, 64 REFS, AUG. 1977

THE ISOTHERMAL RATES OF OXIDE, OXYGEN-STABILIZED ALPHA, AND XI LAYER GROWTH AND OF TOTAL OXYGEN CONSUMPTION OF ZIRCALOY-4 IN STEAM WERE DETERMINED FROM 900 TO 1500 DEG C (1652-2732 DEG C) AND DESCRIBED IN TERMS OF ANALYTICAL EXPRESSION FOR THE APPROPRIATE PARABOLIC RATE CONSTANTS. COMPARISON OF THE RESULTS WITH EXISTING RATE DATA DEMONSTRATES THE RELATIVE CONSERVATISM OF THE BAKER-JUST CORRELATION. EXPERIMENTAL TRANSIENT TEMPERATURE OXIDATION BEHAVIOR WAS FOUND TO BE PREDICTABLE BY COMPUTER CODES DESIGNED FOR THAT PURPOSE. AN ERROR ANALYSIS OF THE OXIDATION DATA IS PRESENTED IN BOTH STATISTICAL AND ANALYTICAL TERMS, AND POSSIBLE KINETIC EFFECTS OF HYDROGEN DISSOLVED IN OXIDIZED SAMPLES ARE CONSIDERED AND CONCLUDED TO BE NEGLIGIBLE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545 (88 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

ZIRCONIUM + METAL WATER REACTION + TEMPERATURE + COMPUTER PROGRAM + NRC-3 + JACOBS

127880

CATHCART JV

ZIRCONIUM METAL-WATER OXIDATION KINETICS PROGRAM SPONSORED BY THE NRC DIVISION OF REACTOR SAFETY RESEARCH FOR APRIL-JUNE 1977
 OAK RIDGE NATIONAL LAB., TENN.

PAGE -22

127000 (CONTINUED)

ORNL/NUMES-132 +. 37 PPS, 7 TABS, 10 FIGS, 10 REFS, AUG, 1977

SCALING TESTS OF THE EFFECT OF STEAM PRESSURE ON THE OXIDATION RATE OF ZIRCALOY-4 YIELDED NEGATIVE RESULTS AT 1500 DEG C (2012 DEG F) FOR STEAM AT 3.45 MPa (500 PSI); PROBLEMS WITH REPRODUCIBILITY WERE ENCOUNTERED AT 900 DEG C (1652 DEG F). ADDITIONAL EXPERIMENTS IN 0.9 MPa (1300 PSI) STEAM ARE IN PROGRESS. MEASUREMENTS OF THE OXIDATION RATE OF ZIRCALOY-4 SPECIMENS IN PURE OXYGEN LED TO THE CONCLUSION THAT HYDROGEN PREVIOUSLY REPORTED PRESENT IN OXIDIZED SPECIMENS HAD NO EFFECT ON MEASURED RATES OF OXIDATION IN STEAM. ADDITIONAL IN SITU TESTS IN THE MINIZUC APPARATUS CONFIRMED THE VALIDITY OF PREVIOUS ESTIMATES OF MAXIMUM TEMPERATURE MEASUREMENT UNCERTAINTIES.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545 100 CENTS/PAGE - MINIMUM CHARGE \$2.00

ZIRCONIUM + METAL WATER REACTION + OXIDATION + STEAM + TEMPERATURE + NRC-3 + JACOBS

129000

ORNL/NUMES-133

QUARTERLY PROGRESS REPORT ON FISSION PRODUCT BEHAVIOR IN LWRs FOR THE PERIOD APRIL-JUNE 1977
ORNL/NUMES-133 +. 23 PPS, 5 TABS, 6 FIGS, 6 REFS, SEPT, 1977

THE FISSION PRODUCT RELEASE EXPERIMENTS WERE CONDUCTED IN AN AIR ATMOSPHERE. ONE OF THESE, HIGH BURNUP FUEL TEST 5 (HBU-5), WAS CONDUCTED OVER A 20-HR PERIOD AT 500 DEG C. IN THIS EXPERIMENT, 0.3 X 10¹²SP -61% OF THE TOTAL CESIUM INVENTORY AND 91 X 10¹²SP -61% OF THE RUTHENIUM INVENTORY WERE RELEASED FROM THE ROD. THE SECOND TEST, HIGH BURNUP FUEL TEST 6 (HBU-6), WAS RUN OVER A 3-HR PERIOD AT 700 DEG C. THE EXPERIMENT YIELDED A CESIUM RELEASE AMOUNTING TO 14.3 X 10¹²SP -61% OF INVENTORY, WHEREAS THE RUTHENIUM RELEASE WAS 7.3 X 10¹²SP -61%. THE RUTHENIUM RELEASES ARE MARKEDLY GREATER THAN OBSERVED PREVIOUSLY IN CORRESPONDING TESTS IN STEAM. DESIGN OF THE FISSION PRODUCT TRANSPORT TEST FACILITY HAS PROGRESSED TO THE POINT AT WHICH ESTIMATES OF THE COSTS OF CONSTRUCTION CAN BE MADE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FISSION PRODUCT RELEASE + DESIGN CRITERIA + FUEL BURNUP + COST ANALYSIS + REACTOR, LAFOR + REACTOR, LWR + STEAM + RUTHENIUM + CESIUM + NRC-5 + JACOBS

131233

DICKENS JK + EMERY JP + LOVE TA + MCCONNELL JV

FISSION-PRODUCT ENERGY RELEASE FOR TIMES FOLLOWING THERMAL-NEUTRON FISSION OF ²³⁵U BETWEEN 2 AND 14000 SECONDS
ORNL/NUMES-10 +. 228 PPS, 13 TABS, 41 FIGS, 70 REFS, OCT, 1977

SAMPLES OF MASS 1 TO 10 MICRO GM WERE IRRADIATED FOR 1 TO 100 SEC USING THE FAST PNEUMATIC-TUBE FACILITY AT THE OAK RIDGE RESEARCH REACTOR. FOR THE GAMMA-RAY DATA THE SPECTRA WERE OBTAINED USING AN HE-110 DETECTOR WITH AN ANTICOINCIDENCE MANTLE. THE RAW DATA WERE UNFOLDED TO PROVIDE SPECTRAL DISTRIBUTIONS OF MODERST RESOLUTION. THESE WERE INTEGRATED OVER E(500 GAMMA) AND E(500 BETA) TO PROVIDE TOTAL YIELD AND ENERGY INTEGRALS AS A FUNCTION OF TIME AFTER FISSION. RESULTS ARE LOW COMPARED TO THE PRESENT 1973 AHS DECAY-HEAT STANDARD.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + CODES AND STANDARDS + DECAY HEAT + BETA RAY + GAMMA + FISSION PRODUCT ACTIVITY, CROSS + MEASUREMENT + NRC-3 + JACOBS

131182

DICKENS JK + EMERY JP + FREESTONE BN + LOVE TA

FISSION-PRODUCT BETA AND GAMMA ENERGY RELEASE QUARTERLY PROGRESS REPORT FOR APRIL-JUNE 1977
ORNL/NUMES-10 +. 11 PPS, 2 FIGS, 14 REFS, NOV, 1977

DATA HAVE BEEN OBTAINED FOR GAMMA-RAY ENERGY-RELEASE RATES FOR THERMAL-NEUTRON FISSION OF ²³⁵U. THESE DATA HAVE BEEN PARTLY REDUCED TO RELATIVE (I.E. UNNORMALIZED TO THE NUMBER OF FISSIONS) SPECTRA WHICH ARE VERY SIMILAR TO THOSE OBTAINED FOR THERMAL-NEUTRON FISSION OF ²³⁵U. STUDY OF THE HIGH-ENERGY ELECTRON RESPONSE OF THE BETA-RAY DETECTOR HAS CONTINUED, AND SHALL (< 15 TO 50) DISCREPANCIES IN THEORETICAL AND EXPERIMENTAL SPECTRAL DISTRIBUTIONS ARE BEING INVESTIGATED. ALTHOUGH THE SOURCE OF THESE DISCREPANCIES HAS NOT BEEN DETERMINED, AN ANALYSIS OF THE MAXIMUM EFFECT ON ²³⁵U TOTAL ENERGY-RELEASE RATES ALREADY REPORTED WAS MADE, AND THIS EFFECT IS < 1%.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + DECAY HEAT + PLUTONIUM + CODES AND STANDARDS + GAMMA + BETA RAY + NEUTRON + FISSION PRODUCT ACTIVITY, CROSS + NRC-3 + JACOBS

131000

CATHART JV

ZIRCONIUM METAL-WATER OXIDATION KINETICS PROGRAM SPONSORED BY THE NRC DIVISION OF REACTOR SAFETY RESEARCH FOR JULY-SEPTEMBER 1977
ORNL/NUMES-10 +. 11 PPS, 2 FIGS, 14 REFS, NOV, 1977

131945 *CONTINUED*
ORNL/NUREG/TM-159 +. 17 PPS, 2 TABS, 11 FIGS, 13 REFS, NOV, 1977

THIS REPORT IS THE LAST TO BE ISSUED UNDER THE ZIRCON PROGRAM. ALL MAJOR OBJECTIVES HAVE BEEN ACHIEVED. AN ANALYSIS OF THE RADIAL THERMAL GRADIENTS IN ZIRCONIUM OXIDATION SPECIMENS WAS UNDERTAKEN TO ASSESS ACCURACY OF TEMPERATURE MEASUREMENTS IN THIS STUDY. STEAM PRESSURE EFFECT STUDIES REVEALED AN INCREASING RATE OF OXIDE LAYER GROWTH AT 900 C (1652 F) BUT NO INFLUENCE OF PRESSURE AT 1100 C (2012 F). THE OXIDE THICKNESS INCREASE NOTED AT 900 C AMOUNTED TO ABOUT A FACTOR OF 2 RELATIVE TO RESULTS AT ATMOSPHERIC PRESSURE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ZIRCONIUM + OXIDATION + METAL WATER REACTION + TEMPERATURE GRADIENT + TESTING + JACOBS + NRC-3

131946
PAVEL BE + CATHEART JR + CAMPBELL JJ + ARV SH
ZIRCONIUM METAL-WATER OXIDATION KINETICS - V. OXIDATION OF ZIRCALOY IN HIGH PRESSURE STEAM
OAK RIDGE NATIONAL LAB., TENN.
ORNL/NUREG-31 +. 25 PPS, 5 TABS, 11 FIGS, 10 REFS, DEC, 1977

SUMMARIZES THE RESULTS OF THE HIGH-PRESSURE STEAM OXIDATION TESTS CONDUCTED AS PART OF THE ZIRCONIUM-METAL WATER OXIDATION KINETICS PROGRAM. THE PRIMARY PURPOSE OF THIS PROGRAM IS TO PROVIDE A RELIABLE SET OF KINETIC MEASUREMENTS TO DESCRIBE THE ISOTHERMAL REACTION OF ZIRCALOY-4 PUR TUBES WITH STEAM AT ELEVATED TEMPERATURES. THE GROWTH OF THE OXIDE AND ALPHA LAYERS DURING HIGH PRESSURE OXIDATION WAS COMPARED TO THAT OBTAINED EARLIER FOR OXIDATION IN THE GROWTH BEHAVIOR OF THE LAYERS WERE EVIDENT. THE DATA FOR HIGH PRESSURE OXIDATION AT 900 DEG C EXHIBITED CONSIDERABLE SCATTER, PARTICULARLY FOR OXIDE LAYER GROWTH.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + ZIRCONIUM + METAL WATER REACTION + OXIDATION + STEAM + REACTOR, PUR + TUBING + HIGH TEMPERATURE + NRC-3

131103
CHAPMAN BN
MULTIROD BURST TEST PROGRAM QUARTERLY PROGRESS REPORT FOR APRIL-JUNE 1977
OAK RIDGE NATIONAL LAB., TENN.
ORNL/NUREG/TM-135 +. 59 PPS, TABS, FIGS, REFS, DEC, 1977

CHECKOUT OF THE MULTIROD TEST FACILITY WAS COMPLETED, AND OPERATION AND TEST PROCEDURES WERE PREPARED. FABRICATION OF THE FIRST R & D TEST BUNDLE WAS NEARLY COMPLETED, USING THE BEST FUEL SIMULATORS (INTERNAL WATERS) AVAILABLE AT THE TIME. SINCE THIS WAS THE FIRST TIME SOME OF THE ASSEMBLY PROCEDURES WERE USED UNDER STRICT QUALITY CONTROL MEASURES, DIFFICULTIES WERE ENCOUNTERED; THESE ARE DISCUSSED IN SOME DETAIL. TWO ADDITIONAL SINGLE-ROD BURST TESTS (SR-70 AND SR-291) WERE CONDUCTED. THE FUEL PIN SIMULATORS USED IN THESE TESTS WERE FABRICATED AS PROTOTYPE TO VERIFY THE PROCEDURES USED IN FABRICATING THE SIMULATORS FOR THE FIRST BUNDLE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + ACCIDENT, LOSS OF COOLANT + REACTOR, LWR + SIMULATION + HEATERS + TESTING + BURST PRESSURE + NRC-3 + FAILURE, FUEL ELEMENT

131843
THOMAS KR + VAN DER HAAR T
DESIGN AND SAFETY REPORT FOR MIDDLE CAPSULES - A SERIES OF ZIRCALOY CREEPDOWN IRRADIATION EXPERIMENTS
OAK RIDGE NATIONAL LAB., TENN.
ORNL/NUREG/TM-167 +. 78 PPS, 7 TABS, 20 FIGS, 14 REFS, DEC, 1977

THE PURPOSE IS TO PROVIDE DATA ON DEFORMATION CHARACTERISTICS OF ZIRCALOY TUBES, TYPICAL OF LWR FUEL ELEMENTS, UNDER CONDITIONS OF AXIAL AND TANGENTIAL COMPRESSIVE STRESS. DATA WILL BE USED TO VERIFY AND IMPROVE MATERIAL BEHAVIOR CODES. THE EXPERIMENTS WERE DESIGNED AND WILL BE FABRICATED BY ORNL AND WILL BE IRRADIATED IN THE HIGH FLUX REACTOR (HFR) AT PETTEN. THIS REPORT IS TO SERVE AS A BASIS OF TECHNICAL AND SAFETY ASSESSMENT LEADING TO APPROVAL FOR INSTALLATION AND OPERATION IN THE HFR.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ZIRCALOY + FUEL ELEMENTS + CREEP + IRRADIATION TESTING + DEFORMATION

131942
SPINRAD B1
EVALUATION OF FISSION PRODUCT AFTERHEAT QUARTERLY REPORT APRIL 1-JUNE 30, 1977
OREGON STATE UNIV., CORVALLIS
NUREG-0018-7 +. 37 PPS, 10 FIGS, NOV, 1977

THE GOAL OF THE WORK IS TO IMPROVE THE UNDERSTANDING OF SHUTDOWN POWER IN REACTORS DUE TO RADIOACTIVE DECAY OF FISSION PRODUCTS, PARTICULARLY IN LIGHT WATER REACTORS IN THE PERIOD 0-1000 SECONDS AFTER SHUTDOWN. THE POWER PRODUCED DURING THIS PERIOD IS A PRIMARY DETERMINANT OF CORE SHUTDOWN-COOLING REQUIREMENTS. MUCH OF THE WORK DONE DURING THIS QUARTER WAS IN PREPARATION OF

131042 *CONTINUED*
THE FINAL REPORT FOR THE PROJECT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
R AND D PROGRAM + SAFETY PROGRAM + REACTOR, LWR + DECAY HEAT + COMPARISON, THEORY AND EXPERIENCE + FISSION
PRODUCT ACTIVITY, CROSS + NRC-3 + JACOBS

129463
BAILLONEN BA
LIGHT WATER REACTOR SAFETY RESEARCH PROGRAM QUARTERLY REPORT OCTOBER - DECEMBER 1976
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-77-0904 + NUREG-0193-2 +. 94 PPS, 10 TABS, 32 FIGS, 13 REFS, JULY 1977

ONE SMALL-SCALE STEEL-CONCRETE INTERACTION TEST WAS CONDUCTED. THE RESULTS PAVE THE WAY FOR
CONDUCTING HIGHLY INSTRUMENTED TESTS OF MELT-CONCRETE INTERACTIONS AT PROTOTYPIC TEMPERATURES TO
QUANTIFY THE PHENOMENA. A PRELIMINARY COMPREHENSIVE MODEL OF THE MELT-CONCRETE INTERACTION HAS
BEEN DEVELOPED. SEVERAL IMPROVEMENTS TO THE STEAM EXPLOSION TRIGGERING EXPERIMENT APPARATUS HAVE
GREATLY INCREASED THE EXPERIMENTAL CAPABILITY. THESE IMPROVEMENTS INCLUDE THE INSTALLATION OF A
GREATER CAPACITY POWER SUPPLY TO ALLOW LARGER MELT FRACTIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
REACTOR, LWR + SAFETY PROGRAM + FUEL MELTDOWN + PRESSURE PULSE + OUT OF PILE EXPERIMENT + CONCRETE +
EXPLOSION + STEEL + CHEMICAL REACTION + STEAM + CORE CATCHER + CORE MELTDOWN + NRC-3 + JACOBS + R AND D
PROGRAM

131036
GUTH KL
3-MW PLASMAJET FACILITY THERMAL TESTS OF CONCRETE
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-77-0902 +. 130 PPS, 8 FIGS, 4 REFS, JULY 1977

A TEST WAS MADE TO OBTAIN EXPERIMENTAL DATA RELATIVE TO THE THERMAL RESPONSE OF CONCRETE TO
INCIDENT HEAT FLUX. CONCRETE CYLINDERS WERE POSITIONED IN A SUPERSONIC FLOW OF HEATED NITROGEN
FROM AN ARC HEATER. THE END OF THE CONCRETE CYLINDERS IMPACTED BY THE FLOW WERE SUBJECTED TO
HEAT FLUXES IN THE RANGE OF 0.13 TO 0.35 MW/CM² SQ. MEASUREMENTS INCLUDED COLD WALL SURFACE HEAT
FLUX AND PRESSURE DISTRIBUTIONS, SURFACE AND INDEPTH TEMPERATURES, ABLATION RATES, AND SURFACE
EMISSION SPECTROGRAMS. A DESCRIPTION OF THE TESTS AND A TABULATION OF TEST DATA ARE INCLUDED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
TESTING + CONCRETE + CONTAINMENT + THERMAL CONSIDERATION + HEAT TRANSFER EXPERIMENT + CORE MELTDOWN + R AND D
PROGRAM + REACTOR, LWR + NRC-3

120496
MURFIN UB
A PRELIMINARY MODEL FOR CORE/CONCRETE INTERACTIONS
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-77-0370 +. 45 PPS, 17 FIGS, 15 REFS, AUG. 1977

A PRELIMINARY MODEL IS DESCRIBED FOR COMPUTING THE RATE OF PENETRATION OF CONCRETE BY A MOLTEN LWR
CORE. AMONG THE PHENOMENA INCLUDED ARE CONVECTIVE STIRRING OF THE MELT BY EVOLVED GASES,
ADMIXTURE OF CONCRETE DECOMPOSITION PRODUCTS TO THE MELT, CHEMICAL REACTIONS, RADIATIVE HEAT
LOSS, AND VARIATION OF HEAT TRANSFER COEFFICIENTS WITH LOCAL PRESSURE. THE MODEL IS MOST
APPLICABLE TO A TWO-PHASE MELT (METALLIC PLUS OXIDIC) HAVING A FAIRLY HIGH METALLIC CONTENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
ANALYTICAL MODEL + CORE + CONCRETE + CORE MELTDOWN + REACTOR, LWR + JACOBS + NRC-3 + NRC-7

132009
LIGHT WATER REACTOR SAFETY RESEARCH PROGRAM QUARTERLY REPORT JANUARY - MARCH 1977
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-77-1249 + NUREG-0183-3 +. 66 PPS, REFS, OCT. 1977

DURING THIS QUARTER THE PRIMARY EMPHASIS WAS ON DATA ANALYSIS AND INTERPRETATION PLUS MODEL
DEVELOPMENT. A PRELIMINARY MODEL OF THE MELT/CONCRETE INTERACTION WAS DISTRIBUTED TO SEVERAL
USERS. ANALYSIS OF THE LARGE-SCALE STEEL/CONCRETE TEST RESULTS INDICATES THAT THE TOTAL EROSION
INCREASES WITH EACH ADDITIONAL TEST IN THE SAME CRUCIBLE. THIS INDICATES THAT THERMAL PRE-
CONDITIONING OF THE CONCRETE COULD INCREASE EROSION RATES. PRELIMINARY ANALYSIS ALSO INDICATES
THAT AT LEAST 30 PERCENT OF THE HEAT FROM THE MELT GOES INTO THE CONCRETE. THEORETICAL GAS
ANALYSIS TECHNIQUES INDICATE THAT THE C/H RATIO OF THE GAS RELEASED FROM THE CONCRETE CHANGES
WITH TIME DURING THE EXPERIMENT. THE ANALYSIS INDICATES THE H₂O CONTENT OF THE GAS INCREASES
WITH TEMPERATURE. SUBSTANTIAL AMOUNTS OF H WILL BE PRESENT AT TEMPERATURES ABOVE 2000 C, AND C
MAY PRECIPITATE FROM THE GAS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

132200 *CONTINUED*
R AND D PROGRAM * SAFETY PROGRAM * REACTOR. LV2 * FUEL MELTDOWN * PRESSURE PULSE * OUT OF PILE EXPERIMENT *
CONCRETE * EXPLOSION * STEEL * CHEMICAL REACTION * STEAM * CORE CATCHER * CORE MELTDOWN * NRC-3 * JACOBS

4. 00 - WATER REACTOR SAFETY RESEARCH, ANALYSIS DEVELOPMENT

120177

LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM: QUARTERLY PROGRESS REPORT JANUARY-MARCH 1977
ARGONNE NATIONAL LAB., ILL.
ANL-77-24 +. 120 PPS, TABS, FIGS, REFS, JUNE 1977

THIS PROGRESS REPORT SUMMARIZES THE ARGONNE NATIONAL LABORATORY WORK PERFORMED DURING JANUARY, FEBRUARY, AND MARCH 1977 ON WATER-REACTOR-REACTOR SAFETY PROBLEMS. THE FOLLOWING RESEARCH AND DEVELOPMENT AREAS ARE COVERED: (1) LOSS-OF-COOLANT ACCIDENT RESEARCH: HEAT TRANSFER AND FLUID DYNAMICS; (2) TRANSIENT FUEL RESPONSE AND FISSION-PRODUCT RELEASE PROGRAM; (3) MECHANICAL PROPERTIES OF ZIRCALOY CONTAINING OXYGEN; AND (4) STEAM-EXPLOSION STUDIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, LWR + R AND D PROGRAM + ANL + SAFETY ANALYSIS + ACCIDENT, LOSS OF COOLANT + HEAT TRANSFER ANALYSIS + FLOW STABILITY + FISSION PRODUCT RELEASE + PROPERTY, MECHANICAL + ZIRCALOY + CLADDING + STEAM + EXPLOSION + NRC-3 + NRC-4

120223

LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM: QUARTERLY PROGRESS REPORT, APRIL-JUNE 1977
ARGONNE NATIONAL LAB., ILL.
ANL-77-50 +. 120 PPS, TAB, FIGS, REFS, DEC. 1977

THIS PROGRESS REPORT SUMMARIZES THE ARGONNE NATIONAL LABORATORY WORK PERFORMED ON WATER-REACTOR-REACTOR SAFETY PROBLEMS. THE FOLLOWING RESEARCH AND DEVELOPMENT AREAS ARE COVERED: (1) LOSS-OF-COOLANT ACCIDENT RESEARCH: HEAT TRANSFER AND FLUID DYNAMICS; (2) TRANSIENT FUEL RESPONSE AND FISSION-PRODUCT RELEASE PROGRAM; (3) MECHANICAL PROPERTIES OF ZIRCALOY CONTAINING OXYGEN; AND (4) STEAM-EXPLOSION STUDIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, LWR + R AND D PROGRAM + ANL + SAFETY ANALYSIS + ACCIDENT, LOSS OF COOLANT + HEAT TRANSFER ANALYSIS + FLOW STABILITY + FISSION PRODUCT RELEASE + PROPERTY, MECHANICAL + ZIRCALOY + CLADDING + STEAM + EXPLOSION + NRC-3 + NRC-4

120501

GIESERKE JA + JORDAN M + BAYBUTT P + WOOTEN RO + DENNING RS
FISSION PRODUCT TRANSPORT ANALYSIS - QUARTERLY PROGRESS REPORT FOR APRIL THROUGH JUNE 1977
BATTELLE COLUMBUS LABS., OHIO
BRL-MURE-1901 +. 17 PPS, SEPT. 30, 1977

THIS PROGRESS REPORT COVERS CONTINUING ACTIVITIES ASSOCIATED WITH MODELING THE TRANSPORT AND DEPOSITION OF FISSION PRODUCTS WITHIN PWR AND BWR PRIMARY SYSTEMS DURING THE REFLOOD TIME PERIOD FOLLOWING A TERMINATED LOCA. THE ORIGINAL SCOPE OF THE OVERALL PROJECT HAS BEEN EXPANDED TO INCLUDE CONSIDERATION OF CONDITIONS CONSISTENT WITH THOSE LEADING TO POSTULATED CORE MELTDOWN SITUATIONS. THUS, MAJOR EFFORTS ARE TO BE CONCENTRATED WITH THE ANALYSIS OF POSTULATED MELTDOWN CONDITIONS AND THE EXTENSION AND REVISION OF THE TRAP CODES FOR APPLICABILITY TO THESE CONDITIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS + FISSION PRODUCT TRANSPORT + ANALYTICAL MODEL + CORE REFLOODING + CORE MELTDOWN + COMPUTER PROGRAM + DEPOSITION + NRC-3 + NRC-4

120433

GIESERKE JA + BAYBUTT P + JORDAN M + DENNING RS
FISSION PRODUCT TRANSPORT ANALYSIS: QUARTERLY PROGRESS REPORT FOR JULY THROUGH SEPTEMBER, 1977
BATTELLE COLUMBUS LABS., OHIO
BRL-MURE-1900 +. 27 PPS, 4 TABS, 5 FIGS, 12 REFS, NOV. 30, 1977

TECHNICAL PROGRESS DURING THIS QUARTER CONSISTED OF COMPLETING AND CHECKING THE MODEL FOR EMERGENCY CORE COOLING SYSTEM (ECCS) WATER SCRUBBING OF FISSION PRODUCTS FOR INCLUSION IN THE TRAP CODE, IDENTIFYING FISSION PRODUCT SOURCE TERMS DATA AND INFORMATION FOR INITIAL USE UNDER ASSUMED MELTDOWN CONDITIONS, SPECIFYING THE THERMAL-HYDRAULIC CONDITIONS TO BE CONSIDERED FOR FISSION PRODUCT TRANSPORT IN POSTULATED MELTDOWN ACCIDENTS, AND CHOOSING A CONTROL VOLUME FRAMEWORK FOR INITIAL USE IN DEVELOPMENT OF THE TRANSPORT MODEL FOR MELTDOWN CONDITIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + FISSION PRODUCT TRANSPORT + ANALYTICAL MODEL + CORE REFLOODING + CORE MELTDOWN + COMPUTER PROGRAM + DEPOSITION + NRC-3 + NRC-4

127900

STEWART CH + WHEELER CL + CORA RJ
CORA-IV: THE MODEL AND THE METHOD
BATTELLE PACIFIC NORTHWEST LABS., RICHLAND, WASH.
BNWL-2210 +. 105 PPS, 11 TABS, 50 FIGS, REFS, JULY 1977

IN PREVIOUS CORA PROGRAMS THERE WAS AN INABILITY TO CONSIDER FLOW REVERSAL AND RECIRCULATION AND THE LACK OF A PRESSURE BOUNDARY CONDITION. THE ACHIEVEMENT OF THESE CAPABILITIES WAS THE MAJOR

127626 *CONTINUED*

EMPHASIS IN THE COBRA-IV DEVELOPMENT EFFORT. COBRA-IV CAPABILITIES HAVE BEEN EXTENDED BY IMPROVEMENTS MADE TO THE COBRA-IIIIC FLUID ENERGY EQUATION AND TO THE TRANSIENT FUEL CONDUCTION MODEL. THE FUEL ROD CONDUCTION MODEL WAS MADE MORE EFFICIENT THROUGH THE USE OF AN ORTHOGONAL COLLOCATION METHOD FOR RADIAL HEAT CONDUCTION. COBRA IV IS FULLY 4-DIMENSIONAL AND THE CONTINUING DEVELOPMENT EFFORT WILL BROADEN THE CODE'S APPLICABILITY EVEN FURTHER.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM + FLOW, TWO PHASE + FLOW, RECIRCULATION + FUEL ROD + BLOWDOWN + HEAT TRANSFER, CONVECTION + ACCIDENT, LOSS OF COOLANT + THERMAL HYDRAULIC ANALYSIS + FLOW BLOCKAGE + NRC-4

130057

LEVINE RM
DEVELOPMENT OF A COMPUTER CODE FOR THERMAL HYDRAULICS OF REACTORS (THOR) - EIGHTH QUARTERLY PROGRESS REPORT - JULY-SEPTEMBER 1976
BROOKHAVEN NATIONAL LAB., UPTON, N.Y.
BNL-NUREG-30659 +, 39 PPS, MAY 1977

THE PURPOSE OF THE ADVANCED CODE DEVELOPMENT WORK REPORTED HERE IS TO CONSTRUCT A COMPUTER CODE FOR THE PREDICTION OF THERMODYNAMIC TRANSIENTS IN WATER-COOLED NUCLEAR REACTOR SYSTEMS. THE FUNDAMENTAL FORMULATION OF FLUID DYNAMICS IS TO BE BASED ON THE ONE-DIMENSIONAL DRIFT FLUX MODEL FOR NON-HOMOGENEOUS, NON-EQUILIBRIUM FLOWS OF TWO-PHASE MIXTURES. PARTICULAR EMPHASIS IS PLACED ON COMPONENT MODELING, AUTOMATIC PREDICTION OF INITIAL STEADY STATE CONDITIONS, INCLUSION OF ONE-DIMENSIONAL TRANSIENT NEUTRON KINETICS, FREEDOM IN THE SELECTION OF COMPUTED SPATIAL DETAIL, DEVELOPMENT OF RELIABLE CONSTITUTIVE DESCRIPTIONS, AND MODULAR CODE STRUCTURE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

THERMAL HYDRAULIC ANALYSIS + COMPUTER PROGRAM + REACTOR, PWR + REACTOR, BWR + FLOW + ANALYTICAL MODEL + REACTOR TRANSIENT + REACTOR KINETICS + JACOBS + NRC-2 + NRC-4

129475

ROMANO AJ
REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT, JANUARY 1-MARCH 31, 1977
BROOKHAVEN NATIONAL LAB., UPTON, N.Y.
BNL-NUREG-30661 +, 210 PPS, MAY 1977

THE PROJECTS REPORTED EACH QUARTER ARE THE FOLLOWING: GAS REACTOR SAFETY EVALUATION, THOR CODE DEVELOPMENT, SSC CODE DEVELOPMENT, LUFOR AND LUR SAFETY EXPERIMENTS, FAST REACTOR SAFETY CODE VALIDATION, TECHNICAL COORDINATION OF STRUCTURAL INTEGRITY, AND FAST REACTOR SAFETY RELIABILITY ASSESSMENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, LUFOR + REACTOR, HTGR + REACTOR, LUR + COMPUTER PROGRAM, DIGITAL + CODES AND STANDARDS + ACCIDENT, CORE DISRUPTIVE + JACOBS + SAFETY ANALYSIS + ACCIDENT, CONSEQUENCES + NRC-4 + NRC-7 + NRC-8

129106

ROMANO AJ
REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT APRIL 1-JUNE 30, 1977
BROOKHAVEN NATIONAL LAB., UPTON, N.Y.
BNL-NUREG-30663 +, 193 PPS, TABS, FIGS, JULY 1977

THE REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT DESCRIBES CURRENT ACTIVITIES AND TECHNICAL PROGRESS IN THE PROGRAMS AT BROOKHAVEN NATIONAL LABORATORY SPONSORED BY THE USNRC REACTOR SAFETY RESEARCH DIVISION. THE PROJECTS REPORTED EACH QUARTER ARE THE FOLLOWING: GAS REACTOR SAFETY EVALUATION, THOR CODE DEVELOPMENT, SSC CODE DEVELOPMENT, LUFOR AND LUR SAFETY EXPERIMENTS, FAST REACTOR SAFETY CODE VALIDATION, TECHNICAL COORDINATION OF STRUCTURAL INTEGRITY, AND FAST REACTOR SAFETY RELIABILITY ASSESSMENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, LUFOR + REACTOR, LUR + REACTOR, HTGR + SAFETY ANALYSIS + COMPUTER PROGRAM, DIGITAL + CODES AND STANDARDS + ACCIDENT, CORE DISRUPTIVE + CORE MELTDOWN + ACCIDENT, LOSS OF COOLANT + BLOWDOWN + ACCIDENT ANALYSIS + BOILING + NRC-4 + NRC-7 + NRC-8

132096

CATTON E + CHAN CH + DHIR VL + LIU CY
SUPPRESSION POOL DYNAMICS QUARTERLY PROGRESS REPORT JULY 1, 1976-SEPTEMBER 30, 1976
UNIV. OF CALIF., LOS ANGELES
NUREG-0264-1 +, 89 PPS, 7 TABS, 20 FIGS, 4 REFS, MAY 1977

TRANSIENT AIR-WATER TESTS WERE PERFORMED AS PART OF THE LABORATORY SCALE EXPERIMENTS PROJECT TO INVESTIGATE HYDRAULIC RESPONSE IN A BWR PWR SUPPRESSION CONTAINMENT AT THE ONSET OF A LOCA. THESE TESTS WERE PERFORMED IN A CYLINDRICAL PEXIGLASS TEST CHAMBER. AIR WAS INJECTED DOWNWARD THROUGH PIPES OF DIFFERENT DIAMETERS PLACED IN THE MIDDLE OF THE TEST CHAMBER CONTAINING WATER AT ROOM TEMPERATURE. TESTS WERE PERFORMED TO INVESTIGATE THE EFFECTS OF SUBMERGENCE DEPTH AND DRY WELL PRESSURE ON THE VENT CLEARING PHENOMENA. DATA WERE ALSO OBTAINED FOR THE POOL SWELL HEIGHT

120000 *CONTINUED*

UNDER THE INFLUENCE OF A STEADY SUBMERGED AIR FLOW STREAM, ALTHOUGH IT IS DIRECTLY NOT APPLICABLE TO THE BWR LOCA CONDITION.

AVAILABILITY - HANAGARD, INC., P.O. BOX 3504, DAN RIDGE, TENN. 37030

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, BWR • EXPERIMENT • BUBBLE • HYDRODYNAMIC ANALYSIS • NRC-4 • CONTAINMENT, PRESSURE SUPPRESSION

120000

CATTON I • CHAN CK • SHIR K • CHIU HH • LEE CH
SUPPRESSION POOL DYNAMICS - QUARTERLY PROGRESS REPORT APRIL 1-JUNE 30, 1977
UNIV. OF CALIF., LOS ANGELES
NURS-0004-4 • 52 PPS, 40 FIGS, DEC. 1977

THE RESULTS OF TRANSIENT AIR-WATER TESTS TO INVESTIGATE BASIC HYDRODYNAMICS OF PHENOMENA THAT CAN BE EXPERIENCED IN A BWR PRESSURE SUPPRESSION POOL AT THE ONSET OF A LOSS-OF-COOLANT ACCIDENT ARE REPORTED. TESTS WITH DIFFERENT GASES (AIR, HELIUM, AND ARGON) SHOWED THAT THE VELOCITY OF BUBBLE AND THE DENSITY OF THE GAS AFFECTED THE COMPRESSION OF THE GAS MEDIUM BETWEEN THE WATER SLUG AND THE BLENDING VALVE DURING VENT CLEARING. THE BUBBLE SHAPE IS CONTROLLED BY THE SPREAD BOUNDARY CONDITION. THE INITIAL BUBBLE GROWTH IS DRIVEN BY THE INTERNAL PRESSURE EXPANSION, AUGMENTED LATER BY THE MASS INFLUX.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, BWR • CONTAINMENT, PRESSURE SUPPRESSION • EXPERIMENT • BUBBLE • HYDRODYNAMIC ANALYSIS • HELIUM • AIR • ARGON • NRC-4

120007

CATTON I • SHIR VK • CHAN CK • LEE CH
SUPPRESSION POOL DYNAMICS - QUARTERLY PROGRESS REPORT OCTOBER 1, 1976 - DECEMBER 31, 1976
UNIV. OF CALIF., LOS ANGELES
NURS-0004-2 • 65 PPS, 3 TABS, 33 FIGS, MAY 1977

TRANSIENT AIR-WATER TESTS TO INVESTIGATE BASIC HYDRODYNAMICAL PHENOMENA THAT CAN BE EXPERIENCED IN A BWR PRESSURE SUPPRESSION POOL AT THE ONSET OF A LOSS OF COOLANT ACCIDENT ARE REPORTED. THE TESTS WERE PERFORMED IN A CYLINDRICAL PLEXIGLASS TEST CHAMBER. AIR WAS INJECTED DOWNWARD THROUGH PIPES OF DIFFERENT DIAMETERS PLACED IN THE MIDDLE OF THE TEST CHAMBER CONTAINING WATER AT ROOM TEMPERATURE. THE INFLOW OF THE AIR IS OBSERVED TO FORM A DEPRESSION ON THE LIQUID SURFACE. AS THE LIQUID IS ACCELERATED AND PUSHED INTO THE WATER POOL, THE TAYLOR INSTABILITY BREAKS UP THE INTERFACE AND A WATER SPIKE IS SEEN TO GROW INTO THE GASEOUS REGION, BUT IT IS BROKEN UP PRIOR TO COMPLETE VENT CLEARING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, BWR • EXPERIMENT • BUBBLE • HYDRODYNAMIC ANALYSIS • AIR • NRC-4 • CONTAINMENT, PRESSURE SUPPRESSION

120103

CARD ON
QUARTERLY TECHNICAL PROGRESS REPORT ON BATER REACTOR SAFETY PROGRAM SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH (JAN-MAR 1977)
IDAND NATIONAL ENGINEERING LAB., IDAND FALLS
TRER-NURS-1120 • 130 PPS, 17 TABS, 75 FIGS, REFS, JUNE 1977

LIGHT WATER REACTOR SAFETY RESEARCH PERFORMED JANUARY THROUGH MARCH 1977 IS DISCUSSED, INCLUDING THE FOLLOWING: A TEST SERIES TO EVALUATE THE EFFECTIVENESS OF DIFFERENT EMERGENCY CORE COOLANT INJECTION ARRANGEMENTS WAS COMPLETED. A LOFT TEST SERIES WAS INITIATED TO INVESTIGATE THE EFFECT OF BREAK NOZZLE GEOMETRY ON BREAK FLOW AND SYSTEM RESPONSE DURING A 2000 DOUBLE-BENDED COLO LEG BREAK TEST. THE ANALYSIS OF RESULTS OF LOFT NONNUCLEAR EXPERIMENTS WERE MADE. A REVIEW OF FLECHT AND SENISCALE FORCED FLOW AND GRAVITY FEED REFLOOD HEAT TRANSFER EXPERIMENTS WAS PERFORMED. A MORE GENERAL VERSION OF THE RELAPS PILOT CODE (VERSION 2) WAS PROGRAMMED. EXPERIMENT RESULTS FROM TESTS CONDUCTED AT THE MARViken CONTAINMENT TEST FACILITY IN SWEDEN AND THE BATTLE TEST FACILITY IN FRANKFURT, GERMANY, WERE USED FOR VERIFICATION OF COMPUTER CONTAINMENT CODES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, LWR • SAFETY ANALYSIS • OUT OF PILE EXPERIMENT • COMPUTER PROGRAM, DIGITAL • COMPARISON, THEORY AND EXPERIENCE • EMERGENCY COOLING SYSTEM • BLOWDOWN • ACCIDENT, LOSS OF COOLANT • NRC-2 • NRC-3 • NRC-4

120000

REARIN JA • GERHA GA • BOWEN BP
PRAP-02: A COMPUTER CODE FOR THE STEADY STATE ANALYSIS OF OXIDE FUEL RODS
IDAND NATIONAL ENGINEERING LAB., IDAND FALLS
TRER-NURS-1107 • 200 PPS, 6 TABS, 50 FIGS, 60 REFS, JULY 1977

PRAP-02 SOLVES THE LONG TERM BURNUP RESPONSE OF A LIGHT WATER REACTOR FUEL ROD. THE COUPLED EFFECTS OF FUEL AND CLADDING DEFORMATION, TEMPERATURE DISTRIBUTION, INTERNAL GAS ACCUMULATION,

120600 *CONTINUED*

AND PRESSURE AND MATERIAL PROPERTY ON THE BEHAVIOR OF THE FUEL ROD ARE CONSIDERED. FISSION GAS GENERATION AND RELEASE ARE CALCULATED AS A FUNCTION OF BURNUP. THE CLADDING DEFORMATION MODEL INCLUDES MULTIAXIAL ELASTO-PLASTIC ANALYSIS AND CONSIDERS CREEP. FRAP-52 IS A MODULAR CODE WITH EACH MAJOR COMPUTATIONAL SUBCODE ISOLATED WITHIN THE CODE AND COUPLED TO THE MAIN CODE BY SUBROUTINE CALLS AND DATA TRANSFER THROUGH ARGUMENT LISTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

COMPUTER PROGRAM - FUEL ROD - FUEL BURNUP - DEFORMATION - PRESSURE, INTERNAL - FISSION GAS RELEASE - CLADDING - CRACK - REACTOR, LWR - DISTRIBUTION - TEMPERATURE - NRC-4

120560

QUARTERLY TECHNICAL PROGRESS REPORT ON WATER REACTOR SAFETY PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH APRIL-JUNE 1977
IDAND NATIONAL ENGINEERING LAB., IDAND FALLS
TREE-MUREG-1147 +. 92 PPS, AUG. 1977

RESULTS FROM THE PREVIOUSLY CONDUCTED SEMISCALE ROD-1 ECC INJECTION TEST SERIES WERE ANALYZED. TESTING IN THE LOFT COUNTERPART TEST SERIES WAS ESSENTIALLY COMPLETED, AND THE STEAM GENERATOR TUBE RUPTURE TEST SERIES HAS BEGUN. TWO TESTS IN THE ALTERNATE ECC INJECTION TEST SERIES WERE CONDUCTED WHICH INCLUDED INJECTION OF EMERGENCY CORE COOLANT INTO THE UPPER PLENUM THROUGH USE OF THE LOW PRESSURE INJECTION SYSTEM. THE LOSS-OF-FLUID TEST PROGRAM SUCCESSFULLY COMPLETED NONNUCLEAR LOSS-OF-COOLANT EXPERIMENT LI-6. A NUCLEAR TEST, CC 2-3, IN THE POWER BURST FACILITY REACTOR WAS PERFORMED TO EVALUATE THE POWER OSCILLATION METHOD OF DETERMINING GAP CONDUCTANCE TO DETERMINE THE EFFECTS OF INITIAL GAP SIZE, FILL GAS COMPOSITION, AND FUEL DENSITY ON THE THERMAL PERFORMANCE OF A LIGHT WATER REACTOR FUEL ROD.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

OR AND D PROGRAM - SAFETY PROGRAM - REACTOR, LWR - ACCIDENT, LOSS OF COOLANT - EMERGENCY COOLING SYSTEM - HEAT FLUX, CRITICAL - DNB - THERMOCOUPLE - HEAT CONDUCTANCE, FUEL TO LLAC - COMPUTER PROGRAM - CREEP BEHAVIOR - THERMAL HYDRAULIC ANALYSIS - COMPARISON, THEORY AND EXPERIENCE - NRC-2 - NRC-3 - NRC-4 - JACOBS

127613

STANFORD'S MC - ENGLAND TN
PPDCVS AND PPSPEC: COMPUTER PROGRAMS FOR CALCULATING FISSION-PRODUCT BETA AND GAMMA MULTIGROUP SPECTRA FROM ENDF/B-IV DATA
LOS ALAMOS SCIENTIFIC LAB., N.M.
LA-MUREG-6810-MS +. 82 PPS, 3 FIGS, 12 REFS, MAY 1977

PPDCVS PROCESSES ENDF/B-IV FISSION-PRODUCT DECAY ENERGY DATA TO GENERATE MULTIGROUP BETA AND GAMMA SPECTRA FROM INDIVIDUAL ENDF/B-IV FISSION-PRODUCT NUCLIDES. PPSPEC FURTHER USES THESE SPECTRA AND THE CORRESPONDING NUCLIDE ACTIVITIES CALCULATED BY THE FINDER-10 CODE TO PRODUCE CUMULATIVE BETA AND GAMMA SPECTRA IN THE SAME ENERGY GRIDS IN WHICH PPDCVS GENERATES INDIVIDUAL ISOTOPE DECAY SPECTRA.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

BETA EMITTER - BETA RAY - GAMMA EMITTER - GAMMA - FISSION PRODUCT ACTIVITY, CROSS - ENERGY SPECTRUM - COMPUTER PROGRAM - NRC-4 - NRC-1

129672

JACKSON JP
NUCLEAR REACTOR SAFETY QUARTERLY PROGRESS REPORT JANUARY 1-MARCH 31, 1977
LOS ALAMOS SCIENTIFIC LAB., N.M.
LA-MUREG-6842-PH +. 109 PPS, 27 TABS, 92 FIGS, 67 REFS, JUNE 1977

A NUMBER OF IMPORTANT ADVANCES WERE MADE AS PART OF THE LWR SAFETY RESEARCH PROGRAM. THE TRAC CODE WAS IMPROVED BY ADDING ALL THE REMAINING MODULES NEEDED TO ANALYZE A LARGE BREAK LOCA IN A PWR. SIMRA-1 WAS APPLIED TO THE ANALYSIS OF SEVERAL PERTINENT LWR SAFETY PROBLEMS. THESE INCLUDED TWO-DIMENSIONAL SODIUM VOIDING, EXTENSION OF THE CROP LOSS-OF-FLUID ACCIDENT INITIATION PHASE, AND WORK ENERGY PARTITION FOLLOWING DISASSEMBLY. IN THE HTGR SAFETY PROGRAM, THE REMAINING EQUIPMENT NEEDED FOR THE FUEL HEAT-UP EXPERIMENT HAS BEEN SHIPPED. SEVERAL EXPERIMENTS WERE PERFORMED AS PART OF THE GCPR CORE DISRUPTIVE TEST PROGRAM. IN THE CONTAINMENT EVALUATION AREA, A NEW CAPABILITY IN THE COMPARE CODE THAT ALLOWED TREATMENT OF ICE-CONDENSED CONTAINMENT DESIGN WAS VERIFIED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

REACTOR, SAFETY RESEARCH - REACTOR, HTGR - REACTOR, LWR - REACTOR, LWR - ACCIDENT ANALYSIS - SAFETY ANALYSIS - REACTOR PHYSICS - ANALYTICAL MODEL - OTHERMAL HYDRAULIC ANALYSIS - JACOBS - NRC-4 - NRC-7 - NRC-8 - REACTOR KINETICS - REACTOR, GCPR - CONTAINMENT, ICE CONDENSER

130570

JACKSON JP - STEVENSON MC
NUCLEAR REACTOR SAFETY QUARTERLY PROGRESS REPORT: APRIL 1-JUNE 30, 1977
LOS ALAMOS SCIENTIFIC LAB., N.M.
LA-MUREG-6834 +. 105 PPS, 17 TABS, 70 FIGS, 80 REFS, AUG. 1977

120076 *CONTINUED*

RESEARCH PERFORMED AT LBL IS SUMMARIZED. FOUR AREAS ARE PRESENTED, INCLUDING HTGR, LWR, LWRB, AND REACTOR TECHNOLOGY SAFETY RESEARCH. OTHER AREAS DISCUSSED ARE FISSION PRODUCT TECHNOLOGY, PRIMARY COOLANT IMPURITIES, STRUCTURAL EVALUATION, I AND C SYSTEM, ACCIDENT DELINEATION, AND PHENOMENA MODELING. OTHER RESEARCH AREAS INCLUDE EVALUATION OF TRAC CODE, THERMAL-HYDRAULICS, AND CONTAINMENT EVALUATION. OTHER RESEARCH AREAS INCLUDE EVALUATION OF PROGRAMS INVOLVING SIMMER CODE DEVELOPMENT, CORE DISRUPTIVE ACCIDENT ANALYSIS, SAFETY TEST FACILITY STUDIES, AND OUT-OF-PILE MODELING AND EXPERIMENTS. REACTOR TECHNOLOGY RESEARCH AREAS INCLUDE PROGRAMS CONCERNING THERMAL AND ACCIDENT ANALYSIS, REACTIVITY MEASUREMENTS, CRITICALITY INCIDENTS STUDY, AND WUTH STUDY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

REACTOR, SAFETY RESEARCH • REACTOR, HTGR • REACTOR, LWRB • REACTOR, LWR • ACCIDENT ANALYSIS • SAFETY ANALYSIS • REACTOR PHYSICS • ANALYTICAL MODEL • THERMAL HYDRAULIC ANALYSIS • JACOBS • NRC-6 • NRC-7 • NRC-8 • REACTOR KINETICS

120730

TAGGART RA • LILES ON
POSTS: A POSTPROCESSOR COMPUTER CODE FOR PRODUCING THREE-DIMENSIONAL MOVIES OF TWO-PHASE FLOW IN A REACTOR VESSEL.
LOS ALAMOS SCIENTIFIC LAB., N.M.
LA-MURDER-6004-MS • 77 PPS, SEPT. 1977

THE DEVELOPMENT OF THE TRAC COMPUTER CODE FOR ANALYSIS OF LOCAS IN LIGHT-WATER REACTORS INVOLVES THE USE OF A THREE-DIMENSIONAL, TWO-FLUID HYDRODYNAMICS MODEL TO DESCRIBE THE TWO-PHASE FLOW OF STEAM AND WATER THROUGH THE REACTOR VESSEL. ONE OF THE MAJOR PROBLEMS INVOLVED IN INTERPRETING RESULTS FROM THIS CODE IS THE PRESENTATION OF THREE-DIMENSIONAL FLOW PATTERNS. THIS CODE (POST) IS USED AS A POSTPROCESSOR IN CONJUNCTION WITH A STAND ALONE THREE-DIMENSIONAL TWO-PHASE HYDRODYNAMICS CODE (CYLTP) WHICH IS A TEST BED FOR THE THREE-DIMENSIONAL ALGORITHMS TO BE USED IN TRAC.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

FLOW, TWO PHASE • COMPUTER PROGRAM • JACOBS • REACTOR, LWR • NRC-6 • ACCIDENT, LOSS OF COOLANT

127614

POWLER PG • BRVSON JV
USERS MANUAL FOR THE CORTES GRAPHICS PACKAGE GORPAC
OAK RIDGE NATIONAL LAB., TENN.
ORNL/TN-127 • 37 PPS, FIGS, 3 REFS, AUG. 1977

CORTES IS A SERIES OF FIVE FINITE-ELEMENT COMPUTER PROGRAMS DESIGNED SPECIFICALLY FOR THE ANALYSIS OF PIPING TEJ JOINTS AND SINGLE NOZZLES IN PIPE AND CYLINDRICAL VESSELS SUBJECTED TO A VARIETY OF MECHANICAL AND THERMAL LOADINGS. THIS REPORT PROVIDES NECESSARY USER INFORMATION TO IMPLEMENT AND USE A GRAPHICS PACKAGE FOR THE CORTES COMPUTER PROGRAMS. COMPLETE INPUT INSTRUCTIONS ARE PROVIDED AND SAMPLE INPUT AND OUTPUT ARE GIVEN. CORTES PROGRAMS ARE AVAILABLE UPON REQUEST FROM THE ARGONNE CODE CENTER, ARGONNE NATIONAL LABORATORY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

PIPES AND PIPE FITTINGS • NOZZLE • COMPUTER PROGRAM • THERMAL MECHANICAL EFFECT • STRESS ANALYSIS • DEFORMATION • THERMAL TRANSIENT • STRESS • NRC-6

2. RC - WATER REACTOR SAFETY RESEARCH, METALLURGY AND MATERIALS

123250

HAHN GT • GELLEN PC • HOGANLAND RC
CRITICAL EXPERIMENTS, MEASUREMENTS AND ANALYSES TO ESTABLISH A CRACK ARREST METHODOLOGY FOR NUCLEAR PRESSURE VESSEL STEELS. REPORT JULY THROUGH SEPTEMBER 1976
BATTELLE COLUMBUS LABS., OHIO
BRI-MUREG-1700 • 100 PPS. 8 TABS. 40 FIGS. FEB. 1977

THE PROGRAM IS IMPLEMENTING RECOMMENDATIONS OF A PWR/MPC WORKING GROUP ON CRACK ARREST AND INCLUDES WORK ON DYNAMIC FRACTURE MECHANICS ANALYSIS, MEASUREMENTS OF CRACK ARREST IN A VARIETY OF SYSTEMS USING COMMON EXPERIMENTAL MATERIALS, AND PHOTOELASTIC STUDIES OF FAST FRACTURE AND ARREST. THE INITIAL PHASE, WHICH IS CONCERNED MAINLY WITH THE VALIDATION OF A CRACK ARREST THEORY, HAS BEEN COMPLETED. IN ADDITION TO THE THEORETICAL CALCULATIONS, A CRACK ARREST EXPERIMENT ON SHALLOW GROOVED SEN TEST SPECIMENT OF A533B STEEL ARE DESCRIBED. ALSO, THE FEASIBILITY OF USING ELECTRON BEAM WELDED "COMPOUND" SPECIMENS TO REDUCE IRRADIATED MATERIAL REQUIREMENTS FOR FRACTURE TOUGHNESS AND CRACK ARREST TESTS IS EVALUATED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

R AND D PROGRAM • CRACK • PRESSURE VESSELS • STEEL • STRESS ANALYSIS • NRC-5 • JACOBS

127000

GELLEN PC • KANNINEN WP • HAHN GT • HARSCHALL CH
CRITICAL EXPERIMENTS, MEASUREMENTS, AND ANALYSES TO ESTABLISH A CRACK ARREST METHODOLOGY FOR NUCLEAR PRESSURE VESSEL STEEL NINTH QUARTERLY PROGRESS REPORT OCTOBER-DECEMBER 1976
BATTELLE COLUMBUS LABS., OHIO
MUREG-0200 • BRI-MUREG-1976 • 66 PPS. TABS. FIGS. HEPS. JULY 1977

THIS PROJECT IS A COORDINATED EFFORT DESIGNED TO ESTABLISH A NATIONAL CRACK ARREST METHODOLOGY FOR NUCLEAR PRESSURE VESSELS. THE PROGRAM INVOLVES BATTELLE'S COLUMBUS LABORATORIES, THE UNIVERSITY OF WASHINGTON, MATERIALS RESEARCH LABORATORY, AND THE UNIVERSITY OF ILLINOIS. THE PROGRAM IS IMPLEMENTING RECOMMENDATIONS OF A PWR/MPC WORKING GROUP ON CRACK ARREST AND INCLUDES WORK ON DYNAMIC FRACTURE MECHANICS ANALYSIS, MEASUREMENTS OF CRACK ARREST IN A VARIETY OF SYSTEMS USING COMMON EXPERIMENTAL MATERIALS, AND PHOTOELASTIC STUDIES OF FAST FRACTURE AND ARREST.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

STEEL • PRESSURE VESSELS • CRACK • ANALYTICAL TECHNIQUE • PHOTOELASTICITY • NRC-5

120000

RODABAUGH EC • HAREY WA • EIDER RJ
REVIEW AND ASSESSMENT OF RESEARCH RELEVANT TO DESIGN ASPECTS OF NUCLEAR POWER PLANT PIPING SYSTEMS
BATTELLE COLUMBUS LABS., OHIO
MUREG-0307 • BRI-MUREG-1976 • 671 PPS. 36 TABS. 23 FIGS. JULY 1977

THIS REPORT EVALUATES SIGNIFICANT RESEARCH ON PIPING SYSTEMS AND CORRELATES THAT RESEARCH WITH DESIGN PRACTICES. THE OBJECTIVE IS TO QUANTIFY THE RESEARCH/DESIGN PRACTICES IN TERMS OF THE RELIABILITY OF PIPING USED IN NUCLEAR POWER PLANTS. THIS REPORT DESCRIBES DESIGN PRACTICES RELATIVE TO ASME CODES; REVIEWS OPERATING EXPERIENCE OF PIPING SYSTEMS FOCUSING ON FAILURES; DISCUSSES RESEARCH ON PIPING WITH AND WITHOUT CONTROLLED FLAWS; ASSESSES THE RELIABILITY OF PIPING SYSTEMS; AND REVIEWS AREAS WHERE ADDITIONAL RESEARCH IS NECESSARY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

PIPES AND PIPE FITTINGS • DESIGN CRITERIA • R AND D PROGRAM • CODES AND STANDARDS • OPERATING EXPERIENCE SUMMARY • FAILURE • PIPE • FLAW • RELIABILITY ANALYSIS • NRC-5 • JACOBS

132207

RYSCAL EP • RODABAUGH EC • GROOM JJ • SCHMUESTER OH
RESIDUAL STRESSES AT GIRTH-BUTT WELDS IN PIPES AND PRESSURE VESSELS - FINAL REPORT APRIL 1, 1976 - JUNE 30, 1977
BATTELLE COLUMBUS LABS., OHIO
MUREG-0376 • 150 PPS. 9 TABS. NOV. 1977

A RESIDUAL STRESS MODEL FOR GIRTH-BUTT WELDS IN PRESSURE VESSELS AND PIPES WAS DEVELOPED AND VERIFIED FOR WELDS RANGING FROM 2 TO 30 PASSES. THE MODEL ALSO ACCURATELY PREDICTS RESIDUAL DEFORMATIONS. RESULTS INDICATE THAT THE MODEL CAN BE EXTENDED TO REPRESENT WELD REPAIRS IN PRESSURE VESSELS. IN ADDITION, PRELIMINARY RESULTS DIRECTED AT DEVELOPING A SIMPLIFIED MODEL OF 6140 BUTT WELDS SHOW GOOD AGREEMENT WITH DATA FOR ONE AND TWO-PASS WELDS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

R AND D PROGRAM • SAFETY PROGRAM • WELDS • WELDING • PRESSURE VESSELS • PIPES AND PIPE FITTINGS • STRESS ANALYSIS • MODEL • NRC-5 • JACOBS

131901

ALZWEINER JM • CLARK WA • ANDERSON WE • CROWE JC • VAGINS R
STEAM GENERATOR TUBE INTEGRITY PROGRAM QUARTERLY REPORT JANUARY 1-MARCH 31, 1977
BATTELLE PACIFIC NORTHWEST LABS., RICHMOND, WASH.

120941 *CONTINUED*
 NUREG-0300 + ENCL-2310 +. 130 PPS. TABS. FIGS. OCT. 1977

A DRAFT OF THE DETAILED PROGRAM PLAN IN THE FORM OF AN ANALYSIS BEFORE TEST DOCUMENT HAS BEEN PREPARED. A DRAFT OF THE LITERATURE SURVEY OF OUR STEAM GENERATOR PROBLEMS HAS BEEN COMPLETED. A QUALITY ASSURANCE PLAN UNDER HAS BEEN COMPLETED FOR THE STEAM GENERATOR TUBE INTEGRITY PROGRAM. THE SINGLE FREQUENCY Eddy CURRENT INSPECTION SYSTEM HAS BEEN RECEIVED, CHECKED OUT AND CALIBRATED. BURST, COLLAPSE AND LEAK RATE FACILITIES HAVE BEEN DESIGNED AND CONSTRUCTION WORK HAS BEEN STARTED. THE CORROSION RESEARCH ACTIVITIES TO CHEMICALLY INDUCE TUBE DEFECTS ARE IN PROGRESS. THE TENSILE TESTER FOR MEASURING TUBE HOOP STRENGTH HAS BEEN DESIGNED, PARTLY AND TESTED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + STEAM GENERATOR + TUBING + INTEGRITY + TESTING + REACTOR, PWR + PLAS + CORROSION + TEST, NONDESTRUCTIVE + NRC-S + JACOBS

120943
 NUREG-PH + SCHWENK ED
 PROGRAM TO DEVELOP ACOUSTIC EMISSION-PLAN RELATIONSHIP FOR INSERVICE MONITORING OF NUCLEAR PRESSURE VESSELS -
 PROGRESS REPORT FEBRUARY 1-JULY 1, 1977
 BATTLE PACIFIC NORTHWEST LAB., RICHLAND, WASH.
 NUREG-0300-2 + ENCL-2232-2 +. 64 PPS. FIGS. NOV. 1977

THE PURPOSE OF THE PROGRAM IS TO BUILD AN EXPERIMENTAL EVALUATION OF THE FEASIBILITY OF DETECTING AND ANALYZING PLAS GROWTH IN REACTOR PRESSURE BOUNDARIES BY CONTINUOUSLY MONITORING FOR ACOUSTIC EMISSION (AE). AN AE MONITORING SYSTEM INCORPORATING A NEW CONCEPT WITH DATA RECORDING IN SOLID-STATE DIGITAL MEMORIES AND SIGNAL SOURCE ISOLATION HAS BEEN ADAPTED FOR USE ON THIS PROGRAM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + PRESSURE VESSELS + ACOUSTICS + TEST, NONDESTRUCTIVE + PLAS + NRC-S + JACOBS
 + INSERVICE INSPECTION

120939
 ZIMA BE
 A CORROSION CRITIQUE OF THE 2 1/4 CR-1 NO STEEL FOR LWR STEAM GENERATION SYSTEM APPLICATIONS
 BATTLE PACIFIC NORTHWEST LAB., RICHLAND, WASH.
 NUREG-0307 +. 73 PPS. 15 FIGS. 30 PAGES. 65 REFS. NOV. 1977

THE POSSIBILITY OF CRUCIAL ASSIGNMENTS FOR THE 2 1/4 CR-1NG STEEL WITHIN LWR STEAM GENERATION SYSTEMS HAS BEEN ANTICIPATED FOR OVER A DECADE. DESPITE THIS KNOWLEDGE, FROM THE PERSPECTIVE OF GENERAL CORROSION CREDENTIALS PERTINENT TO THIS SERVICE, THE EVIDENCE IN SUPPORT OF THIS STEEL REMAINS INADEQUATE IN SEVERAL IMPORTANT RESPECTS. THE EFFECT OF PERIODIC CHEMICAL CLEANING ON THE OVERALL WATER/SYSTEM OXIDATION BEHAVIOR OF THIS STEEL IS UNKNOWN. GENERAL METAL ABRASION COULD IMPOSE SEVERE DESIGN PENALTIES ON THIS STEEL WHEN SERVICE TIMES IN EXCESS OF 20 YEARS ARE INVOLVED. IT HAS EXHIBITED BOTH SEVERE PITTING AND CRACKING UNDER CHLORIDE AND CAUSTIC ENVIRONMENTS. BRITTLENESS, PROBABLY RELATED TO SOME DIRECT, OR INDIRECT, FORM OF CORROSION-INDUCED HYDROGEN ACTION, HAS BEEN OBSERVED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + REACTOR, LWR + STEEL + CORROSION + CRACK + CHLORIDE + STEAM GENERATOR + TUBING + BRITTLENESS + OXIDATION + NRC-S + JACOBS

120910
 REICH R + ESTERMAN EP + ERDOGAN F
 CONCEPTS FOR BENCHMARK PROBLEM DEVELOPMENT FOR FRACTURE MECHANICS APPLICATION IN SAFETY EVALUATION OF NUCLEAR PIPING IN SUBCRITICAL SERVICE
 BROOKHAVEN NATIONAL LAB., UPTON, N.Y.
 ENCL-NUREG-50076 +. 34 PPS. JUNE 1977

THE DISCUSSION OF BENCHMARK PROBLEMS PRESENTED IN THIS REPORT IS PART OF A REVIEW AND EVALUATION PROGRAM AIMED AT THE ASSESSMENT OF THE DEVELOPMENT PROBLEM AND ADAPTATION OF FRACTURE MECHANICS METHODS FOR LIFE PREDICTION AND SAFETY ANALYSIS OF NUCLEAR PIPING SYSTEMS. SEVEN PROBLEM AREAS WERE SINGLE OUT FOR STUDY: (1) APPLICATION TO DOUBLY CURVED (SHELL) STRUCTURES, (2) PART-THROUGH CRACK PROBLEMS, (3) LEAK-BEFORE-BREAK ANALYSIS, (4) TRIAXIAL CONSTRAINT EFFECT, (5) FATIGUE-CRACK PROPAGATION, (6) FRACTURE AND LIMIT LOAD, ENLARGED STRAIN GENERAL YIELD CONDITIONS, (7) TIME-DEPENDENT NONLINEAR CREEP BEHAVIOR.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + FRACTURE TOUGHNESS + PIPES AND PIPE FITTINGS + CREEP BEHAVIOR + FATIGUE + CRACK + PREDICTION + NRC-S

120907
 REACTOR PRIMARY COOLANT SYSTEM PIPE RUPTURE STUDY PROGRESS REPORT NO. 30 - JANUARY-MARCH 1977
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 GEAP-10007-30 +. 37 PPS. 7 TABS. 20 FIGS. MAY 1977

129467 *CONTINUED*

THE PIPE BURSTURE STUDY IS DESIGNED TO EXTEND THE UNDERSTANDING OF FAILURE-CAUSING MECHANISMS AND TO PROVIDE IMPROVED CAPABILITY FOR EVALUATING REACTOR PIPING SYSTEMS TO MINIMIZE THE PROBABILITY OF FAILURES. FOLLOWING A DETAILED REVIEW TO DETERMINE THE EFFORT MOST NEEDED TO IMPROVE NUCLEAR SYSTEM PIPING (PHASE I) ANALYTICAL AND EXPERIMENTAL EFFORTS (PHASE II) WAS STARTED IN 1965. THIS PROGRESS REPORT SUMMARIZES THE RECENT ACCOMPLISHMENTS OF A BROAD PROGRAM IN (1) BASIC FATIGUE CRACK GROWTH RATE STUDIES FOCUSED ON LOW PRIMARY PIPING MATERIALS IN A SIMULATED BOR PRIMARY COOLANT ENVIRONMENT, AND (2) STUDIES DIRECTED AT QUANTIFYING WELD SENSITIZATION IN TYPE-304 STAINLESS STEEL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

JACOBS + PIPES AND PIPE FITTINGS + FAILURE + REACTOR COOLANT + FATIGUE + CRACK + GROWTH/DEVELOPMENT + WELDS + SIMULATION + REACTOR, BOR + STEEL, STAINLESS + NRC-5

130775

IRWIN GR + DALLY JV + KOBAYASHI T + FOURNEY BL + ETHERIDGE JR
PHOTOLASTIC STUDIES OF CRACK PROPAGATION AND CRACK ARREST
UNIV. OF MARYLAND, COLLEGE PARK
NUREG-0342 +, 281 PPS, OCT. 1977

DESCRIBES THE THIRD YEAR EFFORT ON RESEARCH PROGRAMS DEALING WITH THE CHARACTERIZATION OF DYNAMIC ASPECTS OF FRACTURE. THE RESULTS INCLUDED IN THIS REPORT ARE: (1) VERIFICATION OF THE DCL ONE-DIMENSIONAL COMPUTER CODE, (2) DETERMINATION OF A - K RELATIONSHIP FROM MODIFIED COMPACT, (3) VERIFICATION OF THE DCL PROCEDURE FOR RESUB 1A) MEASUREMENT WITH MACHINE-LOADED C-DCB SPECIMEN, (4) FRACTURE BEHAVIOR IN DUBLER SPECIMENS, (5) CRACK PROPAGATION IN A THERMALLY STRESSED RING SPECIMEN, (6) DEVELOPMENT OF A TWO-DIMENSIONAL CODE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

JACOBS + PHOTOELASTICITY + CRACK + K AND D PROGRAM + COMPUTER PROGRAM + FRACTURE TOUGHNESS + NRC-5

129392

FREEDERICK JR + FAIRCHILD RC + ANDERSON BN
IMPROVED ULTRASONIC NONDESTRUCTIVE TESTING OF PRESSURE VESSELS ANNUAL PROGRESS REPORT, AUGUST 1, 1975-JULY 31, 1976
UNIV. OF MICHIGAN, ANN ARBOR
NUREG-0007-2 +, 133 PPS, 41 FIGS, SEPT. 1977

A SYNTHETIC APERTURE FOCUSING TECHNIQUE FOR ULTRASONIC TESTING (SAFT UT) IS DESCRIBED. THE TECHNIQUE EMPLOYS A SINGLE SCANNED TRANSDUCER OPERATING IN PULSE-ECHO MODE WITH DIGITAL DATA ACQUISITION AND SYNTHETIC APERTURE POST-PROCESSING TO PROVIDE HIGH LATERAL AND LONGITUDINAL RESOLUTION. THE EXTENSION OF PREVIOUSLY DEVELOPED ALGORITHMS TO PROVIDE VOLUMETRIC PROCESSING AND DISPLAY IS DESCRIBED. THE DESIGN OF A REFRESHED GREY-SCALE DISPLAY TO PROVIDE INTERACTIVE DISPLAY OF SAFT UT DATA IS DESCRIBED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

JACOBS + TEST, NONDESTRUCTIVE + ULTRASONICS + PRESSURE VESSELS + INSTRUMENT, DIGITAL + NRC-5

127402

LOSS PJ
STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS PROGRESS REPORT ENDING 28 FEBRUARY 1977
NAVAL RESEARCH LAB., WASHINGTON, D.C.
NRL/NUREG-RENO-REPORT-3512 +, 53 PPS, 6 TABS, 19 FIGS, 16 REFS, MAY 1977

DESCRIBES RESEARCH PROGRESS IN A CONTINUING PROGRAM TO CHARACTERIZE MATERIALS PROPERTIES PERFORMANCE WITH RESPECT TO STRUCTURAL INTEGRITY OF LIGHT WATER REACTOR PRESSURE BOUNDARY COMPONENTS. PROGRESS FOR THIS REPORTING PERIOD IS SUMMARIZED IN THE FOLLOWING AREAS: (1) RADIATION EMBRITTLEMENT RESISTANCE OF A533-B CLASS 1 STEEL PLATE AS A FUNCTION OF EXTRA LOW COPPER IMPURITY CONTACT, (2) INFLUENCE OF PHOSPHORUS AND COPPER CONTENT ON POST-IRRADIATION LOWER SHELF TOUGHNESS AND POST-IRRADIATION TOUGHNESS RECOVERY FOR A533-B STEEL PLATE AND WPLD METAL, (3) INVESTIGATIONS OF VARIOUS PRESTRESS AND PLASTIC NECK LIGAMENT PHENOMENA AS A MEANS TO MITIGATE CRACK EXTENSION IN A VESSEL DURING A LOSS OF COOLANT ACCIDENT, AND (4) EVALUATION OF CRITICAL FACTORS IN CRACK GROWTH RATE STUDIES IN A PRESSURIZED WATER REACTOR ENVIRONMENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

B AND D PROGRAM + SAFETY PROGRAM + REACTOR, LW + PRESSURE VESSELS + STEEL + IMPACT SHOCK + NOT DATA + IRRADIATION TESTING + GROWTH/DEVELOPMENT + CRACK + FATIGUE + NRC-5 + JACOBS

129396

LOSS PJ
STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS - PROGRESS REPORT ENDING 31 MAY 1977
NAVAL RESEARCH LAB., WASHINGTON, D.C.
NRL/NUREG-RENO-REPORT-3600 +, 45 PPS, 5 TABS, 13 FIGS, 27 REFS, SEPT. 1977

THIS REPORT DESCRIBES RESEARCH PROGRESS IN A CONTINUING PROGRAM TO CHARACTERIZE MATERIALS PROPERTIES PERFORMANCE WITH RESPECT TO STRUCTURAL INTEGRITY OF LIGHT WATER REACTOR PRESSURE

189386 ACFTINUED

SUMMARY COMMENTS. PROGRESS FOR THIS REPORT PERIOD IS SUMMARIZED IN THE FOLLOWING AREAS: (A) EVALUATION OF CRITICAL FACTORS IN FATIGUE CRACK GROWTH RATE STUDIES IN A PRESSURIZED WATER REACTOR ENVIRONMENT; (B) FRACTOGRAPHIC OBSERVATIONS OF FATIGUE CRACK PROPAGATION; (C) THE EARLY PROGRAM AND ITS OBJECTIVES; (D) THE EFFECT OF SUBMERGED ARC WELDING ON THE NOTCH TOUGHNESS OF LOW UPPER SHELF A533-B PLATE EXHIBITING A LOW UPPER SHELF AND ITS WELD MET, AND (F) SURVEY OF FOREIGN RESEARCH RELATED TO THE SAFETY OF LIGHT WATER REACTORS IN JAPAN, UNITED KINGDOM, FRANCE, AND GERMANY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS + FATIGUE + CRACK + PRESSURE VESSELS + STEEL + IRRADIATION TESTING + EMBRITTLEMENT + FRACTURE TOUGHNESS + ULTRASONIC MICROSCOPY + NRC-1 + NRC-8

181148

LOSS P1 + GRAY HA + MATHURNE JR
SIGNIFICANCE OF WARM PRESTRESS TO CRACK INITIATION DURING THERMAL SHOCK
NAVAL RESEARCH LAB., WASHINGTON, D.C.
NRL-REPORT-8148 +. 22 PPS, 2 TABS, 12 FIGS, 11 REFS, SEPT. 1977

DURING A LOSS OF COOLANT ACCIDENT (LOCA) FOLLOWED BY OPERATION OF THE EMERGENCY CORE COOLING SYSTEM, THE INSIDE WALL OF A NUCLEAR PRESSURE VESSEL IS SUBJECTED TO HIGH THERMAL STRESSES THAT CAN CAUSE EXTENSION OF A PREEXISTING FLAW. THE CRACK-TIP STRESS INTENSITY FACTOR (K_{ISUB}) MAY ACHIEVE ITS MAXIMUM VALUE EARLY IN THE TRANSIENT, BUT THE CRITICAL LEVEL FOR CRACK INITIATION (K_{ISUB} IC) MAY NOT BE REACHED UNTIL MINUTES LATER, WHEN THE LOADING HAS DECREASED FROM ITS PEAK. THIS PHENOMENON, TERMED WARM PRESTRESS (WPS), CAN PRECLUDE CRACK EXTENSION WHEN K_{ISUB} IS EQUALS OR EXCEEDS K_{ISUB} IC. THE EXPERIMENTS SHOW THAT THE PHENOMENON OF WARM PRESTRESS SHOULD BE CONSIDERED IN PREDICTIONS OF REACTOR VESSEL INTEGRITY, AND THAT THIS PHENOMENON MAY FORM A KEY ELEMENT UPON WHICH TO BASE ASSURANCE OF VESSEL INTEGRITY DURING A LOCA.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ACCIDENT, LOSS OF COOLANT + PRESSURE VESSELS + THERMAL MECHANICAL EFFECT + CRACK + STRUCTURAL INTEGRITY + NRC-5 + JACOBS

182320

MATHURNE JR + KOZLO JJ + BYRNE ST
EVALUATION OF COMMERCIAL PRODUCTION A533-B STEEL PLATES AND WELD DEPOSITS WITH EXTRA-LOW COPPER CONTENT FOR RADIATION RESISTANCE
NAVAL RESEARCH LAB., WASHINGTON, D.C. + COMBUSTION ENGINEERING INC.
NRL-REPORT-8136 +. 30 PPS, 9 TABS, 25 FIGS, 10 REFS, OCT. 1977

EXPLORED TRENDS IN RADIATION EFFECTS FOR COMMERCIALLY PRODUCED ASTM A533-B STEEL PLATES, WELD DEPOSITS, AND WELD HEAT-AFFECTED ZONES, EXPECTING THREE LEVELS OF CONTROL OVER IMPURITY COPPER CONTENT. THE PRIMARY OBJECTIVE WAS TO ESTABLISH THE TREND OF IMPROVED 288 C (550 F) RADIATION RESISTANCE WITH PROGRESSIVE REDUCTIONS IN ALLOWABLE COPPER CONTENT. THE STUDY HAS CONFIRMED THAT NEW ASTM AND AISI SUPPLEMENTAL SPECIFICATIONS ON COPPER CONTENT FOR NUCLEAR SERVICE ARE SUFFICIENTLY RESTRICTIVE TO OPTIMIZE 288 C (550 F) RADIATION RESISTANCE OF A533 PLATES AND WELD DEPOSITS FOR MOST PROJECTED VESSEL FLUENCE CONDITIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND B PROGRAM + SAFETY PROGRAM + PRESSURE VESSELS + STEEL + WELDS + RADIATION EFFECT + COPPER + IMPURITY + NRC-5 + JACOBS

127081

CAMERICO DA
SIGNIFICANCE OF REHEAT CRACKS TO THE INTEGRITY OF PRESSURE VESSELS FOR LIGHT-WATER REACTORS
OAK RIDGE NATIONAL LAB., TENN.
ORNL-TR-55-15 +. 31 PPS, 1 TAB, 20 FIGS, JULY 1977

REHEAT CRACKS ARE USUALLY MACROSCOPIC DEFECTS, CENTIMETERS LONG AND DEEP, AND ARE DETECTABLE BY USUAL NONDESTRUCTIVE EXAMINATION (NDE) OR AS MICROSCOPIC GRAIN BOUNDARY DECOMPOSITIONS (GBD) THAT ARE BEYOND THE LIMIT OF DETECTION BY COMMERCIAL NDE. THIS REPORT CONCENTRATED ON THE SIGNIFICANCE OF THE MICROSCOPIC CRACKS THAT MAY GO UNDETECTED. THE PROBABILITY THAT GBD EXIST IN THE HEAT-AFFECTED ZONES (HAZ) OF WELDMENTS OF PRESSURE VESSEL STEELS IS HIGH; PARTICULARLY IN SA 500 CLASS 2 WELDMENTS. ON THE BASIS OF THE METALLURGICAL INFORMATION AVAILABLE FOR MICROSTRUCTURES SIMILAR TO THOSE OF THE COARSE-GRAINED REGION OF THE HAZ AS WELL AS ON THE BASIS OF EXPERIMENTAL DATA (ADVERSELY SPARSE), IT CAN BE ASSUMED THAT THE GBD THAT EXIST IN THESE STEELS ARE NOT DETRIMENTAL TO THE INTEGRITY OF THE NUCLEAR PRESSURE VESSELS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CRACK + PRESSURE VESSELS + REACTOR, LWR + WELDS + TEST, NONDESTRUCTIVE + NRC-5 + JACOBS

189943

NEWMAN PA
STRUCTURAL INTEGRITY OF MATERIALS IN NUCLEAR SERVICE - A BIBLIOGRAPHY
OAK RIDGE NATIONAL LAB., TENN.

125943 *CONTINUED*

ORNL/NUREG/NSIC-100 P. 265 PPS. JULY 1977

THIS REPORT CONTAINS 670 ABSTRACTS FROM THE NUCLEAR SAFETY INFORMATION CENTER (NSIC) COMPUTER FILE DATED 1973 THROUGH 1976 COVERING MATERIAL PROPERTIES WITH RESPECT TO THE STRUCTURAL INTEGRITY. ALL MATERIALS IMPORTANT TO THE NUCLEAR INDUSTRY (EXCEPT CONCRETE) ARE COVERED FOR MECHANICAL PROPERTIES, CHEMICAL PROPERTIES, CORROSION, FRACTURE OR FAILURE, RADIATION DAMAGE, CREEP, CRACKING, AND SWELLING. KEYWORD, AUTHOR, AND PERMUTED-TITLE INDEXES ARE INCLUDED FOR THE CONVENIENCE OF THE USER.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA. 22161 (100.25/PULL-SIZE, 03/MP)

POWER PLANT, NUCLEAR * MATERIAL * STRUCTURAL INTEGRITY * BIBLIOGRAPHY * NRC-1 * NRC-5 * JACOBS

127081

MERRILL JC * ROBINSON GC * HOLZ PP * SMITH JE

TEST OF 8-IN.-THICK PRESSURE VESSELS. SERIES 4: INTERMEDIATE TEST VESSELS V-7 AND V-8 WITH INSIDE NOZZLE CORNER CRACKS

ORNL/NUREG/NSIC-100 P. 223 PPS. 30 TABS. 114 FIGS. 82 REFS. AUG. 1977

DESCRIBES FAILURE TESTING OF THE 90-CM-DIAM (30-IN.), 15.2-CM-THICK (6-IN.) STEEL PRESSURE VESSELS, EACH CONTAINING ONE FLAWED NOZZLE. VESSEL V-5 WAS TESTED AT 80 DEG C (190 DEG F) AND FAILED BY LEAKING WITHOUT FRACTURING AFTER EXTENSIVE STABLE CRACK GROWTH. VESSEL V-9 WAS TESTED AT 25 DEG C (75 DEG F) AND FAILED BY FRACTURING. MATERIAL PROPERTIES MEASURED BEFORE THE TESTS WERE USED FOR PRETEST AND POSTTEST FRACTURE ANALYSES. TEST RESULTS SUPPORTED BY ANALYSIS INDICATE THAT INSIDE NOZZLE CORNER CRACKS ARE NOT SUBJECT TO PLANE STRAIN UNDER PRESSURE LOADING. THE PREPARATION OF INSIDE NOZZLE CORNER CRACKS IS DESCRIBED IN DETAIL. EXTENSIVE EXPERIMENTAL DATA ARE TABULATED AND PLOTTED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

PRESSURE VESSELS * NOZZLE * CRACK * TESTING * MSST * NRC-5

127084

WHITMAN GO

HEAVY-SECTION STEEL TECHNOLOGY PROGRAM QUARTERLY PROGRESS REPORT FOR JANUARY-MARCH 1977

ORNL/NUREG/NSIC-100 P. 108 PPS. TABS. FIGS. REFS. SEPT. 1977

STRESS-INTENSITY FACTORS MEASURED ON PHOTOELASTIC MODELS OF BWR VESSELS WITH NOZZLES ARE CONSISTENT WITH VALUES FROM VESSEL BURST TESTS. INTERMEDIATE TEST VESSELS V-7B AND V-8 WERE REPAIRED BY THE HALF-HEAD WELDING TECHNIQUE PRESCRIBED IN SECTION XI OF THE ASME BOILER AND PRESSURE VESSEL CODE IN PREPARATION FOR TESTING WITH FLAWS IN THE REPAIR ZONES, AND THE FIRST TWO CRACK ARREST MODEL VESSELS WERE TESTED. THE FOURTH THERMAL SHOCK EXPERIMENT INDICATED THAT LINEAR ELASTIC FRACTURE MECHANICS ANALYSIS IS APPLICABLE TO A LOSS-OF-COOLANT TYPE OF THERMAL SHOCK. FRACTOGRAPHIC EXAMINATION OF FRACTURE SURFACES OF EARLIER EXPERIMENTS CONFIRMED THE CHARACTER OF THE FRACTURES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

MSST * STEEL * PRESSURE VESSELS * PHOTOELASTICITY * WELDING * FLAW * TESTING * NRC-5 * JACOBS

128365

MOORE SE * BRYSON JB

DESIGN CRITERIA FOR PIPING AND NOZZLES PROGRAM QUARTERLY PROGRESS REPORT FOR OCTOBER-DECEMBER 1976

ORNL/NUREG/NSIC-100 P. 19 PPS. 1 TABS. 1 FIG. 20 REFS. SEPT. 1977

SHORT SUMMARIES ARE GIVEN FOR STUDIES COMPLETED AND REPORTS PUBLISHED DURING THE QUARTER INCLUDING DIMENSIONAL CONTROLS FOR PIPING PRODUCTS, COMBINED LOADINGS ON ANSI BIG-6 TEES, AND STRESSES AT REINFORCED NOZZLES IN SPHERICAL SHELL STRUCTURES. A SUMMARY STATUS REPORT IS ALSO GIVEN FOR WORK CURRENTLY IN PROGRESS ON THE FINITE ELEMENT ANALYSIS OF ISOLATED AND CLOSELY-SPACED NOZZLES IN CYLINDRICAL PRESSURE VESSELS. FINALLY, A SUMMARY REPORT IS GIVEN ON THE ASME CODE AND PVBC COMMITTEE WORK INCLUDING THE CURRENT STATUS OF PROPOSED CODE RULES REVISIONS THAT ARE BASED, IN PART, ON WORK CONDUCTED UNDER THE ORNL DESIGN CRITERIA PROGRAM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

PIPES AND PIPE FITTINGS * NOZZLE * SPHERE * SHELL * STRESS ANALYSIS * CODES AND STANDARDS * NRC-5 * JACOBS

128428

BRYSON JB * JOHNSON WG * BASS BR

STRESSES IN REINFORCED NOZZLE-CYLINDER ATTACHMENTS UNDER INTERNAL PRESSURE LOADING ANALYZED BY THE FINITE-ELEMENT METHOD -- A PARAMETER STUDY

ORNL/NUREG/NSIC-100 P. 215 PPS. 7 TABS. 28 FIGS. 13 REFS. OCT. 1977

128629 *CONTINUED*

THE OBJECTIVE IS TO SUMMARIZE AND PRESENT THE MORE IMPORTANT AND RELEVANT RESULTS OF THE PARAMETER STUDY. THREE TYPES OF MODELS WERE EXAMINED: 6 WHICH WERE MINIMALLY REINFORCED BY A SMALL FILLET RADIUS AT THE JUNCTION AS SPECIFIED BY THE CODE; NOZZLE THICKENED, "STANDARD" REINFORCEMENT (S1) MODELS; AND 30 DEG TRIANGULAR PAD (P30) MODELS. NINETEEN OF THE MODELS ANALYZED WERE OF THE LATTER TWO TYPES. FUTURE WORK WILL BE TO RUN THE ORTHOGONAL SET OF FORCE AND MOMENT LOADINGS APPLIED TO THE BRANCH AND FREE END OF THE DUT FOR THESE SAME 25 MODELS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

STRESS ANALYSIS • NOZZLE • CYLINDER • PRESSURE, INTERNAL • COMPUTER PROGRAM • PRESSURE VESSELS • CODES AND STANDARDS • NRC-S • JACOBS

131064

TWO FIN • DIVISION JV • USED RA • NOZZLE SE
STRESS ANALYSIS OF CYLINDRICAL PRESSURE VESSELS WITH CLOSELY SPACED NOZZLES BY THE FINITE-ELEMENT METHOD.
VOLUME I. STRESS ANALYSIS OF VESSELS WITH TWO CLOSELY SPACED NOZZLES UNDER INTERNAL PRESSURE
OAK RIDGE NATIONAL LAB., TENN.
ORNL/TN-10/VI •. 113 PPS, TABS, FIGS, REFS, NOV. 1977

A FINITE-ELEMENT COMPUTER PROGRAM, MULT-NOZZLE, WAS DEVELOPED FOR THE STRESS ANALYSIS. A COMPLETE DISCUSSION OF MULT-NOZZLE IS PRESENTED IN FOUR VOLUMES. THE PRESENT VOLUME DEVELOPS THE FINITE-ELEMENT IDEALIZATION FOR PRESSURE VESSELS WITH TWO IDENTICAL RADIIALLY ATTACHED CLOSELY SPACED NOZZLES FOR INTERNAL PRESSURE LOADING. THE NOZZLES MAY BE UNREINFORCED OR FULLY REINFORCED ACCORDING TO THE RULES OF ASME AND MAY BE LOCATED IN EITHER A LONGITUDINAL OR A TRANSVERSE PLANE OF THE VESSEL. VALIDATION OF THE PROGRAM FOR ANALYZING THIS TYPE OF STRUCTURE IS DEMONSTRATED BY THE ANALYSIS OF THREE TWO-NOZZLE PRESSURE VESSEL MODELS AND COMPARISON OF RESULTS WITH EXPERIMENTAL DATA. IN GENERAL, QUITE SATISFACTORY RESULTS WERE OBTAINED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

STRESS ANALYSIS • PRESSURE VESSELS • NOZZLE • DESIGN CRITERIA • CODES AND STANDARDS • R AND D PROGRAM • PRESSURE, INTERNAL • COMPUTER PROGRAM • NRC-S • JACOBS

131066

CHEVYRON RD • GALT SE
PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO A POR LOCA-ECC THERMAL SHOCK: EXPERIMENTS TSE-3 AND TSE-4 AND UPDATE OF TSE-1 AND TSE-2 ANALYSIS
OAK RIDGE NATIONAL LAB., TENN.
ORNL/TN-22 •. 145 PPS, 6 TABS, 65 FIGS, 17 REFS, DEC. 1977

THE THERMAL SHOCK PROGRAM WAS ORIENTED TOWARD DETERMINING THE VALIDITY OF LINEAR ELASTIC FRACTURE MECHANICS FOR SMALL CRACKS UNDER LOCA-ECC THERMAL SHOCK CONDITIONS. WITH A SUBSEQUENT EFFORT ON DEMONSTRATING WARM PRESTRESSING UNDER BASICALLY THE SAME CONDITIONS. IN THE FIRST EXPERIMENT, INITIATION WAS NOT EXPECTED AND DID NOT OCCUR. ALTHOUGH THERE WAS A SMALL AMOUNT OF SUBCRITICAL CRACK GROWTH. IN THE SECOND EXPERIMENT, INITIATION OF A SEMICIRCULAR PLAN TOOK PLACE AS EXPECTED; THE FINAL LENGTH ALONG THE SURFACE WAS ABOUT FOUR TIMES THE INITIAL LENGTH, BUT THERE WAS NO RADIAL GROWTH. THE THIRD AND FOURTH EXPERIMENTS WERE SIMILAR, AND THE LONG AXIAL PLAN INITIATED IN GOOD AGREEMENT WITH PREDICTIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, POR • ACCIDENT, LOSS OF COOLANT • PRESSURE VESSELS • FRACTURE TOUGHNESS • CRACK • THERMAL ANALYSIS • GROWTH/DEVELOPMENT • NRC-S

131126

WITMAN GD
HEAVY-SECTION STEEL TECHNOLOGY PROGRAM QUARTERLY PROGRESS REPORT FOR APRIL-JUNE 1977
OAK RIDGE NATIONAL LAB., TENN.
ORNL/TN-10/VI •. 93 PPS, TABS, FIGS, REFS, DEC. 1977

HST COMPRISES STUDIES RELATING TO ALL AREAS OF THE TECHNOLOGY OF THE MATERIALS FABRICATED INTO THICK-SECTION PRIMARY-COOLANT CONTAINMENT SYSTEMS OF LIGHT-WATER-COOLED NUCLEAR POWER REACTORS. THE PRINCIPAL AREA OF INVESTIGATION IS THE BEHAVIOR AND STRUCTURAL INTEGRITY OF STEEL PRESSURE VESSELS CONTAINING CRACKLIKE FLAWS. CURRENT WORK IS ORGANIZED INTO THE FOLLOWING TASKS: (1) PROGRAM ADMINISTRATION, (2) FRACTURE MECHANICS ANALYSES AND INVESTIGATIONS, (3) EFFECT OF HIGH-TEMPERATURE PRIMARY REACTOR WATER ON THE SUBCRITICAL CRACK GROWTH OF REACTOR VESSEL STEEL, (4) INVESTIGATIONS OF IRRADIATED MATERIALS, (5) PRESSURE VESSEL INVESTIGATIONS, (6) THERMAL SHOCK INVESTIGATIONS, (7) FOREIGN RESEARCH, AND (8) PRESTRESSED CONCRETE REACTOR VESSEL (PCRV) TENDON CORROSION STUDIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

HST • R AND D PROGRAM • SAFETY PROGRAM • FRACTURE TOUGHNESS • HIGH TEMPERATURE • CRACK • GROWTH/DEVELOPMENT • IRRADIATION TESTING • PRESSURE VESSELS • STEEL • WELDING • STRUCTURAL INTEGRITY • NRC-S

6. NO - WATER REACTOR SAFETY RESEARCH, SITE SAFETY RESEARCH

131904
 BUSCHBACH TC
 NEW NARRIS SEISMOTECTONIC STUDY ACTIVITIES DURING FISCAL YEAR 1977
 ILL. STATE GEOLOGICAL SURVEY, URBANA
 NUREG-0379 O., 61 PPS, 1 TAB, 17 FIGS, NOV. 1977

THE STUDY IS DESIGNED TO DEFINE THE STRUCTURAL SETTING AND TECTONIC HISTORY IN THE AREA IN ORDER TO REALISTICALLY EVALUATE EARTHQUAKE RISKS IN THE SITING OF NUCLEAR FACILITIES. DURING THE FIRST YEAR OF THE FIVE-YEAR PROGRAM THE PROPOSALS WERE SUBMITTED AND APPROVED. THIS YEAR WAS USED PRINCIPALLY FOR MAKING AN INVENTORY OF EXISTING DATA AND SUPPORTING RESEARCH IN AREAS WHERE THE NEEDS WERE OBVIOUS. NEW INFORMATION FROM AEROMAGNETIC, GROUND MAGNETIC, AND GRAVITY SURVEYS HAS BEEN OBTAINED, AND THE DATA ARE CURRENTLY BEING PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
 R AND D PROGRAM + SAFETY PROGRAM + SEISMOLOGY + TECTONICS + SURVEY + TENNESSEE + NEWARK, MISSISSIPPI + GEOLOGY + NRC-6

128625
 SOCIOECONOMIC IMPACTS: NUCLEAR POWER STATION SITING,
 POLICY RESEARCH ASSOCIATES, PA
 NUREG-0150 O., 146 PPS, 10 TABS, REFS, APR 1977

THE RURAL INDUSTRIAL DEVELOPMENT LITERATURE IS UTILIZED TO GAIN INSIGHTS IN THE SOCIOECONOMIC EFFECTS OF NUCLEAR POWER STATIONS. PREVIOUS STUDIES OF LARGE INDUSTRIAL FACILITIES IN SMALL TOWNS HAVE IMPORTANT IMPLICATIONS FOR ATTEMPTS TO UNDERSTAND AND ANTICIPATE THE IMPACTS OF NUCLEAR STATIONS. EVEN A CASUAL REVIEW OF THE NUCLEAR DEVELOPMENT LITERATURE, HOWEVER, REVEALS THAT INDUSTRIALIZATION RESEARCH IN RURAL SOCIOLOGY, ECONOMIC GEOGRAPHY AND AGRICULTURAL ECONOMICS HAS BEEN LARGELY IGNORED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
 POWER PLANT, NUCLEAR + SITING + ECONOMICS + EFFECT + SOCIO/PHILOSOPHICAL CONSIDERATION + NRC-6

132210
 HENZE WJ + BRATLE LV + KELLER GR + LIDIAN EG
 A TECTONIC OVERVIEW OF THE CENTRAL MIDCONTINENT
 PURDUE UNIVERSITY, WEST LAFAYETTE, IN.
 NUREG-0302 O., 105 PPS, 12 FIGS, REFS, JUL 1977

THE CENTRAL MIDCONTINENT OF THE UNITED STATES IS A PART OF THE NORTH AMERICAN CRATON WHICH, EXCEPT FOR MINOR DEFORMATION, IS COMMONLY REGARDED AS HAVING BEEN A STABLE PLATFORM SINCE CAMBRIAN TIME. HOWEVER, RECENT GEOLOGICAL AND GEOPHYSICAL INVESTIGATIONS, PARTICULARLY SEISMICITY STUDIES, HAVE REVEALED THAT NEAR THE CONFLUENCE OF THE OHIO AND MISSISSIPPI RIVERS, THE CRATON IS STRUCTURALLY COMPLEX AND IS TECTONICALLY ACTIVE. THIS OVERVIEW SUMMARIZES THE STATE OF KNOWLEDGE OF THE CRUST, INCLUDING STRUCTURAL HISTORY, BASEMENT GEOLOGY, AND CONTEMPORARY GEODYNAMICS OF THE CENTRAL MIDCONTINENT. THE TECTONIC MODELS WHICH HAVE BEEN SUGGESTED TO EXPLAIN THE CONTEMPORARY TECTONISM ARE REVIEWED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
 UNITED STATES + OHIO + OHIO + OHIO, MISSISSIPPI + SEISMIC ZONE + TECTONICS + GEOLOGY + MODEL + FAULT + NRC-6

129473
 ALLENSWORTH JA + FINGER JT + HILLOT JA + MURPIN BB + ROEPAN R + VANDEVENDER SG
 UNDERGROUND SITING OF NUCLEAR POWER PLANTS: POTENTIAL BENEFITS AND PENALTIES
 SANDIA LABS., ALBUQUERQUE, N.M.
 SAND-70-0012 + NUREG-02555 O., 262 PPS, 4 TABS, 7 FIGS, AUG. 1977

THE POTENTIAL BENEFITS OF UNDERGROUND SITING ARE: (1) CONDITIONAL IMPROVED CONTAINMENT IN A CORE MELTDOWN ACCIDENT AND (2) PROTECTION OF THE PLANTS FROM AIRCRAFT CRASHES OR BARRAGE MUNITIONS. UNDERGROUND SITING HAS NEGLECTIBLE ADVANTAGE OVER SURFACE SITING FOR CONTAINMENT OF ACCIDENTS WHICH DO NOT INVOLVE MELTDOWN OF THE REACTOR CORE, SINCE CURRENT SURFACE PLANT DESIGN REQUIREMENTS PROVIDE A HIGH DEGREE OF PUBLIC PROTECTION AGAINST THESE ACCIDENTS. SEALING OF CONTAINMENT PENETRATIONS APPEARS TO BE A CRITICAL DESIGN PROBLEM OF UNDERGROUND SITING. GROUNDWATER CONTAMINATION IS MORE LIKELY IN AN UNDERGROUND PLANT, BECAUSE OF THE HIGHER PROBABILITY OF ESCAPE OF CONTAMINATED SUMP WATER. THE PRINCIPAL POTENTIAL PENALTIES ARE CONSTRUCTION DIFFICULTIES, OPERATIONAL PROBLEMS, AND COST. THE INCREASED COST OF UNDERGROUND SITING IS ESTIMATED TO BE 20-40 PERCENT GREATER THAN FOR A SURFACE FACILITY. THE ESCAPED BENEFITS OF UNDERGROUND SITING IN TERMS OF IMPROVED SAFETY DO NOT APPEAR TO OFF-SET THE PENALTIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
 CONTAINMENT, UNDERGROUND + ECONOMICS + COMPARISON + COST BENEFIT + SABOTAGE + SEISMIC DESIGN + GROUND WATER + CONTAMINATION + ACCIDENT ANALYSIS + CORE MELTDOWN + NRC-6 + JACOB.

7. B7 - ADVANCED REACTOR SAFETY RESEARCH: FAST REACTORS

137881
KALIQUHAN + KIER PH + HUMMEL WH
PHYSICS CALCULATIONS FOR THE CLINCH RIVER BREEDER REACTOR
ARGONNE NATIONAL LAB., ILL.
ANL-77-27 +. 154 PPS, TABS, FIGS, JUNE 1977

CALCULATIONS OF POWER AND SODIUM VOID REACTIVITY, UNVOIDED AND VOIDED DOPPLER COEFFICIENTS AND STEEL AND FUEL WORTH HAVE BEEN PERFORMED USING DIFFUSION THEORY AND FIRST-ORDER PERTURBATION THEORY FOR THE LOW DISCHARGE PU-FUELED CORE AT BOL, THE FFPY-GRADE PU-FUELED CORE AT BOL AND FOR THE BEGINNING AND END OF EQUILIBRIUM CYCLE OF THE LOW-PU-FUELED CORE. RESULTS OF BURNUP AND BREEDING RATIO CALCULATIONS PERFORMED TO OBTAIN THE REACTOR COMPOSITIONS DURING THE EQUILIBRIUM CYCLE ARE ALSO REPORTED. EFFECTS OF SODIUM AND STEEL CONTENTS ON THE DISTRIBUTIONS OF SODIUM VOID REACTIVITY AND STEEL WORTH HAVE BEEN STUDIED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, LMFBR + CLINCH RIVER (LMFBR) + REACTIVITY COEFFICIENT + COMPUTER PROGRAM, DIGITAL + SODIUM COEFFICIENT + VOID COEFFICIENT + MATHEMATICAL TREATMENT + NRC-7

137883
PHYSICS OF REACTOR SAFETY, QUARTERLY REPORT JANUARY-MARCH 1977
ARGONNE NATIONAL LAB., ILL.
ANL-77-48 +. 51 PPS, 10 TABS, 26 FIGS, 16 REFS, JUNE 1977

THIS QUARTERLY PROGRESS REPORT SUMMARIZES WORK DONE IN ARGONNE NATIONAL LABORATORY'S APPLIED PHYSICS DIVISION AND COMPONENTS TECHNOLOGY DIVISION FOR THE DIVISION OF REACTOR SAFETY RESEARCH OF THE U.S. NUCLEAR REGULATORY COMMISSION APPLIED PHYSICS DIVISION INCLUDES REPORTS ON REACTOR SAFETY PROGRAM BY MEMBERS OF THE REACTOR SAFETY APPRAISALS GROUP MONTE CARLO ANALYSIS OF THE THEORETICAL FAST REACTOR PHYSICS GROUP, AND PLANNING OF DEMO SAFETY-RELATED CRITICAL EXPERIMENTS BY MEMBERS OF THE ZERO POWER REACTOR (ZPR) PLANNING AND EXPERIMENTS GROUP. WORK ON REACTOR CORE THERMAL-HYDRAULIC PERFORMED IN THE COMPONENTS TECHNOLOGY DIVISION IS ALSO INCLUDED IN THIS REPORT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR PHYSICS + SAFETY ANALYSIS + SAFETY PRINCIPLES AND PHILOSOPHY + THERMAL HYDRAULIC ANALYSIS + MONTE CARLO + PIN PILE EXPERIMENT + CRITICAL ASSEMBLY + ACCIDENT ANALYSIS + REACTOR, LMFBR + NRC-7 + JACOBS

138249
PHYSICS OF REACTOR SAFETY QUARTERLY REPORT APRIL-JUNE 1977
ARGONNE NATIONAL LAB., ILL.
ANL-77-89 +. 48 PPS, REFS, SEPT. 1977

INCLUDES REPORTS ON REACTOR SAFETY PROGRAM BY MEMBERS OF THE REACTOR SAFETY APPRAISALS GROUP. MONTE CARLO ANALYSIS OF SAFETY-RELATED CRITICAL ASSEMBLY EXPERIMENTS BY MEMBERS OF THE THEORETICAL FAST REACTOR PHYSICS GROUP, AND PLANNING OF SAFETY-RELATED CRITICAL EXPERIMENTS BY MEMBERS OF THE ZERO POWER REACTOR (ZPR) PLANNING AND EXPERIMENTS GROUP. WORK ON REACTOR CORE THERMAL-HYDRAULIC CODE DEVELOPMENT IS ALSO INCLUDED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + REACTOR, SAFETY RESEARCH + SAFETY PROGRAM + SAFETY ANALYSIS + EXPERIMENT + CRITICAL ASSEMBLY + MONTE CARLO + THERMAL HYDRAULIC ANALYSIS + REACTOR PHYSICS + NRC-7 + JACOBS

132214
CORANUS WH + SCHRETT RC + SHA WT
NUMERICAL RESULTS OBTAINED FROM THE THREE DIMENSIONAL TRANSIENT SINGLE PHASE VERSION OF THE COMIX COMPUTER CODE
ARGONNE NATIONAL LAB., ILL.
NUREG-0355 + ANL-CT-78-2 +. 91 PPS, 82 FIGS, 9 REFS, OCT. 1977

RESULTS FROM THE FIRST THREE APPLICATIONS OF THE THREE-DIMENSIONAL, TRANSIENT, SINGLE-PHASE VERSION OF THE THERMAL-HYDRAULICS COMPUTER CODE, COMIX, ARE PRESENTED. THE APPLICATIONS INCLUDE AN LMFBR OUTLET PLENUM, A HORIZONTAL PIPE, AND A 19-PI. LMFBR HEXAGONAL FUEL ASSEMBLY. EACH CASE IS DESCRIBED AND THE RESULTS PRESENTED. ALL THREE APPLICATIONS USE LIQUID SODIUM AS THE FLUID. THE PLENUM AND PIPE ANALYSES ARE TRANSIENT, WHILE THE FUEL ASSEMBLY INVOLVES A STEADY-STATE ANALYSIS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + REACTOR, LMFBR + COMPUTER PROGRAM + THERMAL HYDRAULIC ANALYSIS + FUEL ELEMENT CLUSTER + SODIUM + TRANSIENT + STEADY STATE + NRC-7

129282
GIESERE JA + JORDAN M + LEE EV
AEROSOL MEASUREMENTS AND MODELING FOR FAST REACTOR SAFETY: QUARTERLY PROGRESS REPORT FOR JANUARY 1 THROUGH MARCH 31, 1977
BATTELLE COLUMBUS LABS., OHIO
OHIO-NUREG-1075 +. 18 PPS, 8 FIGS, 7 REFS, JUNE 30, 1977

129882 *CONTINUED*

TECHNICAL PROGRESS DURING THIS QUARTER HAS MADE ON EXPERIMENTAL MEASUREMENTS OF SODIUM OXIDE AGGLOMERATE CHARACTERISTICS, HAARM-2 COMPUTER CODE IMPROVEMENTS, CONSIDERATIONS OF SIZE OR SCALE EFFECTS AS IMPLIED BY HAARM-2 CALCULATIONS, AND FUEL MATERIALS VAPORIZATION FOR AEROSOL STUDIES. IN ADDITION, A MID-YEAR REVIEW WAS PRESENTED TO NRC STAFF ON FEBRUARY 11 AND A MEETING ON LARGE SCALE AEROSOL EXPERIMENTS WAS ATTENDED ON MARCH 15 AND 16. TECHNICAL PROGRESS IS DISCUSSED IN THIS REPORT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

AEROSOL * ANALYTICAL MODEL * MEASUREMENT * COMPUTER PROGRAM * LABORATORY EXPERIMENT * REACTOR, LMFBR * OXIDE * SODIUM * NRC-7 * JACOBS

129877

GIESSEN JA * REED LO * JORDAN M * LEE RN
CHARACTERISTICS OF AGGLOMERATES OF SODIUM OXIDE AEROSOL PARTICLES
BATTELLE COLUMBUS LABS., OHIO
ENR-NUREG-1977 *. 32 PPS, 1 TAB, 10 FIGS, 15 REFS, AUG. 1, 1977

ACCURATE MICROSCOPIC PREDICTIONS OF AEROSOL POPULATION BEHAVIOR WITHIN ENCLOSED CONTAINMENTS ARE KNOWN TO DEPEND STRONGLY UPON THE MICROSCOPIC CHARACTERISTICS OF THE INDIVIDUAL PARTICLES COMPRISING THE POPULATION. FOR EXAMPLE, PARTICLE SHAPES AND DENSITIES HAVE PRONOUNCED EFFECTS ON THE SETTLING VELOCITIES OF INDIVIDUAL AEROSOLS. ALSO, COAGULATION RATES DUE TO MECHANISMS WHICH PRODUCE RELATIVE MOTIONS BETWEEN PARTICLES WITHIN THE SUSPENDED AEROSOL ARE KNOWN TO DEPEND UPON THE CROSS SECTIONAL AREAS OF THE INDIVIDUAL PARTICLES. HENCE, IT HAS BEEN THE PRIMARY CONCERN OF THIS STUDY TO EXAMINE EXPERIMENTALLY THE MICROSCOPIC CHARACTERISTICS OF SODIUM OXIDE AEROSOLS PRODUCED IN AIR. THE RESULTS OF THESE MEASUREMENTS CAN NOW BE INCORPORATED INTO THE VARIOUS MICROSCOPIC AEROSOL BEHAVIOR PREDICTION MODELS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

AEROSOL * SAMPLING * AGGLOMERATE * THERMOPHORESIS * ANALYTICAL MODEL * REACTOR, LMFBR * OXIDE * SODIUM * NRC-7 * JACOBS

132943

GIESSEN JA * JORDAN M * SCHMIDT EV * LEE RN
AEROSOL MEASUREMENTS AND MODELING FOR FAST REACTOR SAFETY: QUARTERLY PROGRESS REPORT FOR APRIL 1 THROUGH JUNE 30, 1977
BATTELLE COLUMBUS LABS., OHIO
ENR-NUREG-1983 *. 24 PPS, 1 TAB, 9 FIGS, 4 REFS, OCT. 28, 1977

TECHNICAL EFFORTS THIS QUARTER WERE CONCENTRATED ON IMPROVING THE HAARM-2 CODE PRIMARILY BY ADDING THE RESULTS OF THE SODIUM OXIDE AGGLOMERATE MEASUREMENTS, ON FORMULATING THE MATHEMATICAL BASIS FOR REFERENCE CODE DEVELOPMENT, ON INVESTIGATING THE POSSIBILITIES FOR USING POLYNOMIALS TO EXPRESS PARTICLE SIZE DISTRIBUTIONS IN THE REFERENCE CODE, AND ON EVALUATING THE USE OF A LASER DOPPLER INTERFEROMETRY SYSTEM TO MEASURE SODIUM OXIDE AGGLOMERATE SIZES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM * SAFETY PROGRAM * REACTOR, LMFBR * AEROSOL * ANALYTICAL MODEL * COMPUTER PROGRAM * SODIUM * OXIDE * AGGLOMERATE * PARTICLE SIZE DISTRIBUTION * NRC-7

129475

ROMANO AJ
REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT, JANUARY 1-MARCH 31, 1977
BROOKHAVEN NATIONAL LAB., UPTON, N.Y.
ENR-NUREG-50001 *. 210 PPS, MAY 1977

THE PROJECTS REPORTED EACH QUARTER ARE THE FOLLOWING: GAS REACTOR SAFETY EVALUATION, THOR CODE DEVELOPMENT, SSC CODE DEVELOPMENT, LMFBR AND LWR SAFETY EXPERIMENTS, FAST REACTOR SAFETY CODE VALIDATION, TECHNICAL COORDINATION OF STRUCTURAL INTEGRITY, AND FAST REACTOR SAFETY RELIABILITY ASSESSMENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, LMFBR * REACTOR, HTGR * REACTOR, LWR * COMPUTER PROGRAM, DIGITAL * CODES AND STANDARDS * ACCIDENT, CORE DISRUPTIVE * JACOBS * SAFETY ANALYSIS * ACCIDENT, CONSEQUENCES * NRC-4 * NRC-7 * NRC-8

129180

ROMANO AJ
REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT APRIL 1-JUNE 30, 1977
BROOKHAVEN NATIONAL LAB., UPTON, N.Y.
ENR-NUREG-50003 *. 193 PPS, TABS, FIGS, JULY 1977

THE REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT DESCRIBES CURRENT ACTIVITIES AND TECHNICAL PROGRESS IN THE PROGRAMS AT BROOKHAVEN NATIONAL LABORATORY SPONSORED BY THE USNRC. REACTOR SAFETY RESEARCH DIVISION. THE PROJECTS REPORTED EACH QUARTER ARE THE FOLLOWING: GAS REACTOR SAFETY EVALUATION, THOR CODE DEVELOPMENT, SSC CODE DEVELOPMENT, LMFBR AND LWR SAFETY EXPERIMENTS, FAST REACTOR SAFETY CODE VALIDATION, TECHNICAL COORDINATION OF STRUCTURAL INTEGRITY,

120100 *CONTINUED*
AND FAST REACTOR SAFETY RELIABILITY ASSESSMENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

REACTOR, LWR + REACTOR, LWR + REACTOR, MTR + SAFETY ANALYSIS + COMPUTER PROGRAM, DIGITAL + CODES AND STANDARDS + ACCIDENT, CORE DISRUPTIVE + CORE MELTDOWN + ACCIDENT, LOSS OF COOLANT + BLOWDOWN + ACCIDENT ANALYSIS + BOILING + NRC-4 + NRC-7 + NRC-8

120212
GIAN V. ACHARNAL AN
A PIPE-BREAK MODEL FOR LWR SAFETY ANALYSIS
BROOKHAVEN NATIONAL LAB., UPTON, N.Y.
BNL-NR02-50408 +. 18 PPS, 10 FIGS, 8 REFS, AUG. 1977

A MODEL HAS BEEN DEVELOPED TO PREDICT THE COOLANT DISCHARGE RATE THROUGH THE BREAK DURING AN LWR PIPE BREAK ACCIDENT. THE COOLANT PIPE IS CONSIDERED TO BE SURROUNDED BY AN OUTER PIPE. IF THE GAP SPACING BETWEEN THE TWO PIPES IS LARGE, THE DISCHARGED FLUID IS TREATED AS AN INVISCID FREE JET. IF THE GAP SPACING IS SMALL, THEN CONFINED FLOW ACCOMPANIED BY PRESSURE LOSSES IS CONSIDERED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + REACTOR, LWR + ACCIDENT, LOSS OF COOLANT + DISCHARGE + RATE + JET + SAFETY ANALYSIS + FLOW + MODEL + NRC-7

120213
GUSTAVSEN BR + CHEN JC + KAZIMI HS
HEAT TRANSFER AND FLUID DYNAMIC CHARACTERISTICS OF INTERNALLY HEATED BOILING POOLS
BROOKHAVEN NATIONAL LAB., UPTON, N.Y.
BNL-NR02-50722 +. 100 PPS, 5 TABS, 27 FIGS, 21 REFS, SEPT. 1977

AN ANALYTICAL AND EXPERIMENTAL INVESTIGATION OF THE HEAT TRANSFER FROM AN INTERNALLY HEATED BOILING POOL TO A VERTICAL WALL HAS BEEN CONDUCTED. A MODEL IS PRESENTED FOR THE HEAT TRANSFER BASED ON COMBINED MODES OF NATURAL AND FORCED CONVECTION FLOW IN THE LIQUID BOUNDARY LAYER AT THE VERTICAL WALL. THE EXPERIMENTAL RESULTS, USING JOWLE HEATING OF A ZnSO₄-H₂O SOLUTION, INDICATE THAT THE POOL MAY DEVELOP A LOW-DENSITY "FOAM-LIKE" REGIME AS WELL AS A DENSE "TURBULENT" REGIME. THE ANALYTIC MODEL COMPARES FAVORABLY WITH THE RESULTS IN THE "DENSE" REGIME WHICH SHOW THAT THE HIGHEST HEAT FLUXES DEVELOP NEAR THE UPPER EDGE OF THE POOL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + REACTOR, LWR + HEAT TRANSFER, BOILING + POOL BOILING + MODEL + URANIUM DIOXIDE + NRC-7

120400
SUN T-HR + OKMENT D + CATTON I + KASTENBERG WE
A SIMPLIFIED METHOD OF COMPUTING CLAD AND FUEL STRAIN AND STRESS DURING IRRADIATION
UNIV OF CALIF., LOS ANGELES
NR02-5200 +. 200 PPS, FIGS, JUNE 1977

THE DISSERTATION DEVELOPS A SIMPLIFIED, FAST-RUNNING AXISYMMETRIC COMPUTER CODE, NAMED KRASS (KRASS RUNNING ANALYSIS FOR STRESS AND STRAIN), INTENDED FOR PREDICTION OF FUEL ELEMENT CONDITIONS AFTER A LONG-TERM STEADY-STATE OPERATION IN A LWR. KRASS DIVIDES THE FUEL PELLETS INTO TWO ZONES, AN INNER, HIGHLY PLASTIC HOT REGION, AND AN OUTER, COOLER REGION WHICH TOGETHER WITH THE CLADDING CAN UNDERGO CREEP. RESULTS OF CALCULATIONS INDICATE THAT AXIAL VARIATION OF THE FUEL-CLAD MECHANICAL INTERACTION DEPENDS ON THE FUEL "E" TO BURNUP RATIO, AS WELL AS ON THE POWER RATINGS. THE LARGEST FUEL-CLAD MECHANICAL INTERACTION GENERALLY OCCURS IN THE LOWER THIRD OF THE FUEL ELEMENT WHILE AT LOW POWER RATINGS. THIS INTERACTION IS LARGEST AT THE AXIAL MIDSECTION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM + FUEL ROD + REACTOR, LWR + STRESS ANALYSIS + IRRADIATION TESTING + FUEL CLAD INTERACTION + MATHEMATICAL TREATMENT + STEEL, STAINLESS + SWELLING + NRC-7 + JACOBS

120100
SUD-ANTILLA AJ + CATTON I + KASTENBERG WE
NATURAL CONVECTION IN HORIZONTAL FLUID LAYERS
UNIV. OF CALIF., LOS ANGELES
NR02-5201 +. 150 PPS, 3 TABS, 20 FIGS, JUNE 1977

REPORTS ON BOTH EXPERIMENTAL AND THEORETICAL WORK ON THE PROBLEM OF THERMAL CONVECTION IN HORIZONTAL FLUID LAYERS. THE EXPERIMENTAL WORK INCLUDES DEVELOPING AND USING A THERMAL CONVECTION CELL TO OBTAIN MEASUREMENTS OF THE HEAT FLUX AND TURBULENT CORE TEMPERATURE OF A HORIZONTAL LAYER OF FLUID HEATED INTERNALLY AND SUBJECT TO BOTH STABILIZING AND DESTABILIZING TEMPERATURE DIFFERENCES. THE THEORETICAL WORK CONSISTS OF THE DERIVATION, SOLUTION AND USE OF THE HEAT FIELD EQUATIONS FOR STUDY OF THERMALLY DRIVEN CONVECTION IN HORIZONTAL LAYERS OF INFINITE EXTENT. THE DERIVATION RESULTS IN A COUPLED SET OF MOMENTUM AND ENERGY EQUATIONS, WHICH WERE SIMPLIFIED BY USING THE INFINITE PRANDTL NUMBER LIMIT AND NEGLECTING DIRECT INTERMOLECULAR

129100 *CONTINUED*

INTERACTION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

HEAT TRANSFER, NATURAL CONVECTION • CORE MELTDOWN • CORE CATCHER • OUT OF PILE EXPERIMENT • ANALYTICAL TECHNIQUE • NRC-7 • JACOBS

129105

SHIR VE • CATTON I • KASTENBERG VE
STUDY OF DRYOUT HEAT FLUXES IN BEDS OF INDUCTIVELY HEATED PARTICLES
UNIV. OF CALIF., LOS ANGELES
NUREG-0262 • 81 PPS, 12 TABS, 23 FIGS, 13 REFS, JUNE 1977

REPORTS ON EXPERIMENTAL OBSERVATIONS OF THE DRYOUT HEAT FLUXES FOR INDUCTIVELY HEATED PARTICULATE BEDS. THE DATA WERE OBTAINED WHEN STEEL AND LEAD PARTICLES IN THE SIZE DISTRIBUTION 205-707 MICRONS WERE PLACED IN A 6.7 CM. DIAMETER PYREX GLASS JAR AND INDUCTIVELY HEATED BY PASSING RADIO FREQUENCY CURRENT THROUGH A 13.3 CM. DIAMETER MULTI-TURN WORM COIL ENCIRCLING THE JAR. DISTILLED WATER, METHANOL AND ACETONE WERE USED AS COOLANTS IN THE EXPERIMENTS. WHILE THE BED HEIGHT WAS VARIED FROM 1.0 CM. TO 8.0 CM. THE HEAT FLUX DATA OF THE PRESENT INVESTIGATION AND THAT FROM TWO PREVIOUS INVESTIGATIONS MADE AT ARGONNE LABORATORY AND AT UCLA HAVE BEEN CORRELATED WITH SEMI-THEORETICAL CORRELATIONS BASED ON THE PROPOSED HYDRODYNAMIC MODELS. AN APPLICATION OF THE PRESENT STUDY HAS BEEN MADE TO DETERMINE THE MAXIMUM COOLABLE DEPTHS OF THE CORE DEBRIS AS A FUNCTION OF THE PARTICLE SIZE, BED POROSITY AND DECAY HEAT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

ACCIDENT, CORE DISRUPTIVE • FUEL MELTDOWN • CORE MELTDOWN • DECAY HEAT • HEAT FLUX, DRYOUT • COMPARISON, THEORY AND EXPERIENCE • OUT OF PILE EXPERIMENT • NRC-7 • JACOBS

129104

AMAKES VE • CATTON I • KASTENBERG VE
AN EXPERIMENTAL STUDY OF THE MOLTEN GLASS/WATER THERMAL INTERACTION
UNIV. OF CALIF., LOS ANGELES
NUREG-0263 • 39 PPS, 3 TABS, 11 FIGS, 14 REFS, JUNE 1977

MOLTEN GLASS INTERACTS EXPLOSIVELY WITH WATER UNDER CERTAIN CONTACT MODE CONDITIONS. THE CONTACT MODE FOUND EXPLOSIVE IS AS FOLLOWS: MOLTEN GLASS ENTERS THE WATER BATH IN THE FILM BOILING REGIME (AS PREDICTED BY HENRY'S CORRELATION) AND SOON AFTER ENTRY, THE VAPOR FILM IS PERTURBED SUFFICIENTLY BY AN EXTERNAL PRESSURE PULSE. THE ENSUING REACTION PROCEEDS BASICALLY ALONG THE SAME LINES AS ENERGETIC FUEL/WATER INTERACTIONS OBSERVED BY SEVERAL INVESTIGATORS. IN THE ABSENCE OF THIS PRESSURE PULSE, THE EVENT IS NON-ENERGETIC. THE PRESENT FINDINGS ARE FOR A COMBINATION IN WHICH THE HOT MATERIAL HAS A VERY LOW THERMAL DIFFUSIVITY AND THE CALCULATED INTERFACE TEMPERATURE IS SIGNIFICANTLY (APPROX. 175 C) BELOW ITS MELTING TEMPERATURE. THIS IS SIMILAR TO THE CHARACTERISTICS OF THE UO₂/SODIUM COMBINATION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

REACTOR, LWR • ACCIDENT, CORE DISRUPTIVE • FUEL MELTDOWN • FUEL COOLANT INTERACTION • OUT OF PILE EXPERIMENT • NRC-7 • JACOBS

129109

WONG KU • CATTON I • KASTENBERG VE
A MECHANISTIC STUDY OF FUEL FREEZING, CHANNEL PLUGGING AND CONTINUED COOLABILITY DURING FAST REACTOR OVERPOWER EXCURSIONS
UNIV. OF CALIF., LOS ANGELES
NUREG-0310 • 105 PPS, 4 TABS, 61 FIGS, 35 REFS, JUNE 1977

THIS WORK PRESENTS A MECHANISTIC MODEL WHICH DESCRIBES EVENTS FOLLOWING FUEL PIN FAILURE WHICH MAY LEAD TO IN-CHANNEL FUEL PLATEOUT. THE THERMAL AND HYDRAULIC EFFECTS OF THE PLATE-OUT FUEL ARE ALSO EVALUATED. RESULTS OF THE PRESENT MODEL INDICATE THAT DURING A SLOW TRANSIENT OVERPOWER ACCIDENT, IN-CORE FUEL FREEZING AND CHANNEL PLUGGING MAY BE EXPECTED TO OCCUR. THE EXTENT OF FUEL FREEZING AND CHANNEL PLUGGING ARE SENSITIVE TO FUEL PARTICLE SIZE, AMOUNT OF FUEL INJECTED AND AMOUNT OF FISSION GAS INJECTED INTO THE COOLANT CHANNEL. RESULTS PREDICTED BY THIS MODEL ARE FOUND TO BE CONSISTENT WITH THE TREAT M-0 EXPERIMENT. WHEN THE PARTICLE SIZE AND AMOUNT OF FUEL INJECTED ARE ADJUSTED TO MATCH THE INLET FLOW RATE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

REACTOR, LWR • ACCIDENT, CORE DISRUPTIVE • ANALYTICAL MODEL • OUT OF PILE EXPERIMENT • THERMAL HYDRAULIC ANALYSIS • FAILURE, FUEL ELEMENT • COMPARISON, THEORY AND EXPERIENCE • FLOW BLOCKAGE • NRC-7 • JACOBS

129072

JACKSON JP
NUCLEAR REACTOR SAFETY QUARTERLY PROGRESS REPORT JANUARY 1-MARCH 31, 1977
LOS ALAMOS SCIENTIFIC LAB., N.M.
LA-NUREG-8542-PR • 109 PPS, 27 TABS, 62 FIGS, 67 REFS, JUNE 1977

A NUMBER OF IMPORTANT ADVANCES WERE MADE AS PART OF THE LWR SAFETY RESEARCH PROGRAM. THE TRAC

120672 (CONTINUED)

CODE WAS IMPROVED BY ADDING ALL THE REMAINING MODULES NEEDED TO ANALYZE A LARGE BREAK LOCA IN A PWR. SIMMER-1 WAS APPLIED TO THE ANALYSIS OF SEVERAL PERTINENT LWR/SR SAFETY PROBLEMS. THESE INCLUDED TWO-DIMENSIONAL SODIUM VOIDING, EXTENSION OF THE CORE LOSS-OF-FLOW ACCIDENT INITIATION PHASE, AND VORTEX ENERGY PARTITION FOLLOWING DISASSEMBLY. IN THE NTRC SAFETY PROGRAM, THE REMAINING EQUIPMENT NEEDED FOR THE PWR HEAT-UP EXPERIMENT HAS BEEN SHIPPED. SEVERAL EXPERIMENTS WERE PERFORMED AS PART OF THE GCR CORE DISRUPTIVE TEST PROGRAM. IN THE CONTAINMENT EVALUATION AREA, A NEW CAPABILITY IN THE COMPARE CODE THAT ALLOWED TREATMENT OF ICE-CONDENSER CONTAINMENT DESIGNS WAS VERIFIED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, SAFETY RESEARCH • REACTOR, NTRC • REACTOR, LWR • REACTOR, LWR • REACTOR, LWR • ACCIDENT ANALYSIS • SAFETY ANALYSIS • REACTOR PHYSICS • ANALYTICAL MODEL • THERMAL HYDRAULIC ANALYSIS • JACOBS • NRC-6 • NRC-7 • NRC-8 • REACTOR KINETICS • REACTOR, GCR • CONTAINMENT, ICE CONDENSER

120676

JACKSON, JF • STEVENSON, MC
NUCLEAR REACTOR SAFETY QUARTERLY PROGRESS REPORT: APRIL 1-JUNE 30, 1977
LOS ALAMOS SCIENTIFIC LAB., N.M.
LA-5980-0034 • 100 PPS, 17 TABS, 70 FIGS, 80 REFS, AUG. 1977

RESEARCH PERFORMED AT LANS IS SUMMARIZED. FOUR AREAS ARE PRESENTED, INCLUDING NTRC, LWR, LWR, AND REACTOR TECHNOLOGY SAFETY RESEARCH. OTHER AREAS DISCUSSED ARE FISSION PRODUCT TECHNOLOGY, PRIMARY COOLANT IMPURITIES, STRUCTURAL EVALUATION, I AND C SYSTEM, ACCIDENT DELINEATION, AND PHENOMENA MODELING. OTHER RESEARCH AREAS INCLUDE EVALUATION OF TRAC CODE, THERMAL-HYDRAULICS, AND CONTAINMENT EVALUATION. OTHER RESEARCH CONSISTS OF PROGRAMS INVOLVING SIMMER CODE DEVELOPMENT, CORE DISRUPTIVE ACCIDENT ANALYSIS, SAFETY TEST FACILITY STUDIES, AND OUT-OF-PILE MODELING AND EXPERIMENTS. REACTOR TECHNOLOGY RESEARCH AREAS INCLUDE PROGRAMS CONCERNING THERMAL AND ACCIDENT ANALYSIS, REACTIVITY MEASUREMENTS, CRITICALITY INCIDENTS STUDY, AND WTR STUDY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, SAFETY RESEARCH • REACTOR, NTRC • REACTOR, LWR • REACTOR, LWR • REACTOR, LWR • ACCIDENT ANALYSIS • SAFETY ANALYSIS • REACTOR PHYSICS • ANALYTICAL MODEL • THERMAL HYDRAULIC ANALYSIS • JACOBS • NRC-6 • NRC-7 • NRC-8 • REACTOR KINETICS

127056

BUCHANAN, KERNHOLTZ, GR
LWR SAFETY 5. REVIEW OF CURRENT ISSUES AND BIBLIOGRAPHY OF LITERATURE, 1975-1976
OAK RIDGE NATIONAL LAB., TENN.
ORNL/RNRC/NSIC-120 (VOL. 5 OF ORNL/RNRC/NSIC-125) • 575 PPS, 2 TABS, 36 REFS, JULY 1977

DISCUSSES THE CURRENT STATUS OF LIQUID-METAL FAST BREEDER REACTOR (LWR) DEVELOPMENT AND ONE OF THE PRINCIPAL SAFETY ISSUES, A HYPOTHETICAL CORE-DISRUPTIVE ACCIDENT (MCOA). BIBLIOGRAPHIC INFORMATION ON WORLDWIDE LWRs RELATIVE TO THE DEVELOPMENT AND SAFETY OF THE BREEDER REACTOR IS PRESENTED FOR THE PERIOD 1975 THROUGH 1976. THE BIBLIOGRAPHY CONSISTS OF APPROXIMATELY 1610 ABSTRACTS COVERING EARLY RESEARCH AND DEVELOPMENT AND OPERATING EXPERIENCES LEADING UP TO THE PRESENT DESIGN PRACTICES THAT ARE NECESSARY FOR THE LICENSING OF BREEDER REACTORS. KEY-WORD, AUTHOR, AND PERMUTED-TITLE INDEXES ARE INCLUDED FOR COMPLETENESS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA. 22161
(613.75/FULL-SIZE, 93/PP)

REACTOR, LWR • PREVIEW • BIBLIOGRAPHY • SITING • REACTOR, BREEDER • REACTOR, FAST • R AND O PROGRAM • QUALITY ASSURANCE • OPERATING EXPERIENCE • DESIGN STUDY • ACCIDENT, CORE DISRUPTIVE • NRC-1 • NRC-7 • JACOBS

129191

CHU, HN • EGGEN, DT
SIMULATION OF THE CLADDING FREEZING DURING THE LOSS OF FLOW ACCIDENT IN A GAS-COOLED FAST BREEDER REACTOR
NORTHWESTERN UNIV., EVANSTON, ILL.
NUREG-0815 • 119 PPS, 46 FIGS, 67 REFS, JULY 1977

DURING A POSTULATED LOSS OF FLOW ACCIDENT IN A GCR CORE, STAINLESS STEEL CLADDING COULD MELT IN THE CENTRAL CORE REGION AND FLOW INTO THE LOWER BLANKET REGION WHERE IT COULD FREEZE. THIS REPORT CONSISTS OF BOTH EXPERIMENTAL AND ANALYTICAL WORK. THE OBJECTIVE OF THE EXPERIMENT IS TO SIMULATE THE MOTION OF LIQUID ALLOY THROUGH A TUBE AND OBSERVE THE CONDITIONS WHERE IT SOLIDIFIES IN THE TUBE. THE OBJECTIVE OF THE THEORETICAL MODEL IS TO USE TRANSIENT HEAT TRANSFER ANALYSIS TO DESCRIBE THE FREEZING FRONT OF THE MOVING LIQUID METAL AND THE PLUGGING OF THE CHANNEL. RESULTS INDICATE THAT THE STAINLESS STEEL WOULD FREEZE IN THE LOWER BLANKET REGION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ACCIDENT, LOSS OF FLOW • REACTOR, GCR • CORE MELTDOWN • FLOW BLOCKAGE • OUT OF PILE EXPERIMENT • COMPARISON, THEORY AND EXPERIENCE • THERMAL HYDRAULIC ANALYSIS • NRC-7 • JACOBS

129996

PONTANA, HN • GRESS, TS
LWR AEROSOL RELEASE AND TRANSPORT PROGRAM QUARTERLY PROGRESS REPORT FOR JANUARY-MARCH 1977

127990 (CONTINUED)

OAK RIDGE NATIONAL LAB., TENN.
ORNL/NUMEC/TM-113 +. 47 PPS, 8 TABS, 21 FIGS, 9 REFS, MARCH 1977

THIS PROGRAM IS DESIGNED TO INVESTIGATE RADIOACTIVE RELEASE AND TRANSPORT FROM LMFBRs FOR REACTOR EVENTS OF SEVERITY UP TO AND INCLUDING HYPOTHETICAL CORE-DISRUPTIVE ACCIDENTS (HCDAS). DISCUSSIONS INCLUDE (1) PROGRESS IN THE DEVELOPMENT OF THE ORNL CAPACITOR DISCHARGE VAPORIZATION (CDV) SYSTEM; (2) RESULTS OF EIGHT TESTS COMPLETED THIS QUARTER IN THE CRI-III CDV FACILITY; (3) RESULTS OF PRELIMINARY TESTS USING THE ARC HEARTH FURNACE FOR UO₂ VAPORIZATION IN CRI-III; (4) RESULTS OF THE FIRST SODIUM FIRE-AEROSOL TEST IN THE DESIGN AND CONSTRUCTION OF THE FUEL-AEROSOL SIMULANT TEST (FAST) FACILITY; AND (5) ANALYTICAL STUDIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

AEROSOL + AEROSOL PRODUCTION + AEROSOL, RADIOACTIVE + REACTOR, LMFBR + ACCIDENT, CORE DISRUPTIVE + CORE MELTDOWN + OUT OF PILE EXPERIMENT + NRC-7

127992

WRIGHT AL + SMITH AN + KRESS TS
FUEL AEROSOL SIMULANT TEST (FAST) PLAN
OAK RIDGE NATIONAL LAB., TENN.
ORNL/NUMEC/TM-129 +. 10 PPS, 4 TABS, 5 FIGS, 10 REFS, SE 7, 1977

THIS REPORT PRESENTS THE WORK PLAN FOR THE FUEL AEROSOL SIMULATION TEST (FAST), WHICH CONSTITUTES TASK 5 OF THE AEROSOL RELEASE AND TRANSPORT (ART) PROGRAM, SPONSORED BY THE DIVISION OF REACTOR SAFETY RESEARCH OF THE NUCLEAR REGULATORY COMMISSION. THESE TESTS INVOLVE THE STUDY OF THE FORMATION, RISE, AND CONDENSATION OF A SURROGATE LMFBR FUEL (UO₂) VAPOR BUBBLE PRODUCED IN LIQUID SODIUM TO SIMULATE PHENOMENA ASSOCIATED WITH HYPOTHETICAL CORE-DISRUPTIVE ACCIDENTS (HCDAS). UO₂ BUBBLES WILL BE PRODUCED BY THE CAPACITOR DISCHARGE VAPORIZATION (CDV) TECHNIQUE. TESTS THROUGH FY 1979 WILL BE DONE IN ARGON, WATER, AND IN SODIUM; RESULTS WILL BE ANALYZED USING MODELS DEVELOPED IN THE ART PROGRAM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

AEROSOL + SODIUM + REACTOR, LMFBR + BUBBLE + ACCIDENT, HYPOTHETICAL + ACCIDENT, CORE DISRUPTIVE + URANIUM DIOXIDE + ELECTRIC ARC + TRANSIENT + R AND D PROGRAM + FISSION PRODUCT RELEASE + PLUTONIUM + ARGON + NRC-7 + JACOBS

131441

BERTINI HW + KRESS TS + PARSLY LF + TOBIAS HL
SOURCE TERM ASSESSMENT FOR LMFBRs: AEROSOL RELEASE AND TRANSPORT (ART) ANALYTICAL PROGRAM
OAK RIDGE NATIONAL LAB., TENN.
ORNL/NUMEC/TM-124 +. 63 PPS, 2 TABS, 25 FIGS, 93 REFS, NOV. 1977

ANALYTICAL NEEDS FOR LMFBR HCDAS SOURCE TERM ASSESSMENT ARE OUTLINED AND THE PHYSICAL EVENTS CONNECTED WITH THE FORMATION OF AN AEROSOL SOURCE ARE DISCUSSED. THE IMPORTANT FEATURES OF THE MOVEMENT OF HAZARDOUS MATERIALS FROM THE CORE SITE TO THE BOUNDARIES OF THE SECONDARY CONTAINMENT ARE DESCRIBED, WITH ACCOMPANYING BRIEF REVIEWS OF PRIOR WORK. THE OVERALL PROGRAM IS BROKEN DOWN INTO TRACKABLE PARTS AND APPROACHES FOR ATTACKS ON THE VARIOUS PROBLEMS ARE SUGGESTED. TOPICS INCLUDED THE STATE OF THE DISRUPTED CORE, INTERACTIONS WITH INTACT ABOVE-CORE STRUCTURE, MEGA BUBBLE FORMATION, HEAT EXCHANGE WITH THE SODIUM POOL, AEROSOL FORMATION BY NUCLEATION AND FRAGMENTATION, ATTENUATION EFFECTS, AND AEROSOL BEHAVIOR IN SECONDARY CONTAINMENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

AEROSOL + ACCIDENT, CORE DISRUPTIVE + BUBBLE + ACCIDENT, HYPOTHETICAL + CONTAINMENT + REACTOR, LMFBR + CLINCH RIVER (LMFBR) + INTERFACE + FUEL COOLANT INTERACTION + CORE MELTDOWN + COMPUTER PROGRAM + TRANSPORT + NRC-7

132444

FONTANA HW + KRESS TS
BREEDER REACTOR AEROSOL RELEASE AND TRANSPORT PROGRAM QUARTERLY PROGRESS REPORT FOR APRIL-JUNE 1977
OAK RIDGE NATIONAL LAB., TENN.
ORNL/NUMEC/TM-162 +. 57 PPS, 10 TABS, 23 FIGS, 3 REFS, DEC. 1977

ROUTINE INVESTIGATIONS OF AIRBORNE FIELD AND PRIMARY AEROSOL SIZES VS INITIAL ENERGY STATE WERE CONTINUED IN THE CDV-CRI-III INSTALLATION UNTIL MID-JUNE. FUEL AEROSOL CHARACTERIZATION STUDIES IN THE CRI-III FACILITY CONTINUED THROUGH THE QUARTER WITH EMPHASIS ON INCREASING THE TOTAL AEROSOL CONCENTRATION USING THE ARC HEARTH FURNACE GENERATOR. THE NSPP SODIUM POOL FIRE EXPERIMENTS CONTINUED, WITH TWO ADDITIONAL EXPERIMENTS BEING CONDUCTED THIS QUARTER. PELLET VAPORIZER ASSEMBLIES FOR USE IN UNDER-WATER AND UNDER-SODIUM EXPERIMENTS WERE TESTED IN ARGON AT THE CRI-III FACILITY THIS QUARTER.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + REACTOR, LMFBR + AEROSOL + FISSION PRODUCT RELEASE + FISSION PRODUCT TRANSPORT + SODIUM + FIRE + NSPP + ACCIDENT, CORE DISRUPTIVE + OUT OF PILE EXPERIMENT + NRC-7

127336

ANNUAL CORE PULSE REACTOR UPGRADE QUARTERLY REPORT, OCTOBER-DECEMBER 1976

PAGE -64

127336 (CONTINUED)

SANDIA LABS., ALBUQUERQUE, N.M.

SAND-77-0083 • NUREG-0120-1 • 71 PPS, 11 TABS, 20 FIGS, MARCH 1977

PROGRESS OF NINE TASKS FOR THE ACRU UPGRADE PROJECT IS PRESENTED. THE OBJECT OF THE ACRU UPGRADE IS TO ARRIVE AT A REACTOR MODIFICATION PROVIDING INCREASED PULSED NEUTRON FLUENCE IN THE IRRADIATION CAVITY WITHOUT INCREASED DEGRADATION OF THE PULSE DURATION. THE APPROACH INVOLVES A TWO-REGION CORE CONCEPT. THE INNER REGION, SURROUNDING THE IRRADIATION CAVITY, CONTAINS HIGH-HEAT-CAPACITY FUEL AND THE OUTER REGION CONTAINS URANIUM-ZIRCONIUM FUEL SIMILAR TO ACRU FUEL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS • REACTOR, PULSED • PULSED NEUTRON TECHNIQUE • MODIFICATION • DESIGN STUDY • FUEL, NUCLEAR • REACTOR CONTROL • SAFETY ANALYSIS • NRC-7

127337

ANNUAL CORE PULSE REACTOR UPGRADE QUARTERLY REPORT, JANUARY-MARCH 1977

SANDIA LABS., ALBUQUERQUE, N.M.

SAND-77-0110 • NUREG-0120-2 • 63 PPS, 5 TABS, 10 FIGS, 12 REFS, JUNE 1977

PROGRESS OF NINE TASKS FOR THE ACRU UPGRADE PROJECT IS PRESENTED. THE OBJECT OF THE ACRU UPGRADE IS TO ARRIVE AT A REACTOR MODIFICATION PROVIDING INCREASED PULSED NEUTRON FLUENCE IN THE IRRADIATION CAVITY WITHOUT INCREASED DEGRADATION OF THE PULSE DURATION. THE APPROACH INVOLVES A TWO-REGION CORE CONCEPT. THE INNER REGION, SURROUNDING THE IRRADIATION CAVITY, CONTAINS HIGH-HEAT-CAPACITY FUEL AND THE OUTER REGION CONTAINS URANIUM-ZIRCONIUM FUEL SIMILAR TO ACRU FUEL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS • REACTOR, PULSED • PULSED NEUTRON TECHNIQUE • MODIFICATION • DESIGN STUDY • FUEL, NUCLEAR • REACTOR CONTROL • SAFETY ANALYSIS • NRC-7

128498

MURPHY 00

A PRELIMINARY MODEL FOR CORE/CONCRETE INTERACTIONS

SANDIA LABS., ALBUQUERQUE, N.M.

SAND-77-0270 • 48 PPS, 17 FIGS, 15 REFS, AUG. 1977

A PRELIMINARY MODEL IS DESCRIBED FOR COMPUTING THE RATE OF PENETRATION OF CONCRETE BY A MOLTEN LWR CORE. AMONG THE PHENOMENA INCLUDED ARE CONVECTIVE STIRRING OF THE MELT BY EVOLVED GASES, ABREATHURE OF CONCRETE DECOMPOSITION PRODUCTS TO THE MELT, CHEMICAL REACTIONS, RADIATIVE HEAT LOSS, AND VARIATION OF HEAT TRANSFER COEFFICIENTS WITH LOCAL PRESSURE. THE MODEL IS MOST APPLICABLE TO A TWO-PHASE MELT (METALLIC PLUS OXIDIC) HAVING A FAIRLY HIGH METALLIC CONTENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ANALYTICAL MODEL • CORE • CONCRETE • CORE MELTDOWN • REACTOR, LWR • JACOBS • NRC-3 • NRC-7

129097

FAST REACTOR SAFETY RESEARCH PROGRAM QUARTERLY REPORT JANUARY-MARCH 1977

SANDIA LABS., ALBUQUERQUE, N.M.

SAND-77-0111 • NUREG-0121-2 • 147 PPS, 5 TABS, 60 FIGS, 20 REFS, AUG. 1977

SANDIA LABORATORIES FAST REACTOR SAFETY RESEARCH PROGRAM, IS A COMPREHENSIVE RESEARCH ACTIVITY CONDUCTED ON BEHALF OF NRC AND IS PART OF NRC'S CONFIRMATORY RESEARCH EFFORT TO ASSURE THAT THE NECESSARY SAFETY DATA AND THEORETICAL UNDERSTANDING EXIST TO ADEQUATELY LICENSE AND REGULATE THE LIQUID METAL FAST BREEDER REACTOR (LMFBR). THE PROGRAM REPORTED ON ARE: ACCIDENT ENERGISTICS, CORE DEBRIS BEHAVIOR, SODIUM CONTAINMENT AND STRUCTURAL INTEGRITY, RESEARCH FOR ELEVATED TEMPERATURE DESIGN CRITERIA, FUEL MOTION DETECTION, AND ACRU FUEL MOTION DETECTION SYSTEM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

PREACTOR, LMFBR • OR AND B PROGRAM • ACCIDENT ANALYSIS • ACCIDENT, CORE DISRUPTIVE • SODIUM • CONTAINMENT • STRUCTURAL INTEGRITY • HIGH TEMPERATURE • DESIGN CRITERIA • CORE • MONITOR • NRC-7 • JACOBS

130067

ANNUAL CORE PULSE REACTOR UPGRADE QUARTERLY REPORT APRIL-JUNE 1977

SANDIA LABS., ALBUQUERQUE, N.M.

SAND-77-1133 • NUREG-0120-3 • 35 PPS, 3 TABS, 5 FIGS, 10 REFS, SEPT. 1977

PROGRESS OF NINE TASKS FOR THE ACRU UPGRADE PROJECT IS PRESENTED. THE OBJECT OF THE ACRU UPGRADE IS TO ARRIVE AT A REACTOR MODIFICATION PROVIDING INCREASED PULSED NEUTRON FLUENCE IN THE IRRADIATION CAVITY WITHOUT INCREASED DEGRADATION OF THE PULSE DURATION. THE APPROACH INVOLVES A TWO-REGION CORE CONCEPT. THE INNER REGION, SURROUNDING THE IRRADIATION CAVITY, CONTAINS HIGH-HEAT-CAPACITY FUEL AND THE OUTER REGION CONTAINS URANIUM-ZIRCONIUM FUEL SIMILAR TO ACRU FUEL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS • REACTOR, PULSED • PULSED NEUTRON TECHNIQUE • MODIFICATION • DESIGN STUDY • FUEL, NUCLEAR • REACTOR CONTROL • SAFETY ANALYSIS • NRC-7

132368
 ADVANCED REACTOR SAFETY RESEARCH PROGRAM QUARTERLY REPORT APRIL - JUNE 1977
 SANDIA LABS., ALBUQUERQUE, N.M.
 SAND-77-1134 • NUREG-0181-3 • 207 PPS, 14 TABS, 62 FIGS, 47 REFS, NOV. 1977

THE PROGRAM INCLUDES A BROAD RANGE OF EXPERIMENTS TO SIMULATE ACCIDENTAL TRANSIENT CONDITIONS IN THE LFBFR TO PROVIDE THE REQUIRED DATA BASE TO UNDERSTAND THE CONTROLLING ACCIDENT SEQUENCES AND TO SERVE AS A BASIS FOR VERIFICATION OF THE COMPLEX COMPUTER SIMULATION MODELS AND CODES USED IN ACCIDENT ANALYSIS AND LICENSING REVIEWS. IT IS INVESTIGATING SEVEN MAJOR AREAS OF INTEREST AS FOLLOWS: (A) ACCIDENT ENERGETICS, (B) CORE DEBRIS BEHAVIOR, (C) SODIUM CONTAINMENT AND STRUCTURAL INTEGRITY, (D) RESEARCH OF ELEVATED TEMPERATURE DESIGN CRITERIA, (E) FUEL MOTION, (F) ACPH FUEL REACTION SYSTEM, AND (G) ADVANCED REACTOR SAFETY RESEARCH ASSESSMENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, LFBFR • ACCIDENT ANALYSIS • REACTOR, FAST • ACCIDENT, CORE DISRUPTIVE • ENCLOSURE, LARGE • SODIUM • MELTEN FUEL • NRC-7

130653
 GARDNER PL • ENOMAN CA • REYNOLDS AB
 CONDENSATION OF FUEL ONTO THE ABOVE-CORE STRUCTURE DURING AN LFBFR CORE-DISRUPTIVE ACCIDENT
 UNIV. OF VA., CHARLOTTESVILLE
 NUREG-0304 • 141 PPS, 8 TABS, 16 FIGS, 51 REFS, OCT. 1977

CONDENSATION OF A PURE, SATURATED VAPOR ONTO A VERTICAL, MELTING SUBSTRATE IS ANALYZED FOR BOTH ONE- AND TWO-MATERIAL SITUATIONS. THE RESULTS OF THIS ANALYSIS HAVE APPLICATIONS IN THE AREA OF LFBFR ACCIDENT ANALYSIS. CONDENSATION OF THE FUEL (A MIXTURE OF URANIUM DIOXIDE AND PLUTONIUM DIOXIDE) ONTO THE ABOVE-CORE STRUCTURE (STEEL) IS A MECHANISM FOR REDUCING THE ENERGY AVAILABLE TO MECHANICALLY DAMAGE THE PRIMARY SYSTEM BOUNDARY. CONDENSATION WILL ALSO ALTER THE RADIOACTIVE AEROSOL SOURCE TERM AVAILABLE FOR POTENTIAL RELEASE FROM THE PRIMARY SYSTEM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, LFBFR • ACCIDENT ANALYSIS • ACCIDENT, CORE DISRUPTIVE • CONDENSATION • VAPOR EXCHANGE • CONTAINMENT INTEGRITY • MAIN COOLING SYSTEM • FUEL ROD • AEROSOL, RADIOACTIVE • FISSION PRODUCT TRANSPORT • JACOBS • NRC-7

131819
 KENNEDY HP • ENOMAN CA • REYNOLDS AB • WALTER AE
 PRIMARY AEROSOL PARTICLE-SIZE DISTRIBUTION FROM HOMOGENEOUS NUCLEATION CONDENSATION AND PARTICLE GROWTH
 UNIV. OF VA., CHARLOTTESVILLE
 NUREG-0301 • UNV/529100/NEEP 77/182 • 89 PPS, 1 TAB, 14 FIGS, 49 REFS, DEC. 1977

THE PROCESS OF FORMING AEROSOLS BY HOMOGENEOUS NUCLEATION CONDENSATION FOLLOWED BY CONDENSATION GROWTH WAS STUDIED AS A POTENTIAL SOURCE OF SUBMICROMETER AEROSOLS IN LFBFR ACCIDENTS. A MODEL WAS DEVELOPED FOR CALCULATING PARTICLE SIZE DISTRIBUTIONS FOLLOWING NUCLEATION AND GROWTH. THIS MODEL WAS APPLIED TO EXPERIMENTS AT ORNL WHERE UO₂ FUEL PELLETS WERE PARTIALLY VAPORIZED IN AN ARGON ATMOSPHERE AND THE RESULTING PRIMARY PARTICLE SIZE DISTRIBUTION WAS MEASURED. THE LOW RANGE OF PARTICLE SIZES OBSERVED IN THE ORNL TESTS COULD BE REPRODUCED BY THE MODEL BY ASSUMING LARGE MIXING RATES BETWEEN THE FUEL VAPOR AND THE ARGON, THUS INDICATING THAT HOMOGENEOUS NUCLEATION AND CONDENSATION WAS LIKELY THE SOURCE OF THE SMALL PARTICLES OBSERVED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, LFBFR • AEROSOL • CONDENSATION • MODEL • PARTICLE SIZE DISTRIBUTION • URANIUM DIOXIDE • NRC-7

6. 02 - ADVANCED REACTOR SAFETY RESEARCH, GAS-COOLED REACTORS

127005

SCHWEITZER 05

HYPER SAFETY EVALUATION DIVISION QUARTERLY REPORT JULY-SEPTEMBER 1976
BROOKHAVEN NATIONAL LAB., UPTON, N.Y.
BNL-NUREG-50046 P. 147 PPS, MARCH 1977

SEVERAL EXPERIMENTS ON THE RELEASE OF CESIUM FROM MOIST GRAPHITE IN VACUO AND IN FLOWING HELIUM WERE COMPLETED. THE FIRST PERMEABILITY OF IODINE THROUGH CONCRETE EXPERIMENT WAS BEGUN. X-RAY EMISSION SPECTROSCOPY UTILIZING AN X-RAY TUBE GENERATOR SOURCE AND A Si(Li) DETECTOR WAS SHOWN TO BE APPROPRIATE FOR DETERMINING IMPURITIES IN GRAPHITE SAMPLES. AN ANALYSIS FOR IMPURITY COMPONENTS WITH Z>13 IS EASILY OBTAINED. A NEW ROSSMANER SPECTROMETER WAS PURCHASED AND IS UNDERGOING TESTS. IT IS A DIAGNOSTIC INSTRUMENT THAT WILL AID IN IDENTIFYING THE EFFECT OF IODINE AND ITS COMPOUNDS ON GRAPHITE OXIDATION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

JACOBS • REACTOR, HYPER • CESIUM • IODINE • CONCRETE • HELIUM • SAFETY ANALYSIS • ACOUSTICS • FATIGUE • COMPUTER PROGRAM • COOLANT CHEMISTRY • SAFETY EVALUATION • NRC-6 • GRAPHITE • GRAPHITE

127075

RENNER AJ

REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT, JANUARY 1-MARCH 31, 1977
BROOKHAVEN NATIONAL LAB., UPTON, N.Y.
BNL-NUREG-50061 P. 216 PPS, MAY 1977

THE PROJECTS REPORTED EACH QUARTER ARE THE FOLLOWING: GAS REACTOR SAFETY EVALUATION, THOR CODE DEVELOPMENT, SSC CODE DEVELOPMENT, LWR AND LWR SAFETY EXPERIMENTS, FAST REACTOR SAFETY CODE VALIDATION, TECHNICAL COORDINATION OF STRUCTURAL INTEGRITY, AND FAST REACTOR SAFETY RELIABILITY ASSESSMENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

REACTOR, LWR • REACTOR, HYPER • REACTOR, LWR • COMPUTER PROGRAM, DIGITAL • CODES AND STANDARDS • ACCIDENT, CORE DISRUPTIVE • JACOBS • SAFETY ANALYSIS • ACCIDENT, CONSEQUENCES • NRC-4 • NRC-7 • NRC-8

127100

RENNER AJ

REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT APRIL 1-JUNE 30, 1977
BROOKHAVEN NATIONAL LAB., UPTON, N.Y.
BNL-NUREG-50063 P. 193 PPS, TABS, FIGS, JULY 1977

THE REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT, DESCRIBES CURRENT ACTIVITIES AND TECHNICAL PROGRESS IN THE PROGRAMS AT BROOKHAVEN NATIONAL LABORATORY SPONSORED BY THE USNRC REACTOR SAFETY RESEARCH DIVISION. THE PROJECTS REPORTED EACH QUARTER ARE THE FOLLOWING: GAS REACTOR SAFETY EVALUATION, THOR CODE DEVELOPMENT, SSC CODE DEVELOPMENT, LWR AND LWR SAFETY EXPERIMENTS, FAST REACTOR SAFETY CODE VALIDATION, TECHNICAL COORDINATION OF STRUCTURAL INTEGRITY, AND FAST REACTOR SAFETY RELIABILITY ASSESSMENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

REACTOR, LWR • REACTOR, LWR • REACTOR, HYPER • SAFETY ANALYSIS • COMPUTER PROGRAM, DIGITAL • CODES AND STANDARDS • ACCIDENT, CORE DISRUPTIVE • CORE MELTDOWN • ACCIDENT, LOSS OF COOLANT • SHUTDOWN • ACCIDENT ANALYSIS • BOILING • NRC-4 • NRC-7 • NRC-8

127000

PROCEEDINGS OF THE JAPAN-U.S. SEMINAR ON HYPER SAFETY TECHNOLOGY. SEISMIC RESEARCH, VOLUME 1
U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON
BNL-NUREG-50060(VOL.1) P. 320 PPS, MEETING HELD AT UPTON, N.Y., SEPT. 15-16, 1977

FOUR TECHNICAL SESSIONS WERE HELD WITH A TOTAL OF 24 PAPERS BEING PRESENTED BY JAPANESE AND AMERICAN RESEARCHERS. ALL PAPERS CONCERNED SEISMIC EVENTS, ACCELERATION, OR VIBRATION USUALLY AS IT WOULD AFFECT THE HYPER REACTOR SYSTEM.

AVAILABILITY - CONTACT BROOKHAVEN NATIONAL LAB., TECHNICAL REPORTS DISTRIBUTION, UPTON, N.Y. 11973 FOR AVAILABILITY INFO

JACOBS • EARTHQUAKE • SEISMOLOGY • REACTOR, HYPER • R AND D PROGRAM • SAFETY PROGRAM • NRC-8

127001

PROCEEDINGS OF THE JAPAN-U.S. SEMINAR ON HYPER SAFETY TECHNOLOGY HELIUM TECHNOLOGY VOLUME 11
U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON
BNL-NUREG-50060(VOL.11) P. 297 PPS, MEETING HELD AT UPTON, N.Y., SEPT. 15-16, 1977

TWENTY-THREE PAPERS PRESENTED. SOME OF THE TOPICS WERE: HELIUM TECHNOLOGY, HIGH TEMPERATURE HEAT TRANSFER, SAFETY STUDY ON WTR THERMAL STRUCTURE, LAYERING, HIGH TEMPERATURE INSULATION, SURFACE AREA-SURFACE CORRELATION FOR THE STEAM GRAPHITE REACTION, HIGH TEMPERATURE CORROSION, CREEP OF STRUCTURAL ALLOYS, FISSION PRODUCT BEHAVIOR AND EFFECT OF IMPURITIES IN THE HELIUM.

120951 *CONTINUED*

AVAILABILITY - CONTACT BROOKHAVEN NATIONAL LAB., TECHNICAL REPORTS DISTRIBUTION, UPTON, N.Y. 11973 FOR AVAILABILITY INFO

JACOBS • HELIUM • REACTOR, HTGR • HIGH TEMPERATURE • CORROSION • R AND D PROGRAM • HEAT TRANSFER • FISSION PRODUCT RELEASE • FISSION PRODUCT TRANSPORT • JAY OF PILE EXPERIMENT • ALLOY • GRAPHITE • DEPOSITION • PEACH BOTTOM 1 (HTGR) • INSULATION • NRC-B

120472

JACKSON JP
NUCLEAR REACTOR SAFETY QUARTERLY PROGRESS REPORT JANUARY 1-MARCH 31, 1977
LOS ALAMOS SCIENTIFIC LAB., N.M.
LA-NUREG-0047-PR 1, 109 PPS, 27 TABS, 52 FIGS, 67 REFS, JUNE 1977

A NUMBER OF IMPORTANT ADVANCES WERE MADE AS PART OF THE LWR SAFETY RESEARCH PROGRAM. THE TRAC CODE WAS IMPROVED BY ADDING ALL THE REMAINING MODULES NEEDED TO ANALYZE A LARGE BREAK LOCA IN A PWR. SIMMER-1 WAS APPLIED TO THE ANALYSIS OF SEVERAL PERTINENT LWR SAFETY PROBLEMS. THESE INCLUDED TWO-DIMENSIONAL SODIUM VOIDING, EXTENSION OF THE CORE LOSS-OF-FLOW ACCIDENT INITIATION PHASE, AND WORK ENERGY PARTITION FOLLOWING DISASSEMBLY. IN THE HTGR SAFETY PROGRAM, THE REMAINING EQUIPMENT NEEDED FOR THE FUEL HEAT-UP EXPERIMENT HAS BEEN SHIPPED. SEVERAL EXPERIMENTS WERE PERFORMED AS PART OF THE GCFR CORE DISRUPTIVE TEST PROGRAM. IN THE CONTAINMENT EVALUATION AREA, A NEW CAPABILITY IN THE COMPARE CODE THAT ALLOWED TREATMENT OF ICE-CONDENSED CONTAINMENT DESIGNS WAS VERIFIED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, SAFETY RESEARCH • REACTOR, HTGR • REACTOR, LWR • REACTOR, LWR • ACCIDENT ANALYSIS • SAFETY ANALYSIS • REACTOR PHYSICS • ANALYTICAL MODEL • THERMAL HYDRAULIC ANALYSIS • JACOBS • NRC-A • NRC-7 • NRC-B • REACTOR KINETICS • REACTOR, GCFR • CONTAINMENT, ICE CONDENSER

120970

JACKSON JP • STEVENSON MC
NUCLEAR REACTOR SAFETY QUARTERLY PROGRESS REPORT: APRIL 1-JUNE 30, 1977
LOS ALAMOS SCIENTIFIC LAB., N.M.
LA-NUREG-0034 • 105 PPS, 17 TABS, 70 FIGS, 80 REFS, AUG. 1977

RESEARCH PERFORMED AT LASL IS SUMMARIZED. FOUR AREAS ARE PRESENTED, INCLUDING HTGR, LWR, LWRB, AND REACTOR TECHNOLOGY SAFETY RESEARCH. RESEARCH AREAS DISCUSSED ARE FISSION PRODUCT TECHNOLOGY, PRIMARY COOLANT IMPURITIES, STRUCTURAL EVALUATION, I AND C SYSTEM, ACCIDENT DELINEATION, AND PHENOMENA MODELING. RESEARCH AREAS INCLUDE EVALUATION OF TRAC CODE, THERMAL-HYDRAULICS, AND CONTAINMENT EVALUATION. RESEARCH CONSISTS OF PROGRAMS INVOLVING SIMMER CODE DEVELOPMENT, CORE DISRUPTIVE ACCIDENT ANALYSIS, SAFETY TEST FACILITY STUDIES, AND OUT-OF-PILE MODELING AND EXPERIMENTS. REACTOR TECHNOLOGY RESEARCH AREAS INCLUDE PROGRAMS CONCERNING THERMAL AND ACCIDENT ANALYSIS, REACTIVITY MEASUREMENTS, CRITICALITY INCIDENTS STUDY, AND VHTA STUDY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, SAFETY RESEARCH • REACTOR, HTGR • REACTOR, LWRB • REACTOR, LWR • ACCIDENT ANALYSIS • SAFETY ANALYSIS • REACTOR PHYSICS • ANALYTICAL MODEL • THERMAL HYDRAULIC ANALYSIS • JACOBS • NRC-A • NRC-7 • NRC-B • REACTOR KINETICS

120620

BALL SJ • CLEVELAND JC • CONKLIN JC • SANDERS JP
HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY PROGRESS REPORT JANUARY 1, 1977-MARCH 31, 1977
OAK RIDGE NATIONAL LAB., TENN.
ORNL/NUREG/TM-115 • 17 PPS, 2 FIGS, 6 REFS, JUNE 1977

WORK DONE DURING THE QUARTER WAS MAINLY ON IMPROVING THE BALANCE-OF-PLANT SUBROUTINES TO RUN MORE EFFICIENTLY AND RELIABLY IN THE LOW-POWER AND PLANT SHUTDOWN MODES. SOME OF THE INFORMATION ACQUIRED ON A TRIP TO THE PORT ST. WAIRN REACTOR WAS FACTORED INTO THE CTGR.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, HTGR • SAFETY ANALYSIS • COMPUTER PROGRAM • TURBINE • FLOW STABILITY • PT, ST, WAIRN (HTGR) • NRC-B • JACOBS

120907

CLEVELAND JC • HEDRICH RA • BALL SJ • DELEWE JC
ORTAP: A NUCLEAR STEAM SUPPLY SYSTEM SIMULATION FOR THE DYNAMIC ANALYSIS OF HIGH TEMPERATURE GAS COOLED REACTOR TRANSIENTS
OAK RIDGE NATIONAL LAB., TENN.
ORNL/NUREG/TM-70 • 85 PPS, 1 TAB, 20 FIGS, 13 REFS, SEPT. 1977

APPROACH WAS TO BUILD SUFFICIENT DETAIL INTO COMPONENT MODELS SO THAT COUPLING BETWEEN THE PRIMARY AND SECONDARY SYSTEMS CAN BE ACCURATELY REPRESENTED. SYSTEM COMPONENTS MODELED IN ORTAP INCLUDE REACTOR CORE, REHEATER AND STEAM GENERATOR, HELIUM CIRCULATION AND CIRCULATOR TURBINE, TURBINE GENERATOR PLANT AND MAJOR PLANT CONTROL SYSTEMS. TRANSIENTS WHICH CAN BE ANALYZED WITH ORTAP INCLUDE REACTOR START-UP AND SHUTDOWN, NORMAL AND RAPID LOAD CHANGES, REACTOR TRIP, TURBINE TRIP

PAGE -48

129997 *CONTINUED*

AND SUDDEN REDUCTION IN DEWATER FLOW. ORTAP CAN PREDICT PLANT RESPONSE TO EMERGENCY OR FAILED CONDITIONS SUCH AS PRIMARY SYSTEM DEPRESSURIZATION, LOSS OF PRIMARY COOLANT FLOW AND UNCONTROLLED REMOVAL OF CONTROL POISON FROM CORE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM. DIGITAL * REACTOR, HYD * REACTOR TRANSIENT * REACTOR DYNAMICS * STEAM * ACCIDENT, STEAM LINE RUPTURE * ACCIDENT ANALYSIS * ACCIDENT MODEL * ACCIDENT, TRANSIENT OVERPOWER * NRC-8

129993

GALL SJ * CLEVELAND JC * COMELIN JC * SPANERS JP

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY PROGRESS REPORT, APRIL 1-JUNE 30, 1977

ORNL/NUREG/TM-138

ORNL/NUREG/TM-138 * 27 PAGES, SEPT. 1977

THE STUDIES CONCENTRATED ON DETAILED INVESTIGATIONS OF POSTULATED FORT ST. VRAIN ROD-WITHDRAWAL ACCIDENTS. GENERAL ATOMIC COMPANY'S TAP PROGRAM WAS IMPLEMENTED ON THE LOCAL IBM 360 COMPUTERS, AND THE RESULTS OF SAMPLE PROBLEM RUNS APPEAR TO BE IN REASONABLE AGREEMENT WITH THOSE PUBLISHED BY GENERAL ATOMIC.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

OR AND D PROGRAM * SAFETY PROGRAM * REACTOR, HYD * SAFETY ANALYSIS * COMPUTER PROGRAM * TURBINE * FLOW STABILITY * FT. ST. VRAIN (HYD) * JACOBS * NRC-8

9. RM - HEALTH SAFETY RESEARCH

127625
 H00089 BT
 NUCLEAR DECAY DATA FOR RADIONUCLIDES OCCURRING IN ROUTINE RELEASES FROM NUCLEAR FUEL CYCLE FACILITIES
 OAK RIDGE NATIONAL LAB., TENN.
 ORNL/TN-102 +. 114 PPS. 0 REFS. AUG. 1977

TABULATES THE ATOMIC AND NUCLEAR RADIATIONS EMITTED BY 246 RADIONUCLIDES. FOR EACH RADIONUCLIDE
 ARE GIVEN THE HALF-LIFE AND RECOMMENDED VALUES FOR THE ENERGIES, INTENSITIES AND EQUILIBRIUM
 ABSORBED-DOSE CONSTANTS FOR EACH OF THE ATOMIC AND NUCLEAR RADIATIONS. ALSO GIVEN ARE THE
 DAUGHTER RADIONUCLIDES PRODUCED AND RECOMMENDED VALUES FOR DECAY BRANCHING RATIOS, WHERE
 APPLICABLE. THE RADIOACTIVE DECAY CHAINS AND BRANCHING RATIOS ARE DISPLAYED IN DIAGRAM FORM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

EFFLUENT + RADIOACTIVITY RELEASE + AIRBORNE RELEASE + DOSE + ENERGY + RADIOISOTOPE + ALPHA EMITTER + BETA
 EMITTER + GAMMA EMITTER + NRC-9

10. RE - ENVIRONMENTAL RESEARCH

120047

MURAKA IP • FERRANTE JC • POLICASTRO AJ • DANIELS ED
AN EVALUATION OF ENVIRONMENTAL DATA RELATING TO SELECTED NUCLEAR POWER PLANT SITES. THE KEWAUNEE NUCLEAR
POWER PLANT SITE
ARGONNE NATIONAL LAB., ILL.
ANL/EIS-1 •. 243 PPS, TABS, FIGS, AUG. 1976

ENVIRONMENTAL MONITORING DATA FOR THE YEARS 1973, 1974 AND 1975 PERTAINING TO THE KEWAUNEE PLANT,
WHICH BEGAN OPERATION IN MID-1974, WERE ANALYZED BY THE MOST PRACTICAL QUALITATIVE AND
QUANTITATIVE METHODS. THE RESULTS SHOWED NO SIGNIFICANT IMMEDIATE DELETERIOUS EFFECTS, THUS
CONFIRMING PREOPERATIONAL PREDICTIONS. ALTHOUGH THE PLANT HAS NOT OPERATED LONG ENOUGH TO REVEAL
LONG-TERM DELETERIOUS EFFECTS, THE PRESENT INDICATIONS DO NOT LEAD TO A CONCERNED PREDICTION THAT
ANY ARE DEVELOPING. THE CUMULATIVE EFFECTS OF MULTIPLE LAKEHORE PLANTS WERE NOT CONSIDERED IN
THIS EFFORT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

R AND D PROGRAM • SAFETY PROGRAM • MONITORING PROGRAM, ENVIRONMENTAL • SITE, EFFECT • REACTOR, PWR • KEWAUNEE
(PWR) • DATA COLLECTION • ECOSYSTEM, AQUATIC • BIOTA • NRC-10

120048

FERRANTE JC • MURAKA IP • VASLOU F
AN EVALUATION OF ENVIRONMENTAL DATA RELATING TO SELECTED NUCLEAR POWER PLANT SITES. THE QUAD-CITIES NUCLEAR
POWER STATION SITE
ARGONNE NATIONAL LAB., ILL.
ANL/EIS-2 •. 99 PPS, 16 TABS, 18 FIGS, AUG. 1976

ENVIRONMENTAL MONITORING DATA FOR THE YEARS 1972 THROUGH 1975 PERTAINING TO THE QUAD-CITIES
STATION, WHICH BEGAN COMMERCIAL OPERATION FEB. 10, 1973, WERE ANALYZED BY THE MOST PRACTICAL
QUALITATIVE AND QUANTITATIVE METHODS. FOUR DIFFERENT COOLING METHODS THAT WERE USED ARE
DESCRIBED. THE RESULTS SHOWED NO SIGNIFICANT IMMEDIATE DELETERIOUS EFFECTS OR VIOLATIONS OF
STATE WATER QUALITY STANDARDS, THUS CONFIRMING PREOPERATIONAL PREDICTIONS. ALTHOUGH THE PLANT
HAS NOT OPERATED LONG ENOUGH TO REVEAL LONG-TERM DELETERIOUS EFFECTS, THE PRESENT INDICATIONS DO
NOT LEAD TO A CONCERNED PREDICTION THAT ANY ARE DEVELOPING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

R AND D PROGRAM • SAFETY PROGRAM • MONITORING PROGRAM, ENVIRONMENTAL • SITE, EFFECT • REACTOR, BWR • QUAD
CITIES 1 (BWR) • QUAD CITIES 2 (BWR) • DATA COLLECTION • ECOSYSTEM, AQUATIC • BIOTA • NRC-10

120049

VASLOU F • MURAKA IP
AN EVALUATION OF ENVIRONMENTAL DATA RELATING TO SELECTED NUCLEAR POWER PLANT SITES. THE DUANE ARNOLD ENERGY
CENTER SITE
ARGONNE NATIONAL LAB., ILL.
ANL/EIS-3 •. 10 PPS, 1 FIG, AUG. 1976

ANALYSES WERE MADE ON 1974-1975 DATA PERTAINING TO WATER QUALITY AND THE USE OF BIOCIDES AT THE
DUANE ARNOLD ENERGY CENTER, WHICH BEGAN OPERATING IN FEBRUARY 1975. THE ANALYSES SHOWED NO
IMMEDIATE DELETERIOUS EFFECTS DUE TO PLANT OPERATIONS. ALTHOUGH THE PLANT HAS NOT BEEN IN
OPERATION LONG ENOUGH TO REVEAL ANY LONG-TERM DELETERIOUS EFFECTS, PRESENT INDICATIONS DID NOT
LEAD TO A CONCERNED PREDICTION THAT ANY WERE DEVELOPING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

R AND D PROGRAM • SAFETY PROGRAM • MONITORING PROGRAM, ENVIRONMENTAL • SITE, EFFECT • REACTOR, BWR • ARNOLD
(BWR) • DATA COLLECTION • ECOSYSTEM, AQUATIC • BIOTA • NRC-10

120050

MURAKA IP
AN EVALUATION OF ENVIRONMENTAL DATA RELATING TO SELECTED NUCLEAR POWER PLANT SITES
ARGONNE NATIONAL LAB., ILL.
ANL/EIS-4 •. 13 PPS, 3 FIGS, 1 REP, AUG. 1976

ENVIRONMENTAL MONITORING DATA FOR THE YEARS 1973 AND 1974 PERTAINING TO THE THREE HILE ISLAND
NUCLEAR STATION UNIT 1, WHICH BEGAN OPERATION IN EARLY 1974, WERE ANALYZED BY THE MOST PRACTICAL
QUALITATIVE AND QUANTITATIVE METHODS. TERRESTRIAL BIOLOGICAL RESOURCES WERE CONSIDERED FOR THIS
PLANT. THE EFFECTS OF THE OPERATION OF UNIT 1 ON THE LOCAL TERRESTRIAL ORGANISMS WERE FOUND TO
BE UNDETECTABLE. ALTHOUGH THE PLANT HAS NOT OPERATED LONG ENOUGH TO REVEAL LONG-TERM DELETERIOUS
EFFECTS, THE PRESENT INDICATIONS DO NOT LEAD TO A CONCERNED PREDICTION THAT ANY ARE DEVELOPING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

ENVIRONMENT • THREE HILE ISLAND 1 (PWR) • MONITOR, ENVIRONMENTAL • EFFECT • BIOTA • SOIL • NRC-10

120051

MURAKA IP • POLICASTRO A • DANIELS E
AN EVALUATION OF ENVIRONMENTAL DATA RELATING TO SELECTED NUCLEAR POWER PLANT SITES

120040 (CONTINUED)
 ARGONNE NATIONAL LAB., ILL.
 ANL/RES-5 P. 305 PPS, TABS, FIGS, REFS, NOV. 1976

ENVIRONMENTAL MONITORING DATA FOR THE YEARS 1972 THROUGH 1975 PERTAINING TO THE ZION STATION, WHICH BEGAN COMMERCIAL OPERATION IN 1970, WERE ANALYZED BY THE MOST PRACTICAL QUALITATIVE AND QUANTITATIVE METHODS. THERMAL PLUME, WATER CHEMISTRY AND AQUATIC BIOTA DATA WERE EVALUATED AND THE RESULTS ARE PRESENTED IN THIS REPORT. THE RESULTS SHOWED NO SIGNIFICANT IMMEDIATE DELETERIOUS EFFECTS, THUS CONFIRMING PREDICTIONAL PREDICTIONS. ALTHOUGH THE PLANT HAS NOT OPERATED LONG ENOUGH TO REVEAL LONG-TERM DELETERIOUS EFFECTS, THE PRESENT INDICATIONS DO NOT LEAD TO A CONCERNED PREDICTION THAT ANY ARE DEVELOPING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

ENVIRONMENT + MONITOR, ENVIRONMENTAL + ZION 1 (PWR) + PLUME BEHAVIOR + AQUATIC ORGANISM + ECOSYSTEM, AQUATIC + WATER QUALITY + EFFECT + NRC-10

120060
 MURAKA IP
 EVALUATION OF ENVIRONMENTAL DATA RELATING TO SELECTED NUCLEAR POWER PLANT SITES - THE PRAIRIE ISLAND NUCLEAR PLANT SITE
 ARGONNE NATIONAL LAB., ILL.
 ANL/RES-6 P. 114 PPS, 11 TABS, 10 FIGS, 5 REFS, NOV. 1976

ENVIRONMENTAL MONITORING DATA FOR 1973-1975 PERTAINING TO THE PRAIRIE ISLAND NUCLEAR GENERATING STATION (WHICH BEGAN COMMERCIAL OPERATION IN DECEMBER 1973) WERE ANALYZED. EVALUATIONS OF AQUATIC AND TERRESTRIAL BIOTIC DATA ARE PRESENTED. THE DATA INDICATE NO SIGNIFICANT IMMEDIATE DELETERIOUS EFFECTS ON THE BIOTA FROM PLANT OPERATION, THUS CONFIRMING PREOPERATIONAL PREDICTIONS. ALTHOUGH THE STATION HAS NOT OPERATED LONG ENOUGH TO REVEAL LONG-TERM DELETERIOUS EFFECTS, PRESENT INDICATIONS DO NOT LEAD TO A CONCERNED PREDICTION THAT ANY ARE DEVELOPING. RECOMMENDATIONS ARE SUGGESTED FOR IMPROVING MONITORING TECHNIQUES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

MONITORING PROGRAM, ENVIRONMENTAL + DATA COLLECTION + PRAIRIE ISLAND 1 (PWR) + REACTOR, PWR + ECOSYSTEM, AQUATIC + BIOTA + SITE, EFFECT + NRC-8

132050
 MURAKA IP
 AN EVALUATION OF ENVIRONMENTAL DATA RELATING TO SELECTED NUCLEAR POWER PLANT SITES. THE NINE MILE POINT NUCLEAR POWER STATION SITE UNIT 1
 ARGONNE NATIONAL LAB., ILL.
 ANL/RES-7 P. 41 PPS, 3 TABS, 17 FIGS, OCT. 1976

FISH IMPINGEMENT MONITORING DATA FOR THE YEARS 1974 AND 1975 PERTAINING TO THE NINE MILE POINT UNIT ONE, WHICH BEGAN COMMERCIAL OPERATION IN DECEMBER 1969, WERE ANALYZED BY THE MOST PRACTICAL QUALITATIVE AND QUANTITATIVE METHODS. THE MATHEMATICAL MODELS USED IN DESCRIBING THE FISH KILL RATES SHOWED A MARKED IMPROVEMENT OVER THE PREVIOUS ASSUMPTIONS THAT FISH KILLS (BY SPECIES) CAN BE CONSIDERED AS RANDOM EVENTS INDEPENDENT OVER TIME. HOWEVER, WITH MORE ACCURATE ESTIMATES ON FISH KILLS NOW AVAILABLE, IT IS STILL NOT POSSIBLE TO DETERMINE THE SIGNIFICANCE OF THESE FISH KILLS. THE RESULTS SHOWED THAT PREDICTIONS WERE HIGHER THAN THOSE OBSERVED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

R AND D PROGRAM + SAFETY PROGRAM + MONITORING PROGRAM, ENVIRONMENTAL + SITE, EFFECT + REACTOR, BWR + NINE MILE POINT 1 (BWR) + ANIMAL, FISH + MORTALITY + DATA COLLECTION + ECOSYSTEM, AQUATIC + BIOTA + NRC-10

120700
 MURAKA IP + POLICASTRO AJ + FERRANTE JC
 AN EVALUATION OF ENVIRONMENTAL DATA RELATING TO SELECTED NUCLEAR POWER PLANT SITES
 ARGONNE NATIONAL LAB., ILL.
 ANL/RES-8 P. 30 PPS, 2 FIGS, 2 TABS, NOV. 1976

ANALYSIS OF THE FIELD DATA GATHERED DURING THE MONITORING PROGRAMS AT SEVEN NUCLEAR POWER PLANT SITES SHOW NO SIGNIFICANT ECOLOGICAL IMPACTS. THIS CONCLUSION IS WITHIN THE CONSTRAINTS OF THE QUALITY OF AVAILABLE DATA FROM THESE SITES. CURRENT MONITORING PROGRAMS ARE, HOWEVER, NOT DESIGNED TO MEET THE NEEDS FOR STATISTICAL ANALYSIS AND, CONSEQUENTLY, THE MONITORING DATA ARE OFTEN ILL-SUITED FOR MODERN STATISTICAL PROCEDURES. RECOMMENDATIONS ARE PROPOSED FOR REVISING MONITORING SCHEMES SO THAT MORE PRECISE CONCLUSIONS CAN BE MADE FROM FEWER FIELD MEASUREMENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

ENVIRONMENT + POWER PLANT, NUCLEAR + MONITOR, ENVIRONMENTAL + STATISTICAL ANALYSIS + COMPARISON + REACTOR (PWR) + QUAD CITIES 1 (BWR) + ARNOLD (BWR) + THREE MILE ISLAND 1 (PWR) + ZION 1 (PWR) + PRAIRIE ISLAND 1 (PWR) + NINE MILE POINT 1 (BWR)

132020
 ADAMS SM + CUNNINGHAM PA + GRAY DO
 A CRITICAL EVALUATION OF THE RADIOBIOLOGICAL ENVIRONMENTAL TECHNICAL SPECIFICATIONS. VOLUME 1 - PROGRAM

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130430 (CONTINUED)
DESCRIPTION, SUMMARY, AND RECOMMENDATIONS
OAK RIDGE NATIONAL LAB., TENN.
ORNL/NUREG/TA-691VOL.11 00. 50 PPS, 7 TABS, 11 REFS, JUNE 1977

A COMPREHENSIVE STUDY OF THE DATA COLLECTED AS PART OF THE ENVIRONMENTAL TECHNICAL SPECIFICATIONS PROGRAM FOR EIGHT NUCLEAR POWER PLANTS WAS CONDUCTED. THIS REPORT INCLUDES A SUMMARY OF THE SCREENING PHASE IN WHICH THE ADEQUACY OF THE HYDROTHERMAL AND ECOLOGICAL MONITORING DATA FOR EACH PLANT WERE EVALUATED, AND THE SUMMARY AND RECOMMENDATIONS RESULTING FROM A DETAILED EXAMINATION OF THE THREE NUCLEAR POWER PLANTS (SURRY, PEACH BOTTOM AND SAN ONOFRE) SELECTED IN THE INITIAL SCREENING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + ENVIRONMENT + TECHNICAL SPECIFICATIONS, ENVIRONMENTAL + POWER PLANT, NUCLEAR + THERMAL POLLUTION + ECOSYSTEM + MONITOR, ENVIRONMENTAL + PREVIEW + TEST, OPERATIONAL + BIOLOGICAL CONTROL

122540
ADAMS BN + CUNNINGHAM PA + GRAY OD + KURAR KO
A CRITICAL EVALUATION OF THE NONRADIOLOGICAL ENVIRONMENTAL TECH SPECS. VOL. 2 - SURRY POWER PLANT UNITS 1 AND 2
OAK RIDGE NATIONAL LABORATORY, TENNESSEE
ORNL/NUREG/TA-701VOL.2 OF ORNL/NUREG/TA-691 00. 269 PPS, 25 TABS, 90 FIGS, FEB. 1977

A COMPREHENSIVE STUDY OF THE DATA COLLECTED AS PART OF THE ENVIRONMENTAL TECH SPECS PROGRAM FOR UNITS 1 AND 2 OF THE SURRY NUCLEAR POWER PLANT WAS CARRIED OUT. THE PROGRAM INCLUDED AN ANALYSIS OF THE HYDROTHERMAL AND ECOLOGICAL MONITORING DATA COLLECTED FROM 1973 THROUGH 1975. THE HYDROTHERMAL ANALYSIS INCLUDES A DISCUSSION OF MODELS USED IN PLUME PREDICTIONS PRIOR TO PLANT OPERATION AND AN EVALUATION OF THE PRESENT HYDROTHERMAL MONITORING PROGRAM. THE ECOLOGICAL ANALYSIS INCLUDES VALIDATION OF IMPACTS PREDICTED IN THE FINAL ENVIRONMENTAL STATEMENT USING THE OPERATIONAL MONITORING DATA.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

SURRY 1 (PWR) + SURRY 2 (PWR) + TECHNICAL SPECIFICATIONS, ENVIRONMENTAL + DATA COLLECTION + MONITORING PROGRAM, ENVIRONMENTAL + THERMAL ANALYSIS + THERMAL POLLUTION + PLUME BEHAVIOR + ECOLOGY + HYDROLOGY + METEOROLOGY + NRC-8

129290
ADAMS BN + CUNNINGHAM PA + GRAY OD
A CRITICAL EVALUATION OF THE NONRADIOLOGICAL ENVIRONMENTAL TECHNICAL SPECIFICATIONS VOL. 3 PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3
OAK RIDGE NATIONAL LAB., TENN.
ORNL/NUREG/TA-711VOL.3 OF ORNL/NUREG/TA-691 00. 176 PPS, 26 TABS, 44 FIGS, APRIL 1977

A COMPREHENSIVE STUDY OF DATA COLLECTED AS PART OF THE ENVIRONMENTAL TECH SPECS PROGRAM FOR PEACH BOTTOM 2 AND 3 WAS CONDUCTED FOR NRC. THE PROGRAM INCLUDED AN ANALYSIS OF BOTH THE HYDROTHERMAL AND ECOLOGICAL MONITORING DATA COLLECTED FROM 1967-1976. THE HYDROTHERMAL ANALYSIS INCLUDES A DISCUSSION OF MODELS USED IN PLUME PREDICTIONS PRIOR TO PLANT OPERATION AND AN EVALUATION OF THE PRESENT HYDROTHERMAL MONITORING PROGRAM. THE ECOLOGICAL ANALYSIS INCLUDES VALIDATION OF IMPACTS PREDICTED IN THE FINAL ENVIRONMENTAL STATEMENT USING THE OPERATIONAL MONITORING DATA. SPECIFIC RECOMMENDATIONS ARE MADE FOR IMPROVING BOTH THE PRESENT HYDROTHERMAL AND ECOLOGICAL MONITORING PROGRAMS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

PEACH BOTTOM 2 (BWR) + PEACH BOTTOM 3 (BWR) + REACTOR, BWR + TECHNICAL SPECIFICATIONS, ENVIRONMENTAL + DATA COLLECTION + HYDROLOGY + THERMAL ANALYSIS + ANALYTICAL MODEL + PLUME BEHAVIOR + ECOLOGY + MONITORING PROGRAM, ENVIRONMENTAL + NRC-10

126790
BRADLAN AM + AKIN EJ + BARTON JM + BLEDSOE JL
DEVELOPMENT OF A UNIFIED TRANSPORT APPROACH FOR THE ASSESSMENT OF POWER-PLANT IMPACT
OAK RIDGE NATIONAL LAB., TENN.
ORNL/NUREG/TA-69 00. 47 PPS, 12 FIGS, 36 REFS, APRIL 1977

PROGRESS DURING THE FIRST 18 MONTHS IN IMPLEMENTATION OF THE UNIFIED TRANSPORT APPROACH (UTA) IS SUMMARIZED IN THIS REPORT, WHICH COVERS THE PERIOD THROUGH DEC. 1976. THE GOAL OF THIS PROJECT IS TO DEVELOP MATHEMATICAL MODELS FOR FAST-TRANSIENT, ONE- AND TWO-DIMENSIONAL TRANSPORT OF THERMAL, RADIOLOGICAL, CHEMICAL, AND BIOLOGICAL PROPERTIES IN RIVERS, ESTUARIES, LAKES, AND COASTAL REGIONS FOR ASSESSING THE IMPACT OF POWER-PLANT OPERATIONS. THE UTA PROVIDES A COMMON BASIS FOR CALCULATING THE TRANSPORT OF INTENSIVE QUANTITIES THAT DEPEND ON BASIC FLOW PROPERTIES, WHICH CAN BE OBTAINED FROM A COMMON SET OF DATA FOR GEOMETRY, BATHYMETRY, AND METEOROLOGY THAT MUST BE PREPARED ONLY ONCE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

TRANSPORT + ANALYTICAL MODEL + TRANSIENT + THERMAL ANALYSIS + RADIOLOGICAL IMPACT + BIOLOGICAL CONTROL + CHEMICAL ANALYSIS + RIVER + ESTUARY + LAKE + SITE, EFFECT + FLOW DISTRIBUTION + NRC-10 + POWER PLANT, NUCLEAR

11. RT - TRANSPORTATION SAFETY RESEARCH

133210

MACRAUSON CP

SHOCK AND VIBRATION ENVIRONMENTS FOR LARGE SHIPPING CONTAINER DURING TRUCK TRANSPORT (PART 1)

SANDIA LABS. ALBUQUERQUE, N.M.

SAND-77-1110 P. 23 PPS. 2 TABS. 6 FIGS. 1 REF. SEPT. 1977

CURRENTLY AVAILABLE DATA WERE TAKEN ON TRUCKS BEARING LIGHTER LOADS THAN THE LOADS OF CURRENT INTEREST. IN ADDITION, THE NEW DATA ARE EXPECTED TO BE USEFUL IN THE DETERMINATION OF ANY TRENDS OF VIBRATION AND SHOCK ENVIRONMENTS WITH INCREASED CARGO WEIGHT. THESE NEW DATA WERE OBTAINED ON A "DIGGTRAC" BASIS DURING TRUCK TRANSPORT OF 195 700 N (44,000 LB) CARGO WHICH CONSISTED OF A SPENT FUEL CONTAINER AND ITS SUPPORTING STRUCTURE FROM MERCURY, NEVADA, TO ALBUQUERQUE, NEW MEXICO.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM • SAFETY PROGRAM • SHIPPING CONTAINER • TRUCK • IMPACT SHOCK • VIBRATION • NRC-12

12. NS - SAFEGUARDS RESEARCH

132518

BRUMBACH SO + PERRY BO
AUTORADIOGRAPHIC TECHNIQUE FOR RAPID INVENTORY OF PLUTONIUM-CONTAINING FAST CRITICAL ASSEMBLY FUEL
ARGONNE NATIONAL LAB., ILL.
ANL-77-67 +. 20 PPS, 9 FIGS, 4 REFS, OCT, 1977

A NONDESTRUCTIVE AUTORADIOGRAPHIC TECHNIQUE IS DESCRIBED WHICH CAN PROVIDE A VERIFICATION OF THE PIECE COUNT AND THE PLUTONIUM CONTENT OF PLUTONIUM-CONTAINING FUEL ELEMENTS. THIS TECHNIQUE USES THE SPONTANEOUSLY EMITTED GAMMA RAYS FROM PLUTONIUM TO FORM IMAGES OF FUEL ELEMENTS ON PHOTOGRAPHIC FILM. AUTORADIOGRAPHY HAS THE ADVANTAGE OF PROVIDING AN INVENTORY VERIFICATION WITHOUT THE OPENING OF CONTAINERS OR THE HANDLING OF FUEL ELEMENTS. MISSING FUEL ELEMENTS, SUBSTITUTION OF NONRADIOACTIVE MATERIAL, AND SUBSTITUTION OF ELEMENTS OF DIFFERENT SIZE ARE DETECTABLE. RESULTS ARE PRESENTED FOR FUEL ELEMENTS IN VARIOUS STORAGE CONFIGURATIONS AND FOR FUEL ELEMENTS CONTAINED IN A FAST CRITICAL ASSEMBLY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

MEASUREMENT + PLUTONIUM + RADIOGRAPHY + TEST, NONDESTRUCTIVE + FUEL STORAGE + SAFEGUARDS, NUCLEAR MATERIAL + FUEL ELEMENTS

131947

COLE RJ + BENNETT CA + EDELHEITZ M + WOOD HT + BROWN PJ + ROBERTS PP
STRUCTURE AND DRAFTING OF SAFEGUARDS REGULATORY DOCUMENTS
BATTILLE PACIFIC NORTHWEST LABS., RICHLAND, WASH.
NUREG-037123, SUMMARY, VOL. 1 AND 2 + BNWL-2488 +. 236 PPS, FIGS, REFS, NOV, 1977

DEVELOPED HYPOTHESES ON THE RELATIONSHIP BETWEEN THE STRUCTURE AND DRAFTING OF SAFEGUARDS REGULATORY DOCUMENTS AND THE ABILITY OF THE DOCUMENT USERS TO UNDERSTAND AND IMPLEMENT THEM IN A WAY THAT REFLECTS THE INTENT AND REQUIREMENTS OF THE NRC. LICENSING OFFICES, LICENSEES, INSPECTORS, AND THE GENERAL PUBLIC MUST UNDERSTAND THE NRC'S REQUIREMENTS IF THE REGULATORY SYSTEM IS TO FUNCTION EFFECTIVELY AND IN COMPLIANCE WITH LEGAL REQUIREMENTS. A SERIES OF FOUR DECISIONS THAT WILL BE REQUIRED TO IMPROVE COMMUNICATIONS WITH LICENSERS WERE SET FORTH.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + SAFEGUARDS, NUCLEAR MATERIAL + AGENCY, NRC + PROCEDURES AND MANUALS + COMMUNICATION + NRC-13

127334

ENGI 0
A SMALL-SCALE ENGAGEMENT MODEL WITH ARRIVALS: ANALYTICAL SOLUTIONS
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-8054 + NUREG-0238 +. 27 PPS, 2 TABS, 7 FIGS, APRIL 1977

THIS REPORT PRESENTS AN ANALYTICAL MODEL OF SMALL-SCALE BATTLES. THE IMPETUS WAS PROVIDED BY A NEED TO CHARACTERIZE HYPOTHETICAL BATTLES BETWEEN GUARDS AT A NUCLEAR FACILITY AND THEIR POTENTIAL ADVERSARIES. THE SOLUTION PROCEDURE CAN BE USED TO FIND MEASURES OF A NUMBER OF CRITICAL PARAMETERS: FOR EXAMPLE, THE WIN PROBABILITIES AND THE EXPECTED DURATION OF THE BATTLE. NUMERICAL SOLUTIONS ARE OBTAINABLE IF THE TOTAL NUMBER OF INDIVIDUAL COMBATANTS ON THE OPPOSING SIDES IS LESS THAN 10. FOR SMALLER FORCE SIZE BATTLES, WITH ONE OR TWO COMBATANTS ON EACH SIDE, SYMBOLIC SOLUTIONS CAN BE FOUND.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

SABOTAGE + SAFEGUARDS, NUCLEAR MATERIAL + SYSTEM ANALYSIS + COMPUTER PROGRAM + MODEL + SENSITIVITY ANALYSIS + NRC-13 + THEFT/DIVERSION + JACOBS

132811

ENGI 0
NUCLEAR FACILITY SAFEGUARDS SYSTEMS MODELING USING DISCRETE EVENT SIMULATION
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-77-0075 + CONF-770430-2 +. 5 PPS, FROM 8TH ANNUAL CONFERENCE ON MODELING & SIMULATION, PITTSBURGH, PA., APRIL 21, 1977

THE THREAT OF THEFT OR DISPERSAL OF SPECIAL NUCLEAR MATERIAL AT A NUCLEAR FACILITY IS TREATED BY STUDYING THE TEMPORAL RELATIONSHIPS BETWEEN ADVERSARIES HAVING AUTHORIZED ACCESS TO THE FACILITY (INSIDERS) AND SAFEGUARDS SYSTEM EVENTS BY USING A GASP IV DISCRETE EVENT SIMULATION. THE SAFEGUARDS SYSTEM EVENTS--DETECTION, ASSESSMENT, DELAY, COMMUNICATIONS, AND NEUTRALIZATION--ARE MODELED FOR THE GENERAL INSIDER ADVERSARY STRATEGY, WHICH INCLUDES DEGRADATION OF THE SAFEGUARDS SYSTEM ELEMENTS FOLLOWED BY AN ATTEMPT TO STEAL OR DISPERSE SPECIAL NUCLEAR MATERIAL. THE PERFORMANCE MEASURE USED IN THE ANALYSIS IS THE ESTIMATED PROBABILITY OF SAFEGUARDS SYSTEM SUCCESS IN COUNTERING THE ADVERSARY BASED UPON A PREDETERMINED SET OF ADVERSARY ACTIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + SAFEGUARDS, NUCLEAR MATERIAL + THEFT/DIVERSION + MODEL + SIMULATION + COMPUTER PROGRAM + NRC-13

129486

BUSAN VL + CHAPMAN LD
A STRUCTURE FOR THE DECOMPOSITION OF SAFEGUARDS RESPONSIBILITIES
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-77-0680 + CONF-770642-2 +. 20 PPS. FROM 3RD INTERNATIONAL CONFERENCE ON ION BEAM ANALYSIS; WASHINGTON, D.C., JUNE 27, 1977

A MAJOR MISSION OF SAFEGUARDS IS TO PROTECT AGAINST THE USE OF NUCLEAR MATERIALS BY ADVERSARIES TO MAN SOCIETY. A HIERARCHICAL STRUCTURE OF SAFEGUARDS RESPONSIBILITIES AND ACTIVITIES TO ASSIST IN THIS MISSION IS DEFINED. THE STRUCTURE BEGINS WITH THE DEFINITION OF INTERNATIONAL OR MULTI-NATIONAL SAFEGUARDS AND CONTINUES THROUGH DOMESTIC, REGIONAL, AND FACILITY SAFEGUARDS. THE FACILITY SAFEGUARDS IS DECOMPOSED INTO PHYSICAL PROTECTION AND MATERIAL CONTROL RESPONSIBILITIES. IN ADDITION, IN-TRANSIT SAFEGUARDS SYSTEMS ARE CONSIDERED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

OR AND D PROGRAM + SAFETY PROGRAM + *SAFEGUARDS, NUCLEAR MATERIAL + TRANSPORTATION AND HANDLING + THEFT/DIVERSION + NRC-13 + JACOBS

129481

KETON SC + DE LAQUIL P
CONFLICT SIMULATION FOR SURFACE TRANSPORT SYSTEMS
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-77-0624 +. 23 PPS. PRESENTED AT 18TH ANNUAL MEETING OF THE INSTITUTE FOR NUCLEAR MATERIALS MANAGEMENT; WASHINGTON, D.C., JUNE 28-30, 1977

AN IMPORTANT ELEMENT IN THE ANALYSIS OF TRANSPORTATION SAFEGUARDS SYSTEMS IS THE DETERMINATION OF THE OUTCOME OF AN ARMED ATTACK AGAINST THE SYSTEM. SUCH INFORMATION IS NECESSARY TO UNDERSTAND RELATIONSHIPS AMONG THE VARIOUS DEFENDER TACTICS, WEAPONS SYSTEMS, AND ADVERSARY ATTRIBUTES. A BATTLE MODEL, SABRES, WHICH CAN SIMULATE SAFEGUARDS ENGAGEMENTS IS UNDER DEVELOPMENT. THIS PAPER BRIEFLY DESCRIBES THE FIRST PHASE OF SABRES AND PRESENTS SOME EXAMPLES OF ITS CAPABILITIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

OR AND D PROGRAM + SAFETY PROGRAM + *TRANSPORTATION AND HANDLING + *SAFEGUARDS, NUCLEAR MATERIAL + COMPUTER PROGRAM + THEFT/DIVERSION + TRUCK + NRC-13 + JACOBS

129484

GALLAGHER RJ + STIMRELL KG + WAGNER MR
THE CONFIGURATION OF ROAD CONVOYS: A SIMULATION STUDY
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-77-0625 +. 23 PPS. FROM 18TH ANNUAL MEETING OF THE INSTITUTE FOR NUCLEAR MATERIALS MANAGEMENT; WASHINGTON, D.C., JUNE 28-30, 1977

AN IMPORTANT ELEMENT IN THE EVALUATION OF TRANSPORTATION SAFEGUARDS SYSTEMS IS THE ANALYSIS OF CONVOY CONFIGURATIONS. THE ANALYSIS SHOULD EXAMINE THE INFLUENCE OF VEHICLE DEPLOYMENT AND TACTICS, COMMUNICATIONS STRATEGIES, VULNERABILITIES, AND ADVERSARY CHARACTERISTICS ON SURVIVABILITY AND EMERGENCY SIGNAL PROBABILITY. A COMPUTERIZED MODEL, SOURCE, HAS BEEN DEVELOPED WHICH SIMULATES THE INITIAL INTERACTION BETWEEN A CONVOY AND AN ADVERSARY FORCE. THIS PAPER BRIEFLY DESCRIBES THE MODEL AND PRESENTS EXAMPLE RESULTS FOR SEVERAL VEHICLE CONVOY CONFIGURATIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

TRANSPORTATION AND HANDLING + SAFEGUARDS, NUCLEAR MATERIAL + *TRUCK + ANALYTICAL MODEL + COMPUTER PROGRAM + *THEFT/DIVERSION + NRC-13 + JACOBS

129489

BERGIER KP
ESTIMATING THE AVAILABILITY OF LLEA OFFICERS
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-77-0626 +. 21 PPS. PRESENTED AT 18TH ANNUAL MEETING OF THE INSTITUTE FOR NUCLEAR MATERIALS MANAGEMENT; WASHINGTON, D.C., JUNE 28-30, 1977

AN IMPORTANT ELEMENT IN THE ANALYSIS OF TRANSPORTATION SAFEGUARDS SYSTEMS IS THE DETERMINATION OF THE AVAILABILITY OF LOCAL LAW ENFORCEMENT AGENTS. SUCH INFORMATION IS NECESSARY TO IDENTIFY AREAS WHERE THERE ARE FEW OFFICERS AVAILABLE AND TO MAKE COMPARISONS OF ALTERNATIVE ROUTES. A COMPUTERIZED MODEL, COPS, HAS BEEN DEVELOPED WHICH RAPIDLY ESTIMATES THE TOTAL NUMBER OF OFFICERS ALONG A HIGHWAY ROUTE. THIS PAPER BRIEFLY DESCRIBES THE MODEL AND PRESENTS FRAMPLE RESULTS FOR SEVERAL ROUTES IN CALIFORNIA AND NEVADA.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

OR AND D PROGRAM + SAFETY PROGRAM + *SAFEGUARDS, NUCLEAR MATERIAL + COMPUTER PROGRAM + TRUCK + NRC-13 + JACOBS + *TRANSPORTATION AND HANDLING

131946

ENGLE D + BOOZER DO

PAGE -06

131946 *CONTINUED*

THE USE OF ISEN IN STUDYING THE IMPACT OF GUARD TACTICS ON FACILITY SAFEGUARDS SYSTEM EFFECTIVENESS
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-77-0818C +. 14 PPS, 2 TABS, 1 FIG, 4 REFS, JULY 1977

THE INSIDER SAFEGUARDS EFFECTIVENESS MODEL (ISEM) IS A STOCHASTIC, DISCRETE EVENT, MONTE-CARLO SIMULATION MODEL USED TO ASSESS THE EFFECTIVENESS OF PHYSICAL PROTECTION SYSTEMS FOR FACILITIES WHICH STORE, PROCESS, OR USE SNM. ISEM SIMULATES THE INTERACTION OF A GROUP OF INSIDERS (ADVERSARIES WHO ARE GUARDS OR OTHER EMPLOYEES HAVING AUTHORIZED ACCESS TO THE FACILITY) WITH THE FACILITY'S SAFEGUARDS SYSTEM. THE SENSITIVITY OF THE SAFEGUARDS SYSTEM EFFECTIVENESS TO A VARIETY OF GUARD TACTICS IS EXPLORED IN THIS PAPER. THE EVOLUTION OF COMPREHENSIVE GUARD TACTICS FOR PROTECTING A HYPOTHETICAL FACILITY IS DEMONSTRATED. ATTENTION IS FOCUSED ON THE POTENTIAL THREAT POSED BY INSIDERS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + MODEL + HUMAN FACTORS + MONTE CARLO + NRC-13 + SAFEGUARDS, NUCLEAR MATERIAL + THEFT/DIVERSION

132408

CHAPMAN LB
PHYSICAL PROTECTION OF NUCLEAR FACILITIES QUARTERLY PROGRESS REPORT JANUARY-MARCH 1977
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-77-0999 +. 31 PPS, 2 TABS, 1 FIG, JULY 1977

THE PHYSICAL PROTECTION OF NUCLEAR FACILITIES PROGRAM CONSISTS OF FOUR MAJOR AREAS--EVALUATION METHODOLOGY DEVELOPMENT, PATH GENERATION/SELECTION METHODOLOGY, FACILITY CHARACTERIZATION, AND COMPONENT FUNCTIONAL PERFORMANCE CHARACTERIZATION. ACTIVITIES IN EACH OF THESE AREAS ARE SUMMARIZED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

SAFEGUARDS, NUCLEAR MATERIAL + THEFT/DIVERSION + SABOTAGE + R AND D PROGRAM + SAFETY PROGRAM + SAFETY EVALUATION + NRC-13 + JACOBS

132814

MULLEN BL + HOLBRIDGE DB
SPTH3: A SUBROUTINE FOR FINDING SHORTEST SABOTAGE PATHS
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-77-1000 +. 33 PPS, 1 TAB, 5 FIGS, 6 REFS, JULY 1977

THIS DOCUMENT EXPLAINS HOW TO CONSTRUCT A SABOTAGE GRAPH WHICH MODELS ANY FIXED-SITE FACILITY AND HOW TO USE THE SUBROUTINE SPTH3 TO FIND SHORTEST PATHS IN THE GRAPH. THE SHORTEST SABOTAGE PATHS REPRESENT PHYSICAL ROUTES THROUGH THE SITE WHICH WOULD ALLOW AN ADVERSARY TO TAKE ADVANTAGE OF THE GREATEST WEAKNESSES IN THE SYSTEM OF BARRIERS AND ALARMS. THE SUBROUTINE SPTH3 IS A TOOL WITH WHICH SAFEGUARDS DESIGNERS AND ANALYSTS CAN STUDY THE RELATIVE EFFECTS OF DESIGN CHANGES ON THE ADVERSARY ROUTING PROBLEM. IN ADDITION TO SHOWING HOW TO USE SPTH3, THIS REPORT DISCUSSES THE METHODS USED TO FIND SHORTEST PATHS AND SEVERAL IMPLEMENTATION DETAILS WHICH CAUSE SPTH3 TO BE EXTREMELY EFFICIENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

SABOTAGE + COMPUTER PROGRAM + PROBABILITY + SECURITY + NRC-13 + JACOBS

132886

GINNE RL
THE EVALUATION OF SAFEGUARDS SYSTEMS FOR NUCLEAR MATERIALS IN TRANSIT - THE DEVELOPMENT OF THE PROGRAM PLAN
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-77-0840 +. 33 PPS, 13 FIGS, JULY 1977

THIS REPORT IS BASED UPON PRESENTATIONS GIVEN TO THE NRC ADMINISTRATION AND STAFF AND TO SANDIA LABORATORIES, LIVERMORE AND ALBUQUERQUE, STAFF IN FEBRUARY 1977. THE PURPOSE OF THESE PRESENTATIONS WAS TO DESCRIBE THE PROGRAM PLAN FOR THE PHYSICAL PROTECTION OF NUCLEAR MATERIAL IN TRANSIT AND TO PRESENT HIGHLIGHTS OF THE CURRENT STATUS FROM SEVERAL OF THE SECURITY STUDY AREAS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

TRANSPORTATION AND HANDLING + SAFEGUARDS, NUCLEAR MATERIAL + SYSTEM ANALYSIS + PROTECTION SYSTEM + SECURITY + NRC-13 + JACOBS

132888

JACOBSON SE
OPTIMAL IMPROVEMENT OF GRAPHS RELATED TO NUCLEAR SAFEGUARDS PROBLEMS
SANDIA LABS., ALBUQUERQUE, N.M.
SAND-78-0456 +. 20 PPS, 1 TAB, 1 FIG, 3 REFS, OCT. 1977

DEVELOPS THE METHODOLOGY FOR OPTIMALLY IMPROVING GRAPHS RELATED TO NUCLEAR SAFEGUARDS ISSUES. IN PARTICULAR, GIVEN A FIXED NUMBER OF DOLLARS, THE REPORT PROVIDES A METHOD FOR OPTIMALLY ALLOCATING SUCH DOLLARS OVER THE ARCS OF A WEIGHTED GRAPH (THE WEIGHTS VARY AS A FUNCTION OF

132432 *CONTINUED*

DOLLARS SPENT ON ARCS) SO AS TO IMPROVE THE SYSTEM EFFECTIVENESS MEASURE WHICH IS THE SHORTEST OF ALL SHORTEST PATHS TO SEVERAL TARGETS. ARC HEIGHTS CAN BE EITHER CLOCK TIMES OR DETECTION PROBABILITIES AND THE ALGORITHM DOES NOT EXPLICITLY CONSIDER ALL PATHS TO THE TARGETS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

A AND U PROGRAM * SAFETY PROGRAM * *SAFEGUARDS, NUCLEAR MATERIAL * MODEL * OPTIMIZATION * PROBABILITY * NRC-13

129381

MULME UL * HOLDRIDGE DB

KSPN: A SUBROUTINE FOR THE K SHORTEST PATHS IN A SABOTAGE GRAPH

SANDIA LABS., ALBUQUERQUE, N.M.

SAND-77-1165 P. 26 PPS, AUG. 1977

FINDING SHORTEST PATHS IN A WEIGHTED GRAPH MODEL IS ONE WAY FOR A SAFEGUARDS ANALYST TO LOCATE WEAKNESSES IN A FACILITY'S BARRIER AND ALARM SYSTEM. KSPN CAN BE USED TO RANK SABOTAGE PATHS ACCORDING TO PATH LENGTH SO THAT THE K SHORTEST PATHS CAN BE STUDIED BY MORE DETAILED METHODS TO SEE WHICH POSSESS ADDITIONAL PROPERTIES ATTRACTIVE TO AN ADVERSARY. WHILE EMPHASIZING HOW TO USE KSPN, THIS REPORT EXPLAINS THE K-TH SHORTEST PATH ALGORITHM OF HOFFMAN, PAULY AND GREYFUS AND CONTAINS A SAMPLE PROBLEM, TEST RESULTS AND PROGRAM LISTINGS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*SABOTAGE * SECURITY * COMPUTER PROGRAM * THEFT/DIVERSION * JACOBS * NRC-13

128395

BAUMER WA

A SURVEY OF THREAT STUDIES RELATED TO THE NUCLEAR POWER INDUSTRY

SANDIA LABS., ALBUQUERQUE, N.M.

SAND-77-9254 P. 47 PPS, 2 TABS, 5 FIGS, 27 REFS, AUG. 1977

A CONSIDERABLE EFFORT HAS BEEN DIRECTED TOWARD THE DETERMINATION OF THREAT CHARACTERISTICS, RESULTING IN A VOLUMINOUS COLLECTION OF DOCUMENTS. THIS REPORT SUMMARIZES SEVERAL OF THE MAJOR STUDIES IN ORDER TO MAKE THE INFORMATION MORE ACCESSIBLE. THIS SUMMARY INCLUDES ONLY STUDIES INVOLVING ATTACKS ON NUCLEAR MATERIAL, PLUS THOSE INCIDENTS WHICH BECAUSE OF THEIR OBJECTIVES, RESOURCES, OR MOTIVATIONS MAY LEAD INSIGHT INTO POTENTIAL THREAT AGAINST NUCLEAR FACILITIES OR MATERIAL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

SABOTAGE * THEFT/DIVERSION * SAFEGUARDS, NUCLEAR MATERIAL * NRC-13 * JACOBS

132432

JACOBSEN SE

OPTIMAL IMPROVEMENT OF GRAPHS RELATED TO NUCLEAR SAFEGUARDS PROBLEMS

SANDIA LABS., ALBUQUERQUE, N.M.

SAND-76-0435 P. 30 PPS, 1 TAB, 1 FIG, 3 REFS, OCT. 1977

DEVELOPS THE METHODOLOGY FOR OPTIMALLY IMPROVING GRAPHS RELATED TO NUCLEAR SAFEGUARDS ISSUES. IN PARTICULAR, GIVEN A FIXED NUMBER OF DOLLARS, THE REPORT PROVIDES A METHOD FOR OPTIMALLY ALLOCATING SUCH DOLLARS OVER THE ARCS OF A WEIGHTED GRAPH (THE WEIGHTS VARY AS A FUNCTION OF DOLLARS SPENT ON ARCS) SO AS TO IMPROVE THE SYSTEM EFFECTIVENESS MEASURE WHICH IS THE SHORTEST OF ALL SHORTEST PATHS TO SEVERAL TARGETS. ARC HEIGHTS CAN BE EITHER CLOCK TIMES OR DETECTION PROBABILITIES AND THE ALGORITHM DOES NOT EXPLICITLY CONSIDER ALL PATHS TO THE TARGETS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

A AND U PROGRAM * SAFETY PROGRAM * *SAFEGUARDS, NUCLEAR MATERIAL * MODEL * OPTIMIZATION * PROBABILITY * NRC-13

132431

ADAMS RG * FAUJELL AA

TENTATIVE JOB ANALYSIS FOR A HIGH-LEVEL, FIXED-SITE, NUCLEAR SECURITY OFFICER

SANDIA LABS., ALBUQUERQUE, N.M.

SAND-77-1000 P. 32 PPS, 3 REFS, OCT. 1977

A TENTATIVE JOB ANALYSIS FOR A HIGH-LEVEL, FIXED-SITE, NUCLEAR SECURITY OFFICER IS PRESENTED. THE PRIMARY OBJECTIVE OF THE REPORT IS TO PROVIDE A FRAMEWORK FOR EVALUATING THE FUNCTIONS OF A SECURITY OFFICER IN PHYSICAL PROTECTION SYSTEMS. SEVERAL JOB REQUIREMENTS RELATED TO DUTIES, BASIC SKILLS, PERSONAL CONTACTS, SUPERVISION, WORKING CONDITIONS, AND DECISION MAKING ARE PRESENTED. INDIVIDUAL CHARACTER TRAITS DESIRABLE IN SECURITY OFFICERS ARE DESCRIBED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

O AND U PROGRAM * SAFETY PROGRAM * SAFEGUARDS, NUCLEAR MATERIAL * *SECURITY * *SYSTEM ANALYSIS * ON SITE * NRC-13

130577

BALDONADO OC • KEVANY H • RODNEY G • PITTS D • HAZUR H • STEPHENS P • OLCUTT V
EXECUTIVE SUMMARY OF SAFEGUARDS SYSTEMS CONCEPTS FOR NUCLEAR MATERIAL TRANSPORTATION
SYSTEM DEVELOPMENT CORP., McLEAN, VA
NUREG-0330 • 11 PPS, SEPT. 1977

THE NRC OFFICE OF NUCLEAR REGULATORY RESEARCH CONTRACTED WITH SYSTEM DEVELOPMENT CORPORATION (SDC) TO DEVELOP INTEGRATED SYSTEM CONCEPTS FOR THE SAFEGUARD OF SPECIAL STRATEGIC NUCLEAR MATERIALS (SSNM) AGAINST MALICIOUS ACTION DURING INTERFACILITY TRANSPORT. THE CONDUCT AND FINDINGS OF THE PROJECT ARE OUTLINED. THE STUDY WAS DIVIDED INTO THREE MAJOR SUBTASKS: THE DEVELOPMENT OF ADVERSARY ACTION SEQUENCES, THE ASSESSMENT OF THE VULNERABILITY OF THE TRANSPORT OF NUCLEAR MATERIALS TO ADVERSARY ACTION, THE DEVELOPMENT OF CONCEPTUAL SAFEGUARDS SYSTEM DESIGN REQUIREMENTS TO REDUCE VULNERABILITIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*SAFEGUARDS, NUCLEAR MATERIAL • DESIGN CRITERIA • *SPECIAL NUCLEAR MATERIAL • TRANSPORTATION AND HANDLING • JACOBS • NRC-13

130732

BALDONADO OC • KEVANY H • RODNEY G • PITTS D • HAZUR H • STEPHENS P • OLCUTT V
SAFEGUARDS SYSTEMS CONCEPTS FOR NUCLEAR MATERIAL TRANSPORTATION
SYSTEM DEVELOPMENT CORP., McLEAN, VA
NUREG-0335 • 222 PPS, TABS, FIGS, REFS, SEPT. 1977

THE NRC OFFICE OF NUCLEAR REGULATORY RESEARCH COMMISSIONED A PROJECT TO DEVELOP INTEGRATED SYSTEM CONCEPTS FOR THE SAFEGUARD OF NUCLEAR MATERIALS AGAINST MALICIOUS ACTION DURING INTERFACILITY TRANSPORT. THE CONDUCT AND FINDINGS OF THE PROJECT ARE PRESENTED. POTENTIAL THREATS BY TERRORISTS AND OTHERS TO INTERFERE WITH NUCLEAR MATERIALS IN TRANSIT ARE ADDRESSED AND MEASURES WHICH CAN BE TAKEN TO REDUCE BOTH THE LIKELIHOOD OF SUCH THREATS AND THE PROBABILITY OF SUCCESS IF CARRIED OUT ARE RECOMMENDED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*SAFEGUARDS, NUCLEAR MATERIAL • DESIGN CRITERIA • *SPECIAL NUCLEAR MATERIAL • TRANSPORTATION AND HANDLING • NRC-13 • JACOBS

13. RR - EFFLUENT AND RADIATION FIELD SOURCE TERMS IN OPERATING REACTORS

132814

BERMAN SR + CLAYTON ED + DUNST BR
CRITICAL SEPARATION BETWEEN SUBCRITICAL CLUSTERS OF 2.35 WTB 235U ENRICHED UO₂ RODS IN WATER WITH FIRED
NEUTRON POISONS
BATTELLE PACIFIC NORTHWEST LABS., RICHLAND, WASH.
PML-2438 +. 36 PPS, 11 TABS, 11 FIGS, 8 REFS, OCT, 1977

THE EXPERIMENTS ARE CONCERNED WITH THE CRITICAL SEPARATION BETWEEN WATER FLOODED SUBCRITICAL CLUSTERS OF FUEL RODS IN THE PRESENCE OF VARIOUS FIRED NEUTRON POISONS. THE EXPERIMENTS ARE CARRIED OUT IN A 1.0M X 3M X 2.1M DEEP TANK PROVIDED WITH FEATURES SPECIFICALLY DESIGNED AND BUILT FOR THESE EXPERIMENTS. THE INITIAL SERIES OF EXPERIMENTS IN THIS PROGRAM ARE COVERED IN THIS REPORT AND INVOLVE ALUMINUM CLAD 2.35 WTB 235U ENRICHED UO₂ RODS ABOUT 12MM IN DIAMETER BY 916MM IN LENGTH. THE CRITICAL SEPARATION BETWEEN THREE SUBCRITICAL CLUSTERS OF THESE RODS ALIGNED IN A ROW WAS DETERMINED WITH AND WITHOUT SEVERAL NEUTRON ABSORBER MATERIALS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + CRITICALITY EXPERIMENT + FUEL ROD + SHIPPING CONTAINER + URANIUM DIOXIDE + POISON, FIRED + NRC-12 + NRC_RC

130943

DICKINSON J + GAYLORD MD + ROTHE RE + TUCK G
REFERENCE CRITICAL EXPERIMENTS - PROGRESS REPORT FOR PERIOD APRIL 1, 1977 THROUGH JUNE 30, 1977
ATOMIC INTERNATIONAL, GOLDEN, COLO.
RPP-NUREG-2698 +. 42 PPS, SEPT, 1977

CRITICALITY MEASUREMENTS ON PLASTIC-REFLECTED ARRAYS OF URANYL NITRATE SOLUTION CYLINDERS AND SIMILAR MEASUREMENTS IN A SINGLE MINIMALLY-REFLECTED 50-CM-DIAMETER TANK WERE PERFORMED. COMPLETING THE EXPERIMENTAL PHASE OF TASK 4. ALL OF THE TASK 4 DATA ARE SUMMARIZED, AND THE BEST CORRECTED VALUES FOR SOLUTION PROPERTIES, CYLINDER DIMENSIONS, AND CRITICAL HEIGHT ARE GIVEN. THE FABRICATION OF EQUIPMENT FOR THE EXPERIMENTS IS NEARLY COMPLETE, AND PRELIMINARY TESTING IS UNDERWAY. ABOUT 40% OF THE OXIDE HAS BEEN PRESSED INTO BRIQUETTES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS + CRITICALITY EXPERIMENT + CRITICALITY SAFETY + MEASUREMENT + OXIDE + EQUIPMENT DESIGN + NRC-12

132815

ON J + ROTHE RE + TUCK G
REFERENCE CRITICAL EXPERIMENTS PROGRESS REPORT FOR PERIOD JULY 1, 1977, THROUGH SEPTEMBER 30, 1977
ATOMIC INTERNATIONAL, GOLDEN, COLORADO
RPP-NUREG-2715 +. 26 PPS, 3 TABS, 5 FIGS, DEC, 1977

THE MATERIAL COMPOSITION OF THE CONCRETE REFLECTOR, USED IN THE HIGH-ENRICHED URANIUM SOLUTION EXPERIMENTS (TASK 4), HAS BEEN ANALYZED. TWO METHODS WERE USED AND GOOD AGREEMENT FOUND BETWEEN THE TWO. THE URANIUM OXIDE COMPACTION, NEEDED FOR THE LOW-ENRICHED ARRAY EXPERIMENTS (TASK 5) TO ATTAIN SUFFICIENT DENSITY, HAS BEEN COMPLETED. A SYSTEMATIC STUDY HAS BEGUN TO DETERMINE THE BEST METHOD OF DISTRIBUTING MOISTURE THROUGHOUT THE COMPACTED OXIDE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + SAFETY PROGRAM + CRITICALITY EXPERIMENT + CRITICALITY SAFETY + URANIUM OXIDE + DESIGN CRITERIA + NRC_RC

PAGE -00

10. MC - CRITICALITY

130210

OVER MC • KELLER JM • NOTES BJ • NIECHOWY ED
PROCEDURES SOURCE TERM MEASUREMENT PROGRAM
IDAMS NATIONAL ENGINEERING LAB., IDAMS FALLS
NEDS-0304 •. 68 PGS. 1 TAB. 8 FIGS. 9 REFS. DEC. 1977

AN IN-PLANT SOURCE TERM MEASUREMENT STUDY IS BEING CONDUCTED FOR THE MC AT OPERATING LIGHT WATER NUCLEAR POWER PLANTS. THE PURPOSE OF THE STUDY IS TO PROVIDE THE MC WITH A DATA BASE FOR USE IN THE LICENSING PROCESS OF PRESSURIZED WATER REACTOR POWER PLANTS. THIS REPORT PRESENTS THE SAMPLE COLLECTION AND ANALYSIS TECHNIQUES USED IN THE IN-PLANT SOURCE TERM MEASUREMENT PROGRAM. THE INFORMATION WILL BE USED FOR THE PREDICTION OF RADIOACTIVE MATERIALS IN EFFLUENCES FROM NORMAL OPERATION OF LWRs INCLUDING ANTICIPATED OPERATIONAL OCCURRENCES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND S PROGRAM • SAFETY PROGRAM • REACTOR, LWR • SOURCE, CONTINUOUS • MEASUREMENT • MODEL • PROCEDURES AND
MANUALS • RADIOACTIVITY RELEASE • MC-00

A COLLECTION OF KEYWORDS IS USED TO DENOTE THE MAIN SAFETY RELATED POINTS COVERED IN EACH ARTICLE. THE FOLLOWING INDEX IS AN ALPHABETICAL LISTING OF THE KEYWORDS GIVING REFERENCES TO EACH ARTICLE WHICH HAS YELED TO IT.

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