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July - December 1976**

J. R. Buchanan

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55	Design Data and Safety Features of Commercial Nuclear Power Plants, Vol. I, Docket No. 50-3 through 50-295, by F. A. Heddleson, Dec. 1973	\$15.00
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55	Design Data and Safety Features of Commercial Nuclear Power Plants, Vol. II, Docket No. 50-296 through 50-395, by F. A. Heddleson, Jan. 1972	\$15.00
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110	Indexed Bibliography of Thermal Effects Literature-3, by J. G. Morgan, July 1973	\$12.00
111	Reactor Protection Systems: Philosophies and Instrumentation, Reviews from Nuclear Safety, by E. W. Hagen, July 1973	\$15.00

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Engineering Technology Division

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JULY-DECEMBER 1976

J. R. Buchanan
Nuclear Safety Information Center

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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 U.S. Energy Research and Development Administration. 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. A system of key words is used to index the information cataloged by the Center. 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 Transients
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 and Movement in the Environment
(inactive September 1973)
15. Environmental Surveys, Monitoring, and Radiation Exposure
of Man (inactive September 1973)
16. Meteorological Considerations
17. Operational Safety and Experience
18. Safety Analysis and Design Reports

19. Radiation Dose to Man from Radioactivity Release to the Environment (inactive September 1973)
20. Effects of Thermal Modifications on Ecological Systems (inactive September 1973)
21. Effects of Radionuclides and Ionizing Radiation on Ecological Systems (inactive September 1973)

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ACRS-00023

1405

RABIN SA • ABRAM BO • BATES ME • BISHOP ME • BULLOCK
EXAMINATION AND EVALUATION OF THE EFFECTS OF HIGH-TEMPERATURE
CORROSION ON THE PERFORMANCE OF INCOLOY 800 IN
GENERAL ELECTRIC CO., SOUTHWEST, CALIF., ADVANCED REACTOR OPERATION
GEAP-5416 • 22 PAGES, FIGURES, TABLES, REFERENCES, APPENDICES

PREMATURE FAILURES WERE OBSERVED IN ONE OF THE WALL RODS AND IN
ONE OF THE 12-IN. WALL RODS OF THE MARK-III EXPERIMENTAL REACTOR DURING
ITS INCOLOY 800. FAILURES WERE IN SECTION OF MAXIMUM POWER IN REACTOR
POWER ROD. MOST REASONABLE EXPLANATION IS LOW-TEMPERATURE
ACCELERATED BY HIGHER THAN DESIGN CLADDING TEMPERATURES. EVIDENCE
OF RAPID CORROSION, SUCH AS HAVE BEEN ASSOCIATED WITH THE MARK-III
WERE FOUND.

— ABSTRACT
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*FAILURE, FUEL ELEMENT • CLAD • FAILURE, FUEL • INCOLOY • FAILURE, INTERNAL
SUPERHEAT • THERMAL MECHANICAL EFFECT • CORROSION

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BIBLIOGRAPHY OF REPORTS ON RESEARCH SPONSORED BY
THE NRC OFFICE OF NUCLEAR REGULATORY RESEARCH
JULY-DECEMBER 1976

J. R. Buchanan

ABSTRACT

A bibliography of 142 reports published by contractors of the NRC Office of Nuclear Regulatory Research during the period July through December 1976 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 sorted 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

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 four categories: systems engineering, fuel behavior, analysis development, and metallurgy and materials.

Basic (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.

Systems Engineering (NRC-2)

Safety research in systems engineering is addressed primarily to the study of postulated loss-of-coolant accidents in reactors and the effectiveness of emergency core-cooling systems (ECCS). In general, such research is conducted through two types of tests: (1) separate-effects tests to obtain data on those portions of a postulated accident where transient heat transfer and fluid flow phenomena are isolated, thus reducing the number of test variables and simplifying understanding of those complex phenomena; and (2) integral systems tests to study combined phenomena representing an entire postulated accident sequence, both to assess the significance of knowledge gained from separate-effects tests and to confirm the predictive capability of computer codes.

Fuel Behavior (NRC-3)

An important goal of reactor safety research is improved understanding of the response of fuel element pellets and cladding to a postulated nuclear accident. NRC examinations in this area involve a combination of laboratory studies and in-reactor tests, which are experiments conducted in an operating nuclear reactor to observe actual fuel response. These activities, in turn, provide data for predicting fuel behavior in accident conditions through the development and verification of analytical codes.

Analysis Development (NRC-4)

Complex digital computer codes can be used to compute the time and space dependence of important factors during total or partial accident sequences and the effects that might result if a full-scale reactor accident were to occur. The credibility of using such codes in reactor safety assessment is based on the success achieved in using the codes to predict results of various reactor safety research separate-effects and integral systems experiments and on the validity of extrapolating these results to full-size, commercial systems. NRC gives code development and application high priority.

Metallurgy and Materials (NRC-5)

This area of research is aimed toward ensuring the continuing integrity of the primary-system pressure boundaries (vessels, components, and piping) in nuclear reactors. The ability of the steel vessel, components, and piping to retain integrity during operating and accident conditions is governed by (1) the material properties and the response of the steel to the reactor environment and (2) the size and orientation of flaws in the vessel, components, or pipeline.

Site Safety Research (NRC-6)

Reports in this category include information on the effect of geologic, seismologic, meteorologic, and hydrologic phenomena on PWR plant siting, as well as structural and foundation engineering and evaluations of novel siting concepts, such as floating and underground sites.

ADVANCED REACTOR SAFETY RESEARCH

High-priority experimental and analytical safety research programs were initiated in 1975 on advanced reactors - liquid-metal cooled (LMFBR) and gas-cooled (GCFBR) fast breeder reactors and high-temperature gas-cooled reactors (HTGR).

Fast Reactors (NRC-7)

The 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.

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 (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, 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.

OTHER RESEARCH

Another important area of research involves the safety verification of more sophisticated shipping systems.

Transportation Safety Research (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.

ORGANIZATION OF BIBLIOGRAPHY

The bibliography which follows contains all reports published from July through December 1976 by NRC contractors that are sponsored in the program described above. 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. NRC-1. Water Reactor Safety Research, Basic
2. NRC-2. Water Reactor Safety Research, Systems Engineering
3. NRC-3. Water Reactor Safety Research, Fuel Behavior
4. NRC-4. Water Reactor Safety Research, Analysis Development
5. NRC-5. Water Reactor Safety Research, Metallurgy and
Materials
6. NRC-6. Water Reactor Safety Research, Site Safety Research
7. NRC-7. Advanced Reactor Safety Research, Fast Reactors
8. NRC-8. Advanced Reactor Safety Research, Gas-Cooled Reactors
9. NRC-12. Transportation Safety Research.

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276-300	9.25
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A bibliography of the reports published by contractors of the NRC Office of Nuclear Regulatory Research, during the period November 1975 through June 1976 is presented along with abstracts from the Nuclear Regulatory Center computer files. The bibliography was then sorted into the subject categories used by NRC to organize the research programs within the subject categories. The reports are sorted by contractor organization and then chronologically. A brief description of the NRC research program precedes the bibliography.

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100-443887-1000

U.S. FIDELITY NATIONAL LAB., FORT MONROE, SUSSEX COUNTY, VIRGINIA

DISCUSSES THE CURRENT STATUS OF LIQUID-METAL FAST BREEDER REACTOR (LMFR) DEVELOPMENT AND ONE OF THE MAJOR SAFETY ISSUES, A HYPOTHETICAL CATHODIC-ANODE SHORT (CMAS). BIOGRAPHICAL INFORMATION ON EDITORIAL BOARD MEMBERS RELATIVE TO THE DEVELOPMENT OF THE LIQUID-METAL REACTOR AS A SAFE SOURCE OF NUCLEAR POWER IS PRESENTED FOR THE PERIOD 1977 THROUGH 1982. THE BIOGRAPHY CONSISTS OF APPROXIMATELY 1-2 PARAGRAPHS COVERING EARLY RESEARCH AND DEVELOPMENT AND OPERATING EXPERIENCES LEADING UP TO THE PRESENT DESIGN PRACTICES THAT ARE NECESSARY FOR THE DESIGN OF LIQUID-METAL REACTORS. RELEVANT QUOTATIONS AND PERTINENT-TITLE JOURNALS ARE INCLUDED FOR COMPLETION.

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REACTION, LAMIN + CHEMICAL + POLYMERIZATION + SETTING + REACTION, INITIAL + REACTION, FAST + SLOW + HARDENING PROGRAM + QUALITY ASSURANCE + OPERATING EXPERIENCE + UNION STUDY + ACCIDENT, HYPOTHETICAL + NUC-

2. NRC-2 WATER REACTOR SAFETY RESEARCH SYSTEMS ENGINEERING

117099
CRAPU HS + JENSEN WF + SACKETT RE
EXPERIMENTAL DATA REPORT FOR SEMISCALE NUO-1 TEST S-2V-1 (INTERNAL TEST WITH ASYMMETRICAL HEAT TRANSFER TESTS)
AEROMET NUCLEAR COMPANY, IDAHO FALLS
ANCR-NUREG-1306 P. 157 PAGES 5 TABS. 350 FIGS. JUNE 1970

THE TESTS WERE CONDUCTED FROM INITIAL SYSTEM TEMPERATURES AND PRESSURES AT 250 DEG F AND 100 PSIA TO 300 DEG F AND 100 PSIA. DURING ALL PUMP TESTS, HEAT FLUX WAS CONSTANT AND INJECTED IN THE CORE PLUM FROM BOTH THE ACCUMULATION AND LPS SYSTEMS. AN ELECTRICALLY HEATED CORE WAS USED IN THE PRESSURE VESSEL TO SIMULATE THE EFFECTS OF A NUCLEAR CORE DURING HEATUP. ALL PUMP TESTS WERE CONDUCTED WITH A FLAT NAVAL PUMP PROFILE. DURING HEATUP, CORE POWER WAS REDUCED FROM THE INITIAL LEVEL TO SUCH A VALUE AS TO SIMULATE THE SURFACE HEAT FLUX RESPONSE OF NUCLEAR FUEL RODS. THE UNINTERPRETED TEST DATA ARE PRESENTED FOR FUTURE ANALYSIS AND INTERPRETATION IN MORE THAN 300 CHARTS IN ENGINEERING UNITS. THE DATA HAVE BEEN ANALYZED TO THE EXTENT NECESSARY TO ACCOUNT FOR ANOMALOUS CHARACTERISTICS IN TRANSDUCER CALIBRATION.

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HEAT TRANSFER + CALCULATED LOSS OF COOLANT + PUMPUP + TESTING + NUCLEAR + CORE (EXTERNAL) + REACTOR CORE + SAFETY PROGRAM + DESIGN + EMERGENCY COOLING + R AND D PROGRAM + THERMAL HYDRAULIC ANALYSIS + JACOBS

117098
CRAPU HS + JENSEN WF + SACKETT RE
EXPERIMENTAL DATA REPORT FOR SEMISCALE NUO-1 TEST S-2V-1 (INTERNAL TEST WITH ASYMMETRICAL HEAT TRANSFER TESTS)
AEROMET NUCLEAR COMPANY, IDAHO FALLS
ANCR-NUREG-1327 P. 175 PAGES 5 TABS. 302 FIGS. JULY 1970

TEST S-2V-1 WAS CONDUCTED FROM AN INITIAL COOLANT PRESSURE TEMPERATURE OF 250 DEG F AND AN INITIAL PRESSURE OF 100 PSIA. AN ASYMMETRICAL HEAT FLUX CORE WAS USED TO INVESTIGATE THE SYSTEM RESPONSE TO A DEPRESSURIZATION TRANSIENT WITH A FOUR DEPRESSURIZATION DIFFERENT FROM THAT ASSOCIATED WITH A SYMMETRICAL COOLANT LOSS. SYSTEM POWER WAS SET TO MAINTAIN A COOLANT TEMPERATURE DIFFERENTIAL OF 50 DEG F AT FULL CORE POWER OF 100 MW. THE PUMP RESISTANCE OF THE INTACT PUMP WAS 1.50 IN LINE AREA SCALE. AN ELECTRICALLY HEATED CORE WITH A FLAT NAVAL PUMP PROFILE WAS USED IN THE PRESSURE VESSEL TO SIMULATE THE EFFECTS OF A NUCLEAR CORE. DURING SYSTEM DEPRESSURIZATION, CORE POWER WAS REDUCED FROM THE INITIAL LEVEL OF 100 MW TO SIMULATE THE SURFACE HEAT FLUX RESPONSE OF NUCLEAR FUEL RODS UNTIL SUCH TIME THAT DEPARTURE FROM NUCLEATE BOILING (DNB) WOULD OCCUR.

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HEAT TRANSFER + CALCULATED LOSS OF COOLANT + PUMPUP + TESTING + NUCLEAR + CORE (EXTERNAL) + REACTOR CORE + SAFETY PROGRAM + DESIGN + EMERGENCY COOLING + R AND D PROGRAM + THERMAL HYDRAULIC ANALYSIS + JACOBS

119510
CRAPU HS + SACKETT RE
EXPERIMENTAL DATA REPORT FOR SEMISCALE NUO-1 TEST S-2V-2 (INTERNAL TEST WITH REDUCED PRESSURES)
IDAHO NATIONAL ENGINEERING LABORATORY
ANCR-NUREG-1328 P. 100 PAGES 5 TABS. 308 FIGS. AUG. 1970

RECORDED TEST DATA ARE PRESENTED FOR TEST S-2V-2 OF THE SEMISCALE NUO-1 SPECIAL HEAT TRANSFER TEST SERIES. THIS TEST IS AMONG SEVERAL SEMISCALE NUO-1 EXPERIMENTS CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING A HYPOTHEZED LOSS-OF-COOLANT ACCIDENT IN A PRESSURIZED WATER REACTOR SYSTEM. A SIMULATED NUCLEATE BOILING HEAT FLUX CORE WAS USED TO INVESTIGATE THE SYSTEM RESPONSE TO A DEPRESSURIZATION TRANSIENT STARTING FROM A CORE INITIAL PRESSURE THAT WAS ASSOCIATED WITH PRESSURIZED WATER REACTOR OPERATION. AN ELECTRICALLY HEATED CORE WITH A FLAT NAVAL PUMP PROFILE WAS USED IN THE PRESSURE VESSEL TO SIMULATE THE EFFECTS OF A NUCLEAR CORE. DURING THE PRESSURE DEPRESSURIZATION, CORE POWER WAS ACCOMPANIED BY SIMULATED EMERGENCY CORE COOLING INJECTION INTO BOTH THE INTACT AND DROPPING LOOPS. THE PURPOSE OF THIS REPORT IS TO MAKE AVAILABLE THE UNINTERPRETED DATA FROM TEST S-2V-2 FOR FUTURE DATA ANALYSIS AND TEST RESULTS REPORTING ACTIVITIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, MA 01104

ACCIDENT LOSS OF COOLANT + ACCIDENT LOSS OF PRESSURE + REACTOR SAFETY RESEARCH + REACTOR CORE + LOSS OF COOLANT EXPERIMENT + NUCLEAR + ACCIDENT MODEL + DESIGN + HEAT FLUX CRITICAL + PUMP CRITICAL + ACCIDENT DESIGN BASIS + EMERGENCY COOLING SYSTEM + THERMAL HYDRAULIC ANALYSIS + JACOBS

119901
CRAPU HS + SACKETT RE
EXPERIMENTAL DATA REPORT FOR SEMISCALE NUO-1 TEST S-2V-3 (DESIGN HEAT TRANSFER TESTS)
IDAHO NATIONAL ENGINEERING LABORATORY, IDAHO FALLS
ANCR-NUREG-1329 P. 103 PAGES 292 FIGS. 8 TABS. AUG. 1970

PRESENTS EXPERIMENTAL DATA FOR TEST S-2V-3 OF THE SEMISCALE NUO-1 DESIGN HEAT TRANSFER TEST SERIES WHICH WAS CONDUCTED TO INVESTIGATE THE THERMAL HYDRAULIC PHENOMENA ACCOMPANYING A LOSS-OF-COOLANT ACCIDENT IN A PWR. A SIMULATED DEPRESSURIZATION TRANSIENT WITH A FLAT NAVAL PUMP PROFILE WAS ACCOMPANIED BY A PUMP TRANSIENT IN THE ELECTRICALLY HEATED CORE TO SIMULATE THE SURFACE HEAT FLUX RESPONSE. THIS WAS NOT ACCOMPANIED BY EMERGENCY CORE COOLING AND OTHER PRESSURE DEPRESSURIZATION SYSTEM COOLANT SHOTS. THE DATA IS PRESENTED IN THE FORM OF CHARTS IN ENGINEERING UNITS AND HAS

119920 CONTINUOUS

UP A NUCLEAR CORE AND DEBOWEN TO THE PRESSURE SUPPLEMENTARY SYSTEM WAS ACCOMPANIED BY DEPARTURE FROM EMERGENCY COOLANT INJECTION INTO THE HEAT EXCHANGER.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, MA 01104

ACCIDENT: LOSS OF COOLANT * ACCIDENT: LOSS OF PRESSURE * DEBOWEN * OUT OF PILE EXPERIMENT * CORE REPLEADING * EMERGENCY COOLING SYSTEM * CORE REPLEADING SYSTEM * PUMP FAILURE AND DEPARTURE FROM EMERGENCY COOLANT

119985

PETERSON AC * PETERSON AC * PETERSON AC

THERMAL AND HYDRAULIC BEHAVIOR OF THE SERVICABLE MOORE TEST STATION DURING PILE REPLEADING TESTS

IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS

TRIS-NUMBER-1001 * 134 PAGES * 5 TABS * 72 FIGS * OCT. 1970

SELECTED EXPERIMENTAL DATA AND RESULTS CALCULATED FROM EXPERIMENTAL DATA OBTAINED FROM THE CONSTANT REPLEADING TESTS ARE ANALYZED. THE HEAT TRANSFER COEFFICIENTS CALCULATED FROM EXPERIMENTAL DATA AND COMPARED WITH THOSE CALCULATED USING HEAT TRANSFER CORRELATIONS DERIVED FROM THE LITERATURE. THE TEST DATA AND THE CORE THERMAL BEHAVIOR CALCULATED WITH THE THERMAL SUPPLY AND PUMPING RATE COMPUTER CODES ARE COMPARED AND THE CORRELATIONS OF THESE CODES TO PREDICT THE THERMAL-HYDRAULIC BEHAVIOR OCCURRING IN THE CORE REGION ARE EVALUATED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, MA 01104

ACCIDENT: LOSS OF COOLANT * ACCIDENT: LOSS OF PRESSURE * DEBOWEN * OUT OF PILE EXPERIMENT * CORE REPLEADING * EMERGENCY COOLING SYSTEM * CORE REPLEADING SYSTEM * PUMP FAILURE AND DEPARTURE FROM EMERGENCY COOLANT * INTERNAL HYDRAULIC ANALYSIS * COMPUTER PROGRAM * ANALYSIS * COMPARISONS * THEORY AND EXPERIMENT

120309

CHAPMAN NS * CULLINS DL * JACKSON RE

EXPERIMENT DATA REPORT ON SERVICABLE MOORE TEST STATION DURING PILE REPLEADING TESTS

IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS

TRIS-NUMBER-1032 * 177 PAGES * 5 TABS * 313 FIGS * OCT. 1970

TEST STATION WAS CONDUCTED FROM AN INITIAL CORE COOLANT TEMPERATURE OF 200 F AND AN INITIAL PRESSURE OF 2000 PSIA. A SIMULATED DOUBLE-ENDED OFF SET SHEAR CORE COOLANT SHEAR WAS USED TO INVESTIGATE THE SYSTEM RESPONSE TO A DEPRESSURIZATION AND REPLEADING THROUGHOUT USING VOLUME SCALING INJECTION TO ELIMINATE THE HOT BALL DELAY TIME. SYSTEM FLUID WAS SET TO ACHIEVE A CORE FLUID TEMPERATURE DIFFERENTIAL OF 60 F AT A PUMP CORE POWER OF 100 MW. THE PUMP REPLEADING OF THE INITIAL COOLANT DURING CORE AREA STAGING. AN ELECTRICALLY HEATED CORE WITH A FAST NEUTRAL POWER PROFILE WAS USED IN THE PRESSURE VESSEL TO SIMULATE THE EFFECTS OF A NUCLEAR CORE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, MA 01104

ACCIDENT: LOSS OF COOLANT * ACCIDENT: LOSS OF PRESSURE * PRESSURE TRANSIENT * THERMAL TRANSIENT * EMERGENCY COOLING * CORE REPLEADING * ANALYSIS

120310

CHAPMAN NS * CULLINS DL * JACKSON RE

EXPERIMENT DATA REPORT ON SERVICABLE MOORE TEST STATION DURING PILE REPLEADING TESTS

IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS

TRIS-NUMBER-1003 * 176 PAGES * 5 TABS * 300 FIGS * OCT. 1970

TEST STATION WAS CONDUCTED FROM AN INITIAL CORE COOLANT TEMPERATURE OF 200 F AND AN INITIAL PRESSURE OF 2000 PSIA. A SIMULATED DOUBLE-ENDED OFF SET SHEAR CORE COOLANT SHEAR WAS USED TO INVESTIGATE THE SYSTEM RESPONSE TO A DEPRESSURIZATION AND REPLEADING THROUGHOUT USING VOLUME SCALING INJECTION PARAMETERS, MODIFIED TO ACCOUNT FOR ADDITIONAL COOLANT INJECTED DIRECTLY INTO THE CORE PLENUM. IN ADDITION, THE VOLUME OF THE CORE PLENUM WAS REDUCED FROM THAT USED IN PREVIOUS TESTS TO MORE ACCURATELY REPRESENT THE CORE PLENUM OF A PWR, BASED ON A SYSTEM VOLUME SCALING. SYSTEM FLUID WAS SET TO ACHIEVE A CORE FLUID TEMPERATURE DIFFERENTIAL OF 60 F AT A CORE POWER LEVEL OF 100 MW.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, MA 01104

ACCIDENT: LOSS OF PRESSURE * EMERGENCY COOLING * CORE REPLEADING * THERMAL TRANSIENT * PRESSURE TRANSIENT * ACCIDENT: LOSS OF COOLANT * ANALYSIS

120311

USP/2 RA

QUARTERLY TECHNICAL PROGRESS REPORT ON WATER REACTOR SAFETY PHYSICS, PREPARED BY THE NUCLEAR REACTOR COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH SINCE JUNE 1970

IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS

TRIS-NUMBER-1004 * 117 PAGES * 70 FIGS * 57 REFS * OCT. 1970

RESULTS OF A REPLEADING HEAT TRANSFER TEST SERIES WERE ANALYZED TO DETERMINE THE REPLEADING HEAT TRANSFER CHARACTERISTICS OF THE SERVICABLE MOORE ELECTRICALLY HEATED CORE. A SPECIAL TEST TO INVESTIGATE THE SENSITIVITY OF DEPARTURE FROM NUCLEATE BOILING TO A CHANGE IN CORE BEHAVIOR WAS CONDUCTED WITH THE SERVICABLE MOORE FACILITY. TWO COOLANT INJECTION PARAMETERS, INCLUDING AN INVESTIGATION OF HOT-SPOT DELAY EFFECTS, WERE CONDUCTED TO DETERMINE CORE BEHAVIOR VERIFICATION.

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

120114

TEST 5-62-6 OF THE NATIONAL BUREAU OF OCEANOGRAPHY TEST SERIES NUMBER 23 WAS CONDUCTED TO SUPPLY DATA FOR THE U.S. NAVAL RESEARCH COMMISSION (NRC) STANDARD NUMBER FIVE. TO ASSURE THE CREDIBILITY OF THE DATA AND ENSURE THE VALIDITY OF STANDARD NUMBER FIVE, AN ANALYSIS OF THE RESULTS OF TEST 5-62-6 WAS PERFORMED AND IS PRESENTED HEREIN. THIS ANALYSIS CONSISTS OF INVESTIGATIONS OF SYSTEM HYDRAULIC AND CUBE THERMAL DATA. THE CREDIBILITY OF THE SYSTEM HYDRAULIC DATA WAS INVESTIGATED THROUGH COMPARISONS OF THE DATA AS A DATA AND CALCULATIONS FROM RELATED DATA (TEST 5-62-7 AND RELAPS) AND, WHEN NECESSARY, THROUGH EXPLANATION OF PHYSICAL EVENTS. THE CREDIBILITY OF THE CUBE THERMAL DATA WAS JUDGED BY A NUMERICAL ANALYSIS OF PHYSICAL EVENTS. THE RESULTS OF THESE INVESTIGATIONS SUBSTANTIATE THE VALIDITY OF TEST 5-62-6 DATA.

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

120312

PROVIDES THE NECESSARY BACKGROUND INFORMATION TO INTERPRET THE TEST INFORMATION FROM THE CUS OF
FLIGHT TEST (CFT) CAMMINENTAL DATA REPORTS FOR TESTS LI-1 THROUGH LI-10. THE INFORMATION
PROVIDED INCLUDES A CFT SYSTEM DESCRIPTION, AN INSTRUMENTATION AND CUS DATA SYSTEM
DESCRIPTION, AND THE TEST DESCRIPTIONS FOR TESTS LI-1 THROUGH LI-10.

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

117842

DESIGNED PURPOSE IN EXPERIMENTAL STUDIES BEING CONDUCTED BY A 1-DIMENSIONAL MODEL. A FLOWMETER PRESSURIZED WATER HEAT AT PRESSURES TO 80 PSIG TO EXAMINE THE UNDETERMINED IN A FLOWMETER IN ERUPTION PHENOMENA AND PROCESSES ASSOCIATED WITH A LOSS-OF-COOLANT ACCIDENT. A SERIES OF PENETRATION STUDIES WAS UNDERTAKEN TO DEFINE THE EFFECTS OF VARIOUS PARAMETERS AND OPERATING CONDITIONS ON WATER PENETRATION TO THE CORE PLUM. OVER A RANGE OF REVERSE CURVE STEAM FLOW RATES, THE PARAMETERS INVESTIGATED INCLUDE CORE PLUM PRESSURE LEVEL, ERUPTION RATE FLOW RATE AND QUALITY, AND CORE SEG STEAM FLOW. RESULTS ARE DISCUSSED.

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

120254

ONMA during the winter is reported on a program daily, 1/3-scale, twice a week from, at
pressure: 10 to 150, to 1000 STEAM-DRIVEN PROGRAMS and a 1/30 scale, and reported on 1000
PROGRAM COMPARISON, EXPERIMENTAL DATA EVALUATION, ANALYTICAL MODEL DEVELOPMENT, TECHNICAL
ASSISTANCE, and SPECIAL STUDIES, AND DEFINITION AND MANAGEMENT OF PROJECTS, and TECHNICAL

DEPARTMENT OF COMMERCE - NAVAL AND TECHNICAL INFORMATION SERVICE, 1200 DEPARTMENT OF COMMERCE, WASHINGTON, D.C. 20540

2.2 Future

OF 1574

DECLASSIFIED BY 2025 RELEASE UNDER E.O. 14176

100-443887-11

DATE: 31. 10. 1976

1954年(1953年) 1月 1日 1月 1日 1月 1日 1月 1日

EXPANDED WORKING CURVES WERE OBTAINED FROM ARA-2, ARA-3 AND T-1, AND SET UP IN A SERIES OF AUTOMATIC EXPERIMENTS. THE T-2 AND T-3 ARE TO BE COMPLETED BY THE END OF THE TEST RUNS IN THE MAIN INSTRUMENTATION AREA. DURING THIS PERIOD, WE HAVE AN INSTRUMENTATION WELDER HAS BEEN SET UP IN ADDITION. THE DATA ACQUISITION SYSTEM HAS ALSO OBTAINED INFORMATION ON THE TEST OPERATIONS IN PLACE IT IS THE INFORMATION OBTAINED FOR INTER. T-2 IS THE CURRENT COMPARISON. THE WELD SENSORS TO BE USED IN THE FIRST OXYGEN COMBUSTION HAVE BEEN INSTALLED IN THE STEADY STATE FURNACE LOOP. THE CALIBRATED SENSORS HAVE BEEN INSTALLED IN THE FURNACE OPERATING. DEVELOPMENT OF THE COMPUTER PROGRAMS FOR ANALYSIS OF THE FURNACE DATA CONTINUES.

DD FORM 117 - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM = HX-2 • REALTIME LOG • PRESSURE LOSS • FLOWMETER OF • SAFETY ANALYSIS • PUBLICATIONS • IN ADVANCE
PROGRAM • FOR MORE INFORMATION • CALL OR WRITE TO US TODAY

AL 9003

CONFIDENTIAL - SECURITY INFORMATION

an evaluation of ECC nucleation data using the existing parameters

總代理：物產貿易株式會社

CONFIDENTIAL - [REDACTED] 12/11/2013 12:11:12 PM

THE RESULTS OF STEAM-DRIVEN AND AIR-DRIVEN CONFINEMENTAL FLOW TEST SIMULATING DISPERSED FLUID DEFORMATION IN SCALES FOR MATHS ARE EVALUATED WITH RESPECT TO VARIOUS DIMENSIONLESS SCALING PARAMETERS. DATA AT SEVERAL SCALES ARE COMPARED WITH A CORRELATION BASED ON THE LARGEST 100-SCALE DATA USING A 30 VARIABLE FORMER EMPLOYING THE ANALOGY CONCEPTS, A CHARACTERISTIC DIMENSION THAT DEPENDS ON SCALE SIZE) AND THE AUTOSCALE METHOD FORMER EMPLOYED WITH SURFACE TENSION AND DOES NOT CONTAIN A CHARACTERISTIC PHYSICAL DIMENSION. ADDITIONAL LARGE SCALE TESTS ARE REQUIRED IN ORDER TO TEST SCALING PARAMETERS FOR CONFINEMENTAL FLOW OF NON-NEUTROPHILIC

DAAG-4117 - NATIONAL Technical Information SERVICE, U. S. Department of Commerce, Springfield, MA 01104

REACTION, FOR • ACCIDENT LOSS OF CONTAINMENT • ACCIDENT, LOSS OF PRESSURE • JOURNAL • CASE REPORTING • EMERGENCY
CONTAINMENT SYSTEM • LOSS OF PILE STABILIZATION • PILE FAILURE AND SUBSIDENCE • PILE

SECRET

ROCK ISLAND COUNTY

COUNTERCURRENT FLOW, COOLED CONDENSING STEAM, AND TRANSIENT STEAM FLOW TESTS AT 1713 - POWER 404-56 MW

PROGRESS REPORT APRIL 1 - JUNE 30, 1976

CASE NAME [REDACTED] DATE OF BIRTH [REDACTED]

CONFIDENTIAL - (U) (S) (C) (E) (F) (G) (H) (I) (J) (K) (L) (M) (N) (O) (P) (Q) (R) (S) (T) (U) (V) (W) (X) (Y) (Z) (AA) (AB) (AC) (AD) (AE) (AF) (AG) (AH) (AI) (AJ) (AK) (AL) (AM) (AN) (AO) (AP) (AQ) (AR) (AS) (AT) (AU) (AV) (AW) (AX) (AY) (AZ) (BA) (BB) (BC) (BD) (BE) (BF) (BG) (BH) (BI) (BJ) (BK) (BL) (BM) (BN) (BO) (BP) (BQ) (BR) (BS) (BT) (BU) (BV) (BW) (BX) (BY) (BZ) (CA) (CB) (CC) (CD) (CE) (CF) (CG) (CH) (CI) (CJ) (CK) (CL) (CM) (CN) (CO) (CP) (CQ) (CR) (CS) (CT) (CU) (CV) (CW) (CX) (CY) (CZ) (DA) (DB) (DC) (DD) (DE) (DF) (DG) (DH) (DI) (DJ) (DK) (DL) (DM) (DN) (DO) (DP) (DQ) (DR) (DS) (DT) (DU) (DV) (DW) (DX) (DY) (DZ) (EA) (EB) (EC) (ED) (EE) (EF) (EG) (EH) (EI) (EJ) (EK) (EL) (EM) (EN) (EO) (EP) (EQ) (ER) (ES) (ET) (EU) (EV) (EW) (EX) (EY) (EZ) (FA) (FB) (FC) (FD) (FE) (FF) (FG) (FH) (FI) (FJ) (FK) (FL) (FM) (FN) (FO) (FP) (FQ) (FR) (FS) (FT) (FU) (FV) (FW) (FX) (FY) (FZ) (GA) (GB) (GC) (GD) (GE) (GF) (GG) (GH) (GI) (GJ) (GK) (GL) (GM) (GN) (GO) (GP) (GQ) (GR) (GS) (GT) (GU) (GV) (GW) (GX) (GY) (GZ) (HA) (HB) (HC) (HD) (HE) (HF) (HG) (HH) (HI) (HJ) (HK) (HL) (HM) (HN) (HO) (HP) (HQ) (HR) (HS) (HT) (HU) (HV) (HW) (HX) (HY) (HZ) (IA) (IB) (IC) (ID) (IE) (IF) (IG) (IH) (II) (IJ) (IK) (IL) (IM) (IN) (IO) (IP) (IQ) (IR) (IS) (IT) (IU) (IV) (IW) (IX) (IY) (IZ) (JA) (JB) (JC) (JD) (JE) (JF) (JG) (JH) (JI) (JJ) (JK) (JL) (JM) (JN) (JO) (JP) (JQ) (JR) (JS) (JT) (JU) (JV) (JW) (JX) (JY) (JZ) (KA) (KB) (KC) (KD) (KE) (KF) (KG) (KH) (KI) (KJ) (KK) (KL) (KM) (KN) (KO) (KP) (KQ) (KR) (KS) (KT) (KU) (KV) (KW) (KX) (KY) (KZ) (LA) (LB) (LC) (LD) (LE) (LF) (LG) (LH) (LI) (LJ) (LK) (LL) (LM) (LN) (LO) (LP) (LQ) (LR) (LS) (LT) (LU) (LV) (LW) (LX) (LY) (LZ) (MA) (MB) (MC) (MD) (ME) (MF) (MG) (MH) (MI) (MJ) (MK) (ML) (MM) (MN) (MO) (MP) (MQ) (MR) (MS) (MT) (MU) (MV) (MW) (MX) (MY) (MZ) (NA) (NB) (NC) (ND) (NE) (NF) (NG) (NH) (NI) (NJ) (NK) (NL) (NM) (NN) (NO) (NP) (NQ) (NR) (NS) (NT) (NU) (NV) (NW) (NX) (NY) (NZ) (OA) (OB) (OC) (OD) (OE) (OF) (OG) (OH) (OI) (OJ) (OK) (OL) (OM) (ON) (OO) (OP) (OQ) (OR) (OS) (OT) (OU) (OV) (OW) (OX) (OY) (OZ) (PA) (PB) (PC) (PD) (PE) (PF) (PG) (PH) (PI) (PJ) (PK) (PL) (PM) (PN) (PO) (PP) (PQ) (PR) (PS) (PT) (PU) (PV) (PW) (PX) (PY) (PZ) (QA) (QB) (QC) (QD) (QE) (QF) (QG) (QH) (QI) (QJ) (QK) (QL) (QM) (QN) (QO) (QP) (QQ) (QR) (QS) (QT) (QU) (QV) (QW) (QX) (QY) (QZ) (RA) (RB) (RC) (RD) (RE) (RF) (RG) (RH) (RI) (RJ) (RK) (RL) (RM) (RN) (RO) (RP) (RQ) (RR) (RS) (RT) (RU) (RV) (RW) (RX) (RY) (RZ) (SA) (SB) (SC) (SD) (SE) (SF) (SG) (SH) (SI) (SJ) (SK) (SL) (SM) (SN) (SO) (SP) (SQ) (SR) (SS) (ST) (SU) (SV) (SW) (SX) (SY) (SZ) (TA) (TB) (TC) (TD) (TE) (TF) (TG) (TH) (TI) (TJ) (TK) (TL) (TM) (TN) (TO) (TP) (TQ) (TR) (TS) (TT) (TU) (TV) (TW) (TX) (TY) (TZ) (UA) (UB) (UC) (UD) (UE) (UF) (UG) (UH) (UI) (UJ) (UK) (UL) (UM) (UN) (UO) (UP) (UQ) (UR) (US) (UT) (UU) (UV) (UW) (UX) (UY) (UZ) (VA) (VB) (VC) (VD) (VE) (VF) (VG) (VH) (VI) (VJ) (VK) (VL) (VM) (VN) (VO) (VP) (VQ) (VR) (VS) (VT) (VU) (VV) (VW) (VX) (VY) (VZ) (WA) (WB) (WC) (WD) (WE) (WF) (WG) (WH) (WI) (WJ) (WK) (WL) (WM) (WN) (WO) (WP) (WQ) (WR) (WS) (WT) (WU) (WV) (WW) (WX) (WY) (WZ) (XA) (XB) (XC) (XD) (XE) (XF) (XG) (XH) (XI) (XJ) (XK) (XL) (XM) (XN) (XO) (XP) (XQ) (XR) (XS) (XT) (XU) (XV) (XW) (XX) (XY) (XZ) (YA) (YB) (YC) (YD) (YE) (YF) (YG) (YH) (YI) (YJ) (YK) (YL) (YM) (YN) (YO) (YP) (YQ) (YR) (YS) (YT) (YU) (YV) (YW) (YX) (YZ) (ZA) (ZB) (ZC) (ZD) (ZE) (ZF) (ZG) (ZH) (ZI) (ZJ) (ZK) (ZL) (ZM) (ZN) (ZO) (ZP) (ZQ) (ZR) (ZS) (ZT) (ZU) (ZV) (ZW) (ZX) (ZY) (ZZ)

PRELIMINARY ANALYSIS OF EXPERIMENTS IN A 1/2-INCH DIAMETER, VERTICAL, CAPPED PRESSURE VESSEL, PERFORMED TO DETERMINE THE EFFECT OF TEMPERATURE ON THE RATE OF COUNTER-CURRENT FILLING DURING A POSTULATED LOSS OF COOLANT ACCIDENT, INDICATE THAT THE FILLING RATE OF COUNTER-CURRENT FLOWS AT VARIOUS SYSTEM PRESSURES AND TEMPERATURES WOULD BE STEADILY AND PREDICTABLY SUPPLEMENTED BY A TRANSIENT STEADY FLOW WITH AN ESTIMATED SUPPLEMENTED PRESSURE WAVE GENERATED BY COMPRESSIONS OF THE COUNTER-CURRENT FLOW DATA AND A CORRELATION DEVELOPED BY CHANG AND HANCOCK. GOOD AGREEMENT IS DEMONSTRATED. SAMPLES OF THE OTHER TWO TESTS ARE BEING PERFORMED AND DISCUSSION.

AVAILABILITY - NATIONAL Technical INFORMATION SERVICE, U. S. Department of Commerce, Springfield, MA 01104

REACTOR, POW & ACCIDENT, LOSS OF COOLANT & INCIDENT, LOSS OF PRESSURE & RECOVERY & FAILURE, PIPE & LINE
REFLECTIONS, PUMP TRIP AND EXPERIMENTS, SEC-2 & CORE MELTING SYSTEM, SET UP & PLANT CONTROLS &
EMERGENCY LOCAL SYSTEM

117723

SUBJECT: [REDACTED] b6 b7C

FOR INFORMATION OF THE FBI, THE FOLLOWING INFORMATION WAS OBTAINED FROM THE NEW YORK OFFICE OF THE NEW YORK STATE DEPARTMENT OF CORRECTIONS AND PROBATION, NEW YORK, ON 10/10/68:

MEMBER ELECTRIC CO., 220 N. 3d ST.,

6400-21210 2. 125 Pgs. n. Inq. n. P. 160. 31 APR 55. FILE 1972

A NUMBER OF IMPORTANT LOCALITY MECHANISMS WERE DOCUMENTED FOR OFFICE TO OFFICE TO TAKE IN THE CURRENT
 NEW LOCAL EVALUATION APPROACH. THESE CHANGING MECHANISMS WERE LISTED AS: 1) A NEW EVALUATION APPROACH
 DESIGNED IN THE BUREAU THROUGHOUT THE REGION AND CONSIDER THE LOCAL AND THE BUREAU TO BE A

117725 CONTINUED

UPDRAFT COLLING IN THE UPPER ZONE BY STEAM GENERATED FROM FLASHING DUE TO DEPRESSURIZATION AND HEAT TRANSFER TO THE FLUID INVENTORIES IN THE LOWER ZONE, AND (2) FUEL HEATING DURING THE LOWER PLENUM FLASHING SURGE AND FALLBACK OF THE FLUID (EJECTOR) IN THE UPPER PLENUM DURING THE LOWER PLENUM FLASHING SURGE FROM THE UPPER PLENUM.

AVAILABILITY - ENDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

REACTOR, BWR + CLADDING + HEAT TRANSFER + ACCIDENT, LOSS OF COOLANT + MUCKUP + NRC-2 + EXPERIMENT + SAFETY PROGRAM + BLOWDOWN + H AND D PROGRAM + TEMPERATURE + THERMAL HYDRAULIC ANALYSIS + JACOBS

117726

ROBINSON AP

BWR BLOWDOWN/EMERGENCY CORE COLLING FIRST QUARTERLY PROGRESS REPORT JANUARY 1-MARCH 31, 1976

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

WEAP-21304-1 P. 14 PPS, 1 TAB, 2 FIGS, 3 REFS, APRIL 1976

A MAJOR REQUIREMENT IN THE DESIGN OF POWER REACTOR SYSTEMS IS THE LIMITATION OF FUEL CLADDING TEMPERATURES BELOW SPECIFIED VALUES DURING BOTH NORMAL OPERATION AND AN UNDESIRABLE, BUT POSTULATED, LOSS-OF-COOLANT (LOCA). TO MEET THIS DESIGN REQUIREMENT IT IS NECESSARY TO BE ABLE TO PREDICT SYSTEM PERFORMANCE DURING A LOCA. SINCE THIS TYPE OF INFORMATION IS NOT OBTAINABLE FROM TESTS ON ACTUAL REACTORS, SCALED SYSTEM TEST PROGRAMS ARE USED TO PROVIDE BASIC SYSTEM PERFORMANCE INFORMATION. THE PROGRAM WILL PROVIDE A BASIS FOR EVALUATING BWR SYSTEM PERFORMANCE THROUGHOUT THE ENTIRE LOCA TRANSIENT FROM BREAK INITIATION TO CORE REPLETION. FOR A POSTULATED RUPTURE OF A STEAM LINE OR RECIRCULATION LINE IN A BOWLING WATER REACTOR (BWR), THE PROGRAM WILL: (1) OBTAIN AND EVALUATE BASIC BOWLING WATER TEST SYSTEM LIMITATIONS WHICH HAVE CALCULATED PERFORMANCE CHARACTERISTICS SIMILAR TO A BWR WITH BOWLING WATER BURNER BURNERS USING A HYPOTHETICAL LOCA; AND (2) DETERMINE THE DEGREE TO WHICH BOWLING WATER BWR SYSTEM AND FUEL BURNERS DESCRIBE THE OBSERVED PHENOMENA AND, AS NECESSARY, DEVELOP IMPROVED MODELS WHICH ARE GENERALLY USEFUL IN IMPROVED LOCA ANALYSIS METHODS.

AVAILABILITY - ENDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

REACTOR, BWR + CLADDING + ACCIDENT, LOSS OF COOLANT + COMPARISON, THEORY AND EXPERIENCE + NRC-2 + EXPERIMENT + SAFETY PROGRAM + BLOWDOWN + H AND D PROGRAM + TEMPERATURE + THERMAL HYDRAULIC ANALYSIS + JACOBS

118047

WALKER JP

BWR BLOWDOWN/EMERGENCY CORE COLLING SECOND QUARTERLY PROGRESS REPORT APRIL 1-JUNE 30, 1976

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

WEAP-21304-2 P. 14 PPS, 3 TABS, 2 FIGS, 2 REFS, AUG. 1976

BWR BLOWDOWN/EMERGENCY CORE COLLING (BOWEC) PROGRAM WORK COMPLETED THIS QUARTER IS SUMMARIZED. START UP COUNT-DOWNING FLOW LIMITING (CCFL) TESTING, COMPLETION OF THE BOWEC PRELIMINARY PROGRAM PLAN, FIRST DRAFT OF THE BOWDOWN HEAT TRANSFER (HWT) TEST PLAN, AND RESULTS OF THE LONG THROAT NOZZLE TESTING HIGHLIGHT THE WORK COMPLETED IN THE QUARTER.

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

GENERAL ELECTRIC + ACCIDENT, LOSS OF COOLANT + EMERGENCY COLLING SYSTEM + OUT OF PILE EXPERIMENT + FLOW THEORY AND EXPERIMENTS + FLOW, TWO PHASE + BLOWDOWN + NRC-2 + REACTOR, BWR

120308

WALKER JP

BWR BLOWDOWN/EMERGENCY CORE COLLING PROGRAM - 60-HOUR BOWDOWN HEAT TRANSFER TEST PLAN

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

WEAP-21333 P. 14 PPS, 4 TABS, 5 FIGS, 8 REFS, SEPT. 1976

A TEST PLAN FOR THE 60-HOUR BOWDOWN HEAT TRANSFER TESTING IS PRESENTED. THE TEST PROGRAM WILL PROVIDE A COMPARISON BETWEEN TAT AND BOW BURNERS AND BETWEEN BOW/BOW AND BOW/BOW SCALING. TEST PARAMETERS AND VALUES ARE SELECTED. INDIVIDUAL TESTS ARE DESIGNED AND ARRANGED IN A TEST MATRIX.

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

BLOWDOWN + HEAT TRANSFER + CLADDING + MUCKUP + FUEL ELEMENTS + EMERGENCY COLLING + NRC-2

116371

MCCAULEY ED

REACTOR CONTAINMENT ANALYSIS FOR LOCA SUPPRESSION SYSTEMS VIA TERTIARY PROGRESS REPORT JANUARY 31, 1976-MARCH 31, 1976

LAWRENCE LIVERMORE LABORATORY, CALIFORNIA

LLL-RENG-1000-70-1 P. 20 PPS, 2 TABS, 12 FIGS, 10 REFS, JUNE 30, 1976

DESCRIBES THE GOALS OF THE PROGRAM TO ANALYZE THE FLUID DYNAMICS OF REACTOR PRESSURE-SUPPRESSION CONTAINMENT. DISCUSSES THE CURRENT CALCULATIONAL AND CODE DEVELOPMENT ACTIVITY UNDERWAY TO ANALYZE THE CONTAINMENT. INITIAL STUDIES WITH THE MAINTAIN CODE, INCLUDING THE DESCRIPTION OF A BENCHMARK PROGRAM, ARE REPORTED. DEFINES SEVERAL CLASSES OF PROBLEMS THAT WILL INCREASE THE UNDERSTANDING OF AND THE APPROACH OF THE CHAMP-1 COMPUTER CODE ARE EXPLAINED. AS WELL AS THE STATUS OF THE DEVELOPMENTAL WORK TO CALCULATE MULTIPHASE FLUID FLOW.

110371 CONTINUED
 AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

REACTION, BUR - COMPUTER PROGRAM - CONTAINMENT ANALYSIS - CONTAINMENT, PRESSURE SUPPRESSION - MNL-2 - R AND D PROGRAM - JACOBS

117002
 SMITH TA
 HEAT TRANSFER AND CHARACTER OF LOW PRESSURE WATER IN A HEATED VERTICAL TUBE
 MASSACHUSETTS INSTITUTE OF TECHNOLOGY, CAMBRIDGE
 NUREG-0105 - 110 PAGES, 2 TABS, 20 FIGS, AUG. 1970

LOCAL HEAT TRANSFER COEFFICIENTS IN THE STABLE FILM BOILING AND DISPERSED FILM REGIMES WERE STUDIED FOR THE UPWARD FLOW OF LOW PRESSURE WATER IN A HEATED VERTICAL TUBE. WALL TEMPERATURES WERE MAINTAINED CONSTANT WITH TIME AND ALONG THE TUBE SO THAT BOTH AXIAL AND FLOW DIRECTION GRADIENTS APPROXIMATED ZERO. SEVERAL THEORIES FOR THE DIFFERENT POSSIBLE TYPES OF FILM CLIPPING IN TURBULENT, TUBE IN FILM WERE COMPARED WITH THE EXPERIMENTAL DATA. THE LAMINAR POINT FOR LOW FLOWING RATES (1 INCH/SEC OR LESS) WAS INFERRED FROM THESE COMPARISONS AND GAVE GOOD AGREEMENT WITH THE PLUMMER CRITICAL MASS CRITERION FOR LIQUID CARRY OVER.

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

HEAT TRANSFER - ACCIDENT, LOSS OF COOLANT - MNL-2 - STEEL, STAINLESS - REACTION, LOR - EXPERIMENT - SAFETY PROGRAM - R AND D PROGRAM - FILM BOILING - THERMAL, MNL-4 - JACOBS

117070
 KIRCHNER RL
 REFLUID HEAT TRANSFER IN A LIGHT WATER REACTOR
 MASSACHUSETTS INSTITUTE OF TECHNOLOGY, CAMBRIDGE
 NUREG-0104 VOLS. 1 & 2 - 205 PAGES, 50 FIGS, 70 REFS, AUG. 1970

THE HEAT TRANSFER PROCESSES OCCURRING DURING THE REFLUID PHASE OF A HYPOTHETICAL LOSS OF COOLANT ACCIDENT ARE INVESTIGATED. IT WAS FOUND THAT PEAK CLADDING TEMPERATURE OCCURRING ARE TERMINATED BY THE PRESENCE OF ENTRAINED LIQUID DROPS IN THE VERTICAL PLUMING VAPOR STREAM. THIS DISPERSED FLOW FILM BOILING REGIME IS CREATED DUE TO A COMPLEX SET OF UPSTREAM THERMAL-HYDRAULIC CONDITIONS. WHILE PREDICTION OF THE REFLUIDING OF A NUC REACTOR REQUIRES A CALCULATIONAL METHOD WHICH DESCRIBES THE MANY PROCESSES OCCURRING IN THE CORE DURING THE TRANSIENT. A PHYSICALLY BASED CALCULATIONAL METHOD HAS BEEN DEVELOPED FOR PREDICTION OF REFLUID HEAT TRANSFER.

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

COMPUTER PROGRAM - CLADDING - HEAT TRANSFER - ACCIDENT, LOSS OF COOLANT - ANALYTICAL MODEL - MNL-2 - REACTION, LOR - DROPLET - SAFETY PROGRAM - SATURATED VAPOR - R AND D PROGRAM - TEMPERATURE - MNL-4 - FILM REFLUIDING - JACOBS

115340
 THOMAS GG - HELMS RE - DAVENUM RE
 QUARTERLY PROGRESS REPORT ON BLOWDOWN HEAT TRANSFER PROGRAM FOR JANUARY-MARCH 1970
 OAK RIDGE NATIONAL LABORATORY, TENNESSEE
 OML/NUREG/TH-10 - 70 PAGES, TABS, FIGS, JULY 1970

TWO-PHASE FLOW TESTS HAVE NOT SHOWN THAT WHEN THE TURBINE METER IS UPSTREAM OF THE ORAL MISH, FLOW MEASUREMENT ACCURACY INCREASES SIGNIFICANTLY WHEN A FLOW DISPENSER IS LOCATED IMMEDIATELY DOWNSTREAM OF THE TURBINE METER. A SECOND HEATER TAKEN FROM THE PRODUCTION LOT FOR THIS SURVEY IS NOW BEING TESTED IN THE LTR. RELAPS RODS AND ITS ASSOCIATED SUPPORT PACKAGES IS NOW OPERATIONAL ON THE ORAL COUNTERS. EMPHASIS IS BEING PLACED ON IMPROVING THE TRIP MODEL BY INCORPORATING MEASURES, VALUES OF LOOP VOLUMES, PUMP CAVITATION BEHAVIOR, AND ORAL PRESSURE DROP INTO THE MODEL.

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

COMPUTER PROGRAM - THERMAL EXPERIMENT - REACTION, LOR - FLOW, TWO PHASE - HEAT TRANSFER EXPERIMENT - HEAT TRANSFER, CONVECTION - SAFETY PROGRAM - BLOWDOWN - R AND D PROGRAM - THERMAL HYDRAULIC ANALYSIS - JACOBS

115301
 PER 66
 JUNE 1970 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROVIDED AT OAK RIDGE NATIONAL LABORATORY
 OAK RIDGE NATIONAL LABORATORY, TENNESSEE
 OML/NUREG/TH-00 - 27 PAGES, JULY 1970

THIS PROGRESS REPORT CONTAINS SHORT 11 TO 4 PAGE SUMMARIES OF THE ACCOMPLISHMENTS DURING THE MONTH ON THE FOLLOWING ORNL PROJECTS: HEAVY SECTION STEEL TECHNOLOGY PROGRAM, FISSION PRODUCT BETA AND GAMMA ENERGY RELEASE, LOCA RELEASE FROM LWR FUEL, MULTIPHASE FLOW TESTS, NUCLEAR SAFETY INFORMATION CENTER, FOR BLOWDOWN HEAT TRANSFER-SEPARATE EFFECTS, ZIRCONIUM FUEL CLADDING CORROSION STUDIES, ZIRCONIUM METAL-WATER OXIDATION KINETICS, ALUMINUM RELEASE AND TRANSPORT FROM LWR FUEL, NTR SAFETY ANALYSIS AND RESEARCH, AND DESIGN CRITERIA FOR PIPING AND NOZZLES.

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

115361 (CONTINUED)

ACCIDENT ANALYSIS • AEROSOL • DESIGN CRITERIA • FISSION PRODUCT RELEASE • FAILURE, FUEL ELEMENT • HEAT TRANSFER • CALCULATION, LOSS OF COOLANT • METAL WATER REACTION • OXIDATION • ZIRCONIUM • FISSION PRODUCT TRANSPORT • GROSS DATA • GROSS DATA • REACTOR, LWR • FAILURE, CLADDING • ZIRCALOY • SAFETY PROGRAM • ALUMINUM • HSC • ENERGY LEVEL • R AND D PROGRAM • REACTOR, HTGR • REACTOR, LWR • HSC • PIPES AND PIPE FITTINGS

120402

ORNL-UT • DIAL 42

AN APPARATUS FOR SPOT WELDING SHIELDED THERMOCOUPLES TO THE INSIDE OF SMALL-DIAMETER TUBES AT PRECISE LOCATIONS
ORNL-UT • DIAL 42
ORNL-UT/UT-33 • 17 PPS, 5 FIGS, AUG. 1976

THIS REPORT DESCRIBES EQUIPMENT AND PROCEDURES USED TO SPOT WELD INDIUM- OR STAINLESS-STEEL-SHEATHED THERMOCOUPLES TO THE INSIDE DIAMETER OF ZIRCALOY TUBING TO MEET THE REQUIREMENTS OF THE NUCLEAR URST TEST (NUST) PROGRAM AT ORNL. SPOT WELDING AND JIGS CLAMPING TOOLS WERE FABRICATED TO REMOVE THE GROSS CONTAMINANTS ON THE ZIRCALOY TUBING AT LOCAL AREAS AND SPOT WELD FROM THERMOCOUPLES SEPARATED CIRCUMFERENTIALLY BY 90 DEGREES AT ANY AXIAL DISTRIBUTION DESIRED. IT WAS FOUND NECESSARY TO APPLY A NICKEL COATING TO STAINLESS-STEEL-SHEATHED THERMOCOUPLES TO OBTAIN ACCEPTABLE WELDS. THE MATERIAL AND SHAPE OF THE INNER ELECTRODE AND RESISTANCE BETWEEN INNER AND OUTER ELECTRODES WERE FOUND TO BE CRITICAL PARAMETERS IN OBTAINING ACCEPTABLE WELDS.

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

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, LWR • STEEL, STAINLESS • TANTALUM • OTHER THERMOCOUPLE • STUDIES • WELDING • EQUIPMENT DESIGN • ZIRCALOY • TESTING • NICKEL • COATING • NUC-3

116037

PRE 66

JULY 1976 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT ORNL NATIONAL LABORATORY
ORNL-UT • DIAL 42
ORNL-UT/UT-33 • 25 PPS, AUG. 1976

THIS MONTHLY REPORT CONTAINS SHORT (1 TO 4 PAGE) SUMMARIES OF THE ACCOMPLISHMENTS DURING THE MONTH IN THE FOLLOWING ORNL PROJECTS: HEAVY SECTION STEEL TECHNOLOGY PROGRAM, FISSION PRODUCT DATA AND GROSS ENERGY RELEASE, LOCA RELEASE FROM LWR FUEL, MULTIPLE BURST TESTS, NUCLEAR SAFETY INFORMATION CENTER, FOR ALUMINUM HEAT TRANSFER-SEPARATE EFFECTS, ZIRCALOY FUEL CLADDING COLLAPSE STUDIES, ZIRCONIUM METAL-WATER OXIDATION KINETICS, AEROSOL RELEASE AND TRANSPORT FROM LWR FUEL, HTGR SAFETY ANALYSIS AND RESEARCH, AND DESIGN CRITERIA FOR PIPING AND NOZZLES.

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

ACCIDENT ANALYSIS • AEROSOL • DESIGN CRITERIA • FISSION PRODUCT RELEASE • FAILURE, FUEL ELEMENT • HEAT TRANSFER • METAL WATER REACTION • OXIDATION • ZIRCONIUM • FISSION PRODUCT TRANSPORT • HSC-2 • GROSS DATA • GROSS DATA • HSC-3 • REACTOR, LWR • FAILURE, CLADDING • ZIRCALOY • SAFETY PROGRAM • HSC-3 • ALUMINUM • HSC • ENERGY LEVEL • R AND D PROGRAM • REACTOR, HTGR • REACTOR, LWR • HSC-4 • HSC-7 • HSC • PIPES AND PIPE FITTINGS

117072

PRE 66

AUGUST 1976 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT ORNL NATIONAL LABORATORY
ORNL-UT • DIAL 42
ORNL-UT/UT-33 • 25 PPS, OCT. 1976

PRESENTS HIGHLIGHTS OF THE FOLLOWING PROGRAMS AND ACTIVITIES: HEAVY SECTION STEEL TECHNOLOGY; FISSION PRODUCT DATA AND GROSS ENERGY RELEASE; LOCA RELEASE FROM LWR FUEL; MULTIPLE BURST TESTS; NUCLEAR SAFETY INFORMATION CENTER; FOR ALUMINUM HEAT TRANSFER-SEPARATE EFFECTS; ZIRCALOY FUEL CLADDING COLLAPSE STUDIES; ZIRCONIUM METAL-WATER OXIDATION KINETICS; AEROSOL RELEASE AND TRANSPORT FROM LWR FUEL; HTGR SAFETY ANALYSIS AND RESEARCH; DESIGN CRITERIA FOR PIPING AND NOZZLES; AND GROSS CONVERSION FACTORS FOR IRRADIATION OF MATERIALS.

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

AEROSOL • DESIGN CRITERIA • CLADDING • FISSION PRODUCT RELEASE • METAL WATER REACTION • HSC-2 • GROSS CALCULATION, INTERNAL • ORNL • HSC-3 • NOZZLE • HSC-3 • HSC • R AND D PROGRAM • REACTOR, LWR • HSC-4 • AGENCY, HSC • HSC-7 • HSC • PIPES AND PIPE FITTINGS

120401

SIMPSON 6A • INTRUS 30 • CUM 15

INFRARED INSPECTION AND CALIBRATION OF FUEL-PIN SIMULATORS
ORNL-UT • DIAL 42
ORNL-UT/UT-33 • 40 PPS, 20 FIGS, NOV. 1976

AN INFRARED SYSTEM FOR THE RAPID EVALUATION OF FUEL PIN SIMULATORS IS DESCRIBED. DIGITAL AS WELL AS ANALOG DATA PROCESSING CAPABILITIES PROVIDE COMPUTER ANALYSIS IF DESIRED. THE VARIOUS COMPONENTS OF THE SYSTEM AND THE ERRORS CONTRIBUTED BY EACH ARE DISCUSSED, AND THE PROCEDURES DEVELOPED TO FACILITATE TESTING ARE PRESENTED. A DETAILED DESCRIPTION OF IN-HOUSE DESIGN

120401 CONTINUED

EQUIPMENT IS ALSO GIVEN. A HIGH TEMPERATURE TEST READER IS DESCRIBED, AND IT IS SHOWN THAT DATA EVALUATION MAY REVEAL ANOMALIES IN THE THERMAL BEHAVIOR OF THE SIMULANT NOT REVEALED BY THE NORMAL PROCEDURES.

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

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, PWR • SIMULATION • FUEL ELEMENTS • TESTING • TEST, NONDESTRUCTIVE • NRC-3 • NRC-2

120330

FEE 65

SEPTEMBER 1976 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT OAK RIDGE NATIONAL LABORATORY
OAK RIDGE NATIONAL LABORATORY, TENNESSEE
ORNL/NUREG/TN-40 • 21 PPS, NOVEMBER 1976

THIS PROGRESS REPORT CONTAINS SHORT (1 TO 4 PAGES) SUMMARIES OF THE ACCOMPLISHMENTS DURING THE MONTH ON THE FOLLOWING ONSITE PROJECTS: HEAVY SECTION STEEL TECHNOLOGY PROGRAM, FISSION PRODUCT BETA AND GAMMA ENERGY RELEASE, LOCA RELEASE FROM LOW FUEL, MULTIPHASE TESTS, NUCLEAR SAFETY INFORMATION CENTER, PWR BLUDDEN HEAT TRANSFER-SEPARATE EFFECTS, ZIRCALOY FUEL CLADDING COLLAPSE STUDIES, ZIRCONIUM METAL-WATER OXIDATION KINETICS, REMOVAL RELEASE AND TRANSPORT FROM LOWEN FUEL, PWR SAFETY ANALYSIS AND RESEARCH, AND DESIGN CRITERIA FOR PIPING AND NOZZLES.

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

REACTOR, LWR • NEST • ENERGY LEVEL • FISSION PRODUCT RELEASE • GROSS DATA • GROSS GAMMA • FAILURE, FUEL ELEMENT • NRC • BLUDDEN • HEAT TRANSFER • ZIRCALOY • FAILURE, CLADDING • METAL WATER REACTION • ZIRCONIUM • OXIDATION • REACTOR, LWR • REMOVAL • FISSION PRODUCT TRANSPORT • REACTOR, PWR • ACCIDENT ANALYSIS • DESIGN CRITERIA • PIPES AND PIPE FITTINGS • R AND D PROGRAM • SAFETY PROGRAM • NRC-2 • NRC-3 • NRC-5 • NRC-7 • NRC-8

120330

FEE 65

OCTOBER 1976 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT OAK RIDGE NATIONAL LABORATORY
OAK RIDGE NATIONAL LABORATORY, TENNESSEE
ORNL/NUREG/TN-40 • 23 PPS, NOVEMBER 1976

THIS PROGRESS REPORT CONTAINS SHORT (1 TO 4 PAGES) SUMMARIES OF THE ACCOMPLISHMENTS DURING THE MONTH ON THE FOLLOWING ONSITE PROJECTS: HEAVY SECTION STEEL TECHNOLOGY PROGRAM, FISSION PRODUCT BETA AND GAMMA ENERGY RELEASE, LOCA RELEASE FROM LOW FUEL, MULTIPHASE TESTS, NUCLEAR SAFETY INFORMATION CENTER, PWR BLUDDEN HEAT TRANSFER-SEPARATE EFFECTS, ZIRCALOY FUEL CLADDING COLLAPSE STUDIES, ZIRCONIUM METAL-WATER OXIDATION KINETICS, REMOVAL RELEASE AND TRANSPORT FROM LOWEN FUEL, PWR SAFETY ANALYSIS AND RESEARCH, AND DESIGN CRITERIA FOR PIPING AND NOZZLES.

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

REACTOR, LWR • NEST • ENERGY LEVEL • FISSION PRODUCT RELEASE • GROSS DATA • GROSS GAMMA • FAILURE, FUEL ELEMENT • NRC • BLUDDEN • HEAT TRANSFER • ZIRCALOY • FAILURE, CLADDING • METAL WATER REACTION • ZIRCONIUM • OXIDATION • REACTOR, LWR • REMOVAL • FISSION PRODUCT TRANSPORT • REACTOR, PWR • ACCIDENT ANALYSIS • DESIGN CRITERIA • PIPES AND PIPE FITTINGS • R AND D PROGRAM • SAFETY PROGRAM • NRC-2 • NRC-3 • NRC-5 • NRC-7 • NRC-8

120130

FEE 64

NOVEMBER 1976 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT OAK RIDGE NATIONAL LABORATORY
OAK RIDGE NATIONAL LABORATORY, TENNESSEE
ORNL/NUREG/TN-76 • 25 PPS, DEC. 1976

THIS PROGRESS REPORT CONTAINS SHORT (1 TO 4 PAGES) SUMMARIES OF THE ACCOMPLISHMENTS DURING THE MONTH ON THE FOLLOWING ONSITE PROJECTS: HEAVY SECTION STEEL TECHNOLOGY PROGRAM, FISSION PRODUCT BETA AND GAMMA ENERGY RELEASE, LOCA RELEASE FROM LOW FUEL, MULTIPHASE TESTS, NUCLEAR SAFETY INFORMATION CENTER, PWR BLUDDEN HEAT TRANSFER-SEPARATE EFFECTS, ZIRCALOY FUEL CLADDING COLLAPSE STUDIES, ZIRCONIUM METAL-WATER OXIDATION KINETICS, REMOVAL RELEASE AND TRANSPORT FROM LOWEN FUEL, PWR SAFETY ANALYSIS AND RESEARCH, AND DESIGN CRITERIA FOR PIPING AND NOZZLES.

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

REACTOR, LWR • NEST • ENERGY LEVEL • FISSION PRODUCT RELEASE • GROSS DATA • GROSS GAMMA • FAILURE, FUEL ELEMENT • NRC • BLUDDEN • HEAT TRANSFER • ZIRCALOY • FAILURE, CLADDING • METAL WATER REACTION • ZIRCONIUM • OXIDATION • REACTOR, LWR • REMOVAL • FISSION PRODUCT TRANSPORT • REACTOR, PWR • ACCIDENT ANALYSIS • DESIGN CRITERIA • PIPES AND PIPE FITTINGS • R AND D PROGRAM • SAFETY PROGRAM • NRC-2 • NRC-3 • NRC-5 • NRC-7 • NRC-8

110002

LEE REL • REEL 05

A STUDY OF DROPLET HYDRODYNAMICS IMPORTANT IN LOCA REPLENISHMENT

115000 CONTINUED
STATE UNIVERSITY OF NEW YORK, STATE COLLEGE
DAVID-DAVID-0301 • 20 PPS. 2 PAGES. APRIL 1970

THE PURPOSE OF THE THEORETICAL AND EXPERIMENTAL RESEARCH WORK REPORTED HERE IS TO INVESTIGATE THE FLUID MECHANICAL AND HEAT TRANSFER PHENOMENA THAT ARE BELIEVED TO OCCUR DURING THE REFLUX PORTION OF THE HYPOTHESIZED LOCA. WORK HAS PROCEEDED IN THE FABRICATION OF A SOLID PARTICLE/AIR TWO-PHASE FLOW SYSTEM TO SIMULATE THE FLUID MECHANICAL BEHAVIOR OCCURRING IN THE REFLUX PHENOMENON. COMPLETING THIS, THE NEXT A SHORT RANGE LADDER DOPPLER VELOCIMETER WILL BE USED WITH SURFACE BALL PROBE ANALYSIS AND SYNTHESIZING OF THE JOURNALING DOPPLER SIGNALS TO OBTAIN INFORMATION ON THE TWO-PHASE FLOW. THEORETICAL MODELS HAVE BEEN FORMULATED TO EXPLAIN THE MODELS OF DOPPLER FORMATION THAT MAY BE PRESENT IN THE REFLUX PHENOMENON.

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

DOPPLER EFFECT • HEAT TRANSFER • ACCIDENT • LOSS OF COOLANT • MEASUREMENT • PARTICLES • ANALYTICAL SIGNAL • HYDRODYNAMIC ANALYSIS • FLOW, TWO PHASE • LASER • PARTICLES • SAFETY PROGRAM • SCENE REFLUXION SYSTEM • H AND O PROGRAM • THERMAL HYDRAULIC ANALYSIS • JOURNAL

3. NRC-3 WATER REACTOR SAFETY RESEARCH, FUEL BEHAVIOR

112324

MACDONALD PE • THOMPSON LG • CLARK CS
 SUMMARY - A HANDBOOK OF MATERIALS PROPERTIES FOR USE IN THE ANALYSIS OF LIGHT WATER REACTOR FUEL ROD BEHAVIOR
 ORNL-1263 • 275 PPS, 36 TABS, 66 FIGS, FEB. 1976

DESCRIBES THE MATERIALS PROPERTIES CORRELATIONS AND COMPUTER MODELS (MATPRO) DEVELOPED FOR USE WITH VARIOUS LWR FUEL ROD BEHAVIOR ANALYTICAL PROGRAMS AT THE IDAHO NATIONAL ENGINEERING LABORATORY. DOCUMENTATION AND FORMULATIONS THAT ARE USUALLY SCATTERED IN NATURE ARE PRESENTED FOR URANIUM DIOXIDE AND MIXED URANIUM-PLUTONIUM OXIDE FUEL, ZIRCALOY CLADDING, WAS PLATE, AND LWR FUEL ROD MATERIAL PROPERTIES.

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

COMPUTER PROGRAM • THERMAL MECHANICAL EFFECT • CREEP PROPERTY • REACTION, LOW • URANIUM DIOXIDE • CORRELATION • OTHER PROPERTY • ZIRCALOY • STRESS STRAIN DATA • PLUTONIUM DIOXIDE • FUEL ROD • PLATE, MECHANICAL • MIXED OXIDE • THERMAL CONDUCTIVITY • JACOBI

120000

PAPER TC
 A FAST RESPONSE, 2.54 PSI (17.24 MPa) TRANSDUCER FOR MEASUREMENT OF GAS PRESSURE IN FUEL RODS
 IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
 NRC-NRRC-1320 • 10 PPS, 6 FIGS, 2 TABS, SEPT. 1976

A STRAIN GAGE PRESSURE TRANSDUCER OF 2,500 PSI (17.24 MPa) RANGE FOR OPERATION IN A GAS BEG F ENVIRONMENT IS DESCRIBED. SPECIFIC DESIGN PARAMETERS ARE GIVEN ALONG WITH THE CALIBRATION RESULTS OBTAINED FROM TYPICAL TRANSDUCERS. APPENDICES INDICATE THE BRIDGE OUTPUT TO BE EXPECTED AND THE ACTUAL OPEN CIRCUIT VALUE OF A STRAIN GAGE CALCULATED FROM MEASUREMENTS TAKEN WITH THE BRIDGE COMPLETED.

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

A AND B PROGRAM • SAFETY PROGRAM • REACTOR, LOW • MEASUREMENT, STRAIN GAGE • INSTRUMENT, PRESSURE • DESIGN CRITERIA • NRC-3

120000

CHARACTERISTICS OF UO₂-ZIRCALOY FUEL ROD MATERIALS FROM SARTON REACTOR FOR USE IN POWER BURST FACILITY
 IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
 NRC-NRRC-1321 • 103 PPS, 20 TABS, 66 FIGS, 20 TABS, SEPT. 1976

AS A PART OF NRC'S FUEL BEHAVIOR PROGRAM, A SERIES OF TESTS ARE BEING CONDUCTED IN THE POWER BURST FACILITY TO STUDY THE PERFORMANCE OF LWR FUEL RODS UNDER OFF-NOMINAL CONDITIONS. IT IS IMPORTANT THAT SOME OF THE TESTS UTILIZE RODS CONTAINING PREVIOUSLY IRRADIATED MATERIALS. THEREFORE, A NUMBER OF ZIRCALOY-CLAD UO₂ FUEL RODS AND ZIRCALOY CLADDING TUBES WERE OBTAINED FROM THE SARTON REACTOR. IN ORDER TO ESTABLISH THE PROPERTIES OF THE MATERIALS PRIOR TO USE IN THE POWER BURST FACILITY TEST PROGRAMS, A SERIES OF DESTRUCTIVE AND NONDESTRUCTIVE EXAMINATIONS WAS CONDUCTED ON THE MATERIALS. THESE EXAMINATIONS CONSISTED OF VISUAL EXAMINATION, NEUTRON RADIOGRAPHY, DIMENSIONAL MEASUREMENTS, GAMMA SCANNING, DEFECT SCANNING, GAS VOLUME MEASUREMENTS, GAS COMPOSITION DETERMINATION, METALLOGRAPHY, CHEMICAL ANALYSIS, AND OTHERS. THE TEST METHODS USED AND RESULTS OBTAINED ARE DESCRIBED IN THIS REPORT.

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

AGENCY, NRC • A AND B PROGRAM • FUEL ROD • ZIRCALOY • CLADDING • URANIUM DIOXIDE • TEST, DESTRUCTIVE • TEST, NONDESTRUCTIVE • SARTON (PWR) • REACTOR, PWR • IDAHO NAT ENG LAB • NRC-3

120107

HEWNER AS • QUAPP RJ • GUTZMANN C • HODGINS RA
 POSTIRRADIATION EXAMINATION RESULTS FOR THE IRRADIATION EFFECTS SCORING TEST I
 IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
 NRC-NRRC-1330 • 67 PPS, 7 TABS, 30 FIGS, 23 TABS, SEPT. 1976

A ZIRCALOY-CLAD UO₂ FUEL ROD WAS OPERATED ABOVE ITS CRITICAL HEAT FLUX DURING THE IN-PILE TEST LOOP OF THE POWER BURST FACILITY AND LATER EXAMINED IN THE HOT CELL. THE RESULTS OF THE POSTIRRADIATION EXAMINATIONS ARE PRESENTED IN THIS REPORT. A ZIR-ONE REACTION AT THE FUEL-CLADDING INTERFACE ENRITLED NEARLY AS MUCH OF THE CLADDING WALL THICKNESS AS THE ZIR-ONE REACTION ON THE EXTERIOR. DATA ON BOTH THE INTERNAL AND EXTERNAL REACTIONS, AND CLADDING AND FUEL MICROSTRUCTURES, ARE PRESENTED. CLADDING ENRITLEMENT AND FUEL FAILURE ARE COMPARED WITH SEVERAL ROD FRAGMENTATION CRITERIA, AND CONCLUSIONS CONCERNING FUEL ROD FAILURE PROPAGATION IN A POWER REACTOR SYSTEM ARE MADE.

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

ZIRCALOY • FUEL ROD • IRRADIATION TESTING • EXAMINATION • URANIUM DIOXIDE • ENRITLEMENT • FAILURE, CLADDING • NRC-3

119900

CARROO CO • RANTINSON JR • MURMAN GW • LEGRAND GL

110000 CONTINUED
POWER-COOLING-REACTOR TEST SERIES, TEST FOR-20, TEST RESULTS REPORT
IDRND NATIONAL ENGINEERING LABORATORY, IDRND FALLS
IDRND-1007-20 157 PPS, 40 TABS, 71 FIGS, SEPT. 1970

THIS REPORT DESCRIBES THE RESULTS OF AN IN-PILE EXPERIMENTAL INVESTIGATION OF PRE- AND POSTCRITICAL HEAT FLUX (CHF) BEHAVIOR OF A SINGLE 30-INCH-OD, PRESSURIZED WATER REACTOR (PWR) TYPE, GAS-FUELED, CORROSION-RESISTANT FUEL ROD. THE NOMINAL COOLANT CONDITIONS FOR PRESSURE AND TEMPERATURE WERE REPRESENTATIVE OF THOSE FOUND IN A COMMERCIAL PWR. NINE SEPARATE DEPARTURE FLUX MEASUREMENTS WERE MADE (NINE CYCLES WERE PERFORMED). RESULTS OF ON-LINE MEASUREMENTS OF THE FUEL AND DEVIATION ARE PRESENTED TOGETHER WITH DISCUSSION OF INSTANTANEOUS PERFORMANCE. A COMPARISON OF THE DATA WITH FUEL ROD ANALYSIS PROGRAM - TRANSIENT 2 (FAP-12) COMPUTED LINE CALCULATIONS IS INCLUDED.

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

ACCIDENT, LOSS OF COOLANT • ACCIDENT, LOSS OF FLUX • CHF • DESIGN • REACTION FOR • IN PILE EXPERIMENT • FUEL BEHAVIOR • COMPARISON, THEORY AND EXPERIENCE • HEAT FLUX, CRITICAL • HEAT TRANSFER, DESIGN • CHF • J

120301
DIETZ HA
QUARTERLY TECHNICAL PROGRESS REPORT ON WATER REACTOR SAFETY PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH APRIL-JUNE 1970
IDRND NATIONAL ENGINEERING LAB., IDRND FALLS
IDRND-1000-1000 117 PPS, 75 FIGS, 57 REFS, OCT. 1970

RESULTS OF A REFLUX HEAT TRANSFER TEST SERIES WERE ANALYZED TO DETERMINE THE REFLUX HEAT TRANSFER CHARACTERISTICS OF THE MINISCALE ROD-1 ELECTRICALLY HEATED CORE. A SPECIAL TEST TO INVESTIGATE THE SENSITIVITY OF DEPARTURE FROM NUCLEATE BOILING TO A CHANGE IN CORE GEOMETRY WAS CONDUCTED WITH THE MINISCALE ROD-1 FACILITY. THE LOSS-OF-COOLANT EXPERIMENTS, INCLUDING AN INVESTIGATION OF HOT-CELL DELAY EFFECTS, WERE CONCLUDED IN LATE. FUEL GEOMETRY VERIFICATION EFFORTS CONCENTRATED ON A USE AND OVER-DRIVE SHELLING MODEL. UNCERTAINTIES IN THE EXPRESSIONS FOR CLADDING EXPANSION AND SPECIFIC HEAT, MODIFICATION OF THE ZIRCALOY PLASTIC DEFORMATION MODEL, AND ANALYSIS OF RESULTS FROM STRAIN STATE AXIAL GAS FLOW EXPERIMENTS UTILIZING PWR RODS WITH ANODIZED AND MILLION CASES.

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

REACTOR, LOSS OF COOLANT • LOSS OF FLUX • CORE REFLUXING • FUEL SHELLING • HEAT FLUX • ENGINEERING SAFETY FEATURE • COMPUTER PROGRAM • CHF • H AND D PROGRAM • CHF-2 • CHF-3 • CHF-4

120304
ARNDT AE • ECKHARTER J
USE OF THIN-FILM TRANSDUCERS TRANSMISSION LINE AND SENSOR TO IMPROVE THE PERFORMANCE OF HIGH TEMPERATURE ULTRASONIC TRANSDUCERS
IDRND NATIONAL ENGINEERING LAB., IDRND FALLS
IDRND-1000-1001 27 PPS, 10 FIGS, 52 REFS, NOV. 1970

AT TEMPERATURES ABOVE APPROXIMATELY 1,000 °C THE TRANSMISSION LINE OFTEN DUE TO THE PROTECTIVE SHEATH AT CONTACT POINTS. NOISE ECHOS AND GENERATED AT THESE JUNCTION POINTS THAT, AT TIMES, PAIR OUT THE SIGNAL ECHOS. A THIN-FILM TRANSDUCER TRANSMISSION LINE AND SENSOR UPPER LOW ATTENUATION CHARACTERISTICS AT HIGH TEMPERATURES, WHICH ALLOW USE OF STANDOFFS TO PROVIDE ISOLATION FROM THE SHEATH. THE ULTRASONIC CHARACTERISTICS OF THE THIN-FILM TRANSDUCER AND THE STANDOFFS ARE DISCUSSED. ALSO, DEGRADATION DUE TO TRANSDUCERS IN A NUCLEAR ENVIRONMENTAL IS ADDRESSED.

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

TEMPERATURE • HIGH TEMPERATURE • MEASUREMENT, TEMPERATURE • ULTRASONICS • TRANSDUCER • THIN-FILM • USE OF H AND D PROGRAM • CHF-3 • FUEL ROD

114202
LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM: QUARTERLY PROGRESS REPORT JULY-DECEMBER 1970
ARGONNE NATIONAL LABORATORY, ILLINOIS
ANL-70-15 76 PPS, 7 TABS, 70 FIGS, 12 REFS, 1970

THIS PROGRESS REPORT SUMMARIZES THE ARGONNE NATIONAL LABORATORY WORK PERFORMED DURING OCTOBER, NOVEMBER, AND DECEMBER 1970 ON WATER-REACTOR SAFETY RESEARCH PROGRAMS. 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; AND (3) MECHANICAL PROPERTIES OF ZIRCALOY CONTAINING HYDROGEN.

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

REACTOR, LOSS OF COOLANT • FISSION PRODUCT RELEASE • HEAT TRANSFER • ACCIDENT, LOSS OF COOLANT • DESIGN • REACTION, FOR • ANALYTICAL MODEL • DYNAMICS, NONLINEAR • ZIRCALOY • HYDROGEN • JACOBS

110317
LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM: QUARTERLY PROGRESS REPORT JAN.-MAR. 1970

110517 UNCLASSIFIED
ARGONNE NATIONAL LABORATORY, ILLINOIS
ANL-70-65 P. 71 PPS, 12 TABS, 60 FIGS, MARCH 1970

THIS PROGRESS REPORT SUMMARIZES THE ARGONNE NATIONAL LABORATORY WORK PERFORMED DURING THE PERIOD JANUARY-MARCH 1970 ON WATER REACTOR SAFETY AND HEAT TRANSFER PROBLEMS. THE WORK IN LUSC OF COOLANT ACCIDENT RESEARCH INCLUDES DRIFT-FLUX MODELS, NITROGEN-WATER BLOWDOWN TESTS AND TRANSIENT CRITICAL HEAT FLUX TESTS. WORK IS ALSO REPORTED ON MODELS OF FUEL-FISSION PRODUCT BEHAVIOR, MECHANICAL PROPERTIES OF ZIRCALOY CONTAINING URYEN, AND STEAM EXPLOSION STUDIES.

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

ACCIDENT, LOSS OF COOLANT • ACCIDENT, LOSS OF PRESSURE • REACTION, SAFETY RESEARCH • OUT OF PILE EXPERIMENT • REACTOR, LWR • ZIRCALOY • NRC-3 • BLOWDOWN • HEAT FLUX, CRITICAL • NRC-4 • THERMAL HYDRAULIC ANALYSIS • JACOBS

120001
LIGHT WATER REACTOR SAFETY RESEARCH PROGRAM QUARTERLY PROGRESS REPORT APRIL-JUNE 1970
ARGONNE NATIONAL LABORATORY, ILLINOIS
ANL-70-67 P. 60 PPS, 2 TABS, 50 FIGS, 1970

SUMMARIZES THE ARGONNE NATIONAL LABORATORY WORK PERFORMED DURING APRIL, MAY, AND JUNE 1970 ON WATER-REACTOR-REACTOR SAFETY RESEARCH AND FLUX PROBLEMS. THE FOLLOWING RESEARCH AND DEVELOPMENT AREAS ARE COVERED: (1) LOSS-OF-COOLANT ACCIDENT RESEARCH: HEAT TRANSFER AND FLOW DYNAMICS; (2) TRANSIENT FUEL RESPONSE AND FISSION-PRODUCT RELEASE PROGRAM; (3) MECHANICAL PROPERTIES OF ZIRCALOY CONTAINING URYEN; AND (4) STEAM-EXPLOSION STUDIES.

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

REACTOR, LWR • ACCIDENT, LOSS OF COOLANT • ZIRCALOY • OUT OF PILE EXPERIMENT • COMPUTER PROGRAM • HEAT FLUX, CRITICAL • HEAT TRANSFER ANALYSIS • BLOWDOWN • FISSION PRODUCT RELEASE • TENSILE PROPERTY • NRC-3 • NRC-4

117311
PROJECT DESCRIPTION FOR FISSION PRODUCT TRANSPORT ANALYSIS
BATTELLE COLUMBUS LABS., OHIO
22 PPS, 3 FIGS, NRC, NO. 70-045, JUNE 6, 1970

STUDIES ARE CURRENTLY BEING INITIATED TO PROVIDE DATA WHICH SHOULD CONSIDERABLY IMPROVE THE DESCRIPTION OF FISSION PRODUCT RELEASE FROM FUEL RODS FOR THE TERMINATED LOCAL. IN ADDITION, EXTENSIVE THERMAL-HYDRAULIC ANALYSES OF EMERGENCY CORE COOLING SYSTEM (ECCS) OPERATION HAVE PRODUCED CONSIDERABLE DATA ON FLOW PATTERN AND OTHER CONDITIONS THROUGHOUT THE PRIMARY SYSTEM LOOP FOR THE TERMINATED LOCAL. THEREFORE, IT IS APPROPRIATE AT THIS TIME TO DEVELOP A PRIMARY SYSTEM FISSION PRODUCT TRANSPORT MODEL WHICH IS CONSISTENT WITH THE EMERGENCY STATE OF KNOWLEDGE SO THAT MORE REALISTIC CALCULATIONS OF THE RATE AND MAGNITUDE OF FISSION PRODUCT RELEASE TO THE CONTAINMENT CAN BE PERFORMED.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1737 M STREET, BETHESDA, D. C. 20814 100 CENTS/PAGE -- MINIMUM CHARGE \$2.00

REACTOR, BWR • CHEMICAL REACTION • DEPOSITION • FUEL MELTOUT • ACCIDENT, LOSS OF COOLANT • REACTOR, PWR • SHIPPING CONTAINER • FISSION PRODUCT TRANSPORT • NRC-2 • THERMAL HYDRAULIC ANALYSIS

117005
GARDEN AA • LARRY LN • PERLIN JS
EVALUATING STRENGTH AND DUCTILITY OF IRRADIATED ZINCUMIN QUARTERLY PROGRESS REPORT APRIL THROUGH JUNE 1970
BATTELLE COLUMBUS LABS., OHIO
OBI-MUREC-1950 P. 27 PPS, 4 TABS, 10 FIGS, JULY 1970

DIAMETER AND TUBE-WALL THICKNESS REDUCTIONS WERE MEASURED ON SPECIMENS TENSILE TESTED AT 600, 700, AND 800 P AND AT VARIOUS STRAIN RATES AT 700 P. REDUCTIONS IN CROSS-SECTIONAL AREA HAVE BEEN CALCULATED. A SLIGHT MINIMUM IN REDUCTION OF AREA OCCURS AT 700 P WITH THE VALUE AT 600 P BEING SIGNIFICANTLY GREATER THAN AT EITHER 600 OR 700 P. THE SMALLEST REDUCTION OF AREA OCCURRED IN A HIGH STRAIN-RATE TEST AT 700 P. THE DEPENDENCE OF AXIAL TENSILE PROPERTIES ON THE ANNEALING OF IRRADIATION DAMAGE WAS INVESTIGATED. AN AGING PEAK IN STRENGTH OCCURRED DURING A 50 P/SEC TRANSIENT TO 1000 P. DUCTILITY, AS MEASURED BY TOTAL TENSILE ELONGATION, DECREASED WHILE THE STRENGTH DECREASED, INCREASING ABOVE THE POSTIRRADIATION VALUE ONLY AFTER STRENGTH CHANGES HAD ESSENTIALLY STABILIZED.

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

TENSILE PROPERTY • ZINCUMIN • IRRADIATION TESTING • NRC-3 • R AND D PROGRAM • DUCTILITY • JACOBS • NRC-2

110000
GIESBRE JA • DAYBUTT P • MERRING PS • JORDAN H • WOOTEN RJ
FISSION PRODUCT TRANSPORT ANALYSIS
BATTELLE COLUMBUS LABS., OHIO
OBI-MUREC-1955 P. 37 PPS, 4 TABS, 4 FIGS, 10 REFS, SEPT. 7, 1970

ANALYSIS METHODS AND COMPUTER MODELS FOR PREDICTING FISSION PRODUCT TRANSPORT AND DEPOSITION UNDER POSTULATED LOSS OF COOLANT ACCIDENT CONDITIONS WITHIN WATER-COOLED REACTOR PRIMARY SYSTEMS AND SHIPPING CONTAINERS ARE BEING DEVELOPED. EXISTING THERMAL-HYDRAULIC ANALYSES (INCLAP-2) WITH THE

11000 CONTINUED

BASES FOR DEFINING THE CONDITIONS DURING THE PRIMARY SYSTEM AND THE TRANSPORT FROM CORE FISSILE PRODUCT RELEASE. THE ANALYSIS TO BE DEVELOPED IN THIS STUDY WILL PROVIDE METHODS FOR MORE REALISTICALLY CALCULATING THE RATE AND MAGNITUDE OF FISSION PRODUCT RELEASE TO THE CONTAINMENT.

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

REACTION, COR + DEPOSITION + ACCIDENT, LOSS OF COOLANT + REACTION, COR + ANALYTICAL MODEL + FISSION PRODUCT TRANSPORT + AEC-J + AEC-N + THERMAL HYDRAULIC ANALYSIS + JACOB

110933

COLLINS J. + GOSWAMI M. + M. CHAKRAS AB
KINETIC ANALYSIS OF CELL-BASED SPECTROMETER STUDIES OF CESIUM-137-AMMONIUM INTERACTIONS
JOHN RICE NATIONAL LABORATORY, TENNESSEE
ORNL/PHUREG/TH-24 P. 13 PPS, 2 FIGS, 10 REFS, JUNE 1970

KINETIC ANALYSIS OF CELL-BASED SPECTROMETER STUDIES HAVE BEEN MADE OF THE PARTIAL PRESSURES OF CESIUM-CONTAINING SPECIES (AMMONIUM) TO BE PRIMARILY CAGED UPON CATIONIC AND ANIONIC SPECIES INVOLVED IN THE REACTION OF CESIUM-137-AMMONIUM COMPOUNDS FORMED FROM REACTIONS THAT INITIALLY CONTAINED EITHER CESIUM-137 OR CESIUM-134. ALTHOUGH ADDITIONAL WORK IS REQUIRED TO FURTHER DEFINE THE KINETICS INVOLVED, THE DATA DEMONSTRATE UNAMBIGUOUSLY A SIGNIFICANT REDUCTION IN CESIUM PARTIAL PRESSURES DUE TO AMMONIUM CESIUM-137 COMPOUND FORMATION AND INDICATE ESSENTIALLY IDENTICAL BEHAVIOR WITH EITHER CESIUM OR CESIUM-134 AS THE STARTING MATERIAL WITH CESIUM.

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

REACTION + CHEMICAL REACTION + ANALYTICAL TECHNIQUE + CHEMICAL EQUILIBRIUM + REACTION, COR + SPECTROMETER, MASS + JACOB

11300

PERKINS BA
ZIRCONIUM METAL-WATER OXIDATION KINETICS II. OXYGEN-16 DIFFUSION IN BETA-ZIRCONIUM
JOHN RICE NATIONAL LABORATORY, TENNESSEE
ORNL/PHUREG/TH-19 P. 01 PPS, TABS, FIGS, JULY 1970

AT THE END OF A HYPOTHETICAL LOSS-OF-COOLANT ACCIDENT IN A LIGHT WATER REACTOR THE MECHANICAL PROPERTIES OF THE "HOT" BETA REGION OF THE ZIRCONIUM FUEL CAND WILL BE DETERMINED TO A LARGE EXTENT BY ITS OXYGEN CONTENT. FOR THIS REASON THE DIFFUSION OF OXYGEN-16 IN BETA-ZIRCONIUM HAS BEEN MEASURED FROM 900 TO 1500 DEG C (1600-2732 DEG F), AND TEMPERATURE MEASUREMENTS HAVE BEEN MADE FOR BETA-ZIRCONIUM AND BETA-ZIRCONIUM-2. THE THICK DIFFUSIVITY HAS BEEN MEASURED OVER THE ENTIRE TEMPERATURE RANGE AND THE CHEMICAL DIFFUSIVITY FROM 1100 TO 1500 DEG C (2012-2732 DEG F). THE EXPERIMENTS WERE PERFORMED BY USING OXYGEN-16 AS THE TRACER AND ACTIVATING IT BY PROTON IRRADIATION.

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

DIFFUSION + MEASUREMENT + METAL-WATER REACTION + OXIDATION + BETA-ZIRCONIUM + HIGH TEMPERATURE + CHEMICAL KINETICS + REACTION, COR + THERMAL PROPERTY + ZIRCONIUM + SAFETY PROGRAM + AEC-J + N AND D PROGRAM + JACOB

117205

LORENZ BA + GOSWAMI M. + COLLINS J. + BARNETT M. + CHAKRAS AB
BEHAVIOR OF IODINE, METHYL IODIDE, CESIUM IODIDE, AND CESIUM IODINE IN STEAM AND AMMONIA
JOHN RICE NATIONAL LABORATORY, TENNESSEE
ORNL/PHUREG/TH-25 P. 07 PPS, 7 TABS, 10 FIGS, 31 REFS, JULY 1970

AN EXPERIMENTAL PROGRAM HAS BEEN INITIATED WITH THE OVERALL GOAL OF PROVIDING A DEFINITIVE UNDERSTANDING OF FISSION PRODUCT BEHAVIOR UNDER CONDITIONS APPLICABLE TO LIGHT WATER REACTOR ACCIDENTS IN WHICH THE FUEL RODS REACH A MAXIMUM TEMPERATURE IN THE RANGE 900 TO 1500 DEG C. IN ORDER TO TEST THE APPARATUS AND TECHNIQUES OF ANALYSIS, A SERIES OF TEN EXPERIMENTS WAS PERFORMED IN WHICH VARIOUS FISSION PRODUCT-TYPE COMPOUNDS WERE EXPOSED TO HIGH-TEMPERATURE STEAM AND AMMONIA ATMOSPHERES. THE SPECIES INVESTIGATED WERE I₂, CH₃I, C₂H₅I, AND CS₂, MAINLY IN A STEAM ATMOSPHERE OVER THE TEMPERATURE RANGE 900 TO 950 DEG C.

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

REACTION + DEPOSITION + FISSION PRODUCT RELEASE + IODINE + IODINE + STEAM + AMMONIA + ORGANIC IODINE + OFF GAS + FISSION PRODUCT RETENTION + AEC-J + N AND D PROGRAM + JACOB

115000

CHAPMAN BA
CHARACTERIZATION OF ZIRCONIUM TUBING PROCURED FOR FUEL CLEANING RESEARCH PROGRAMS
JOHN RICE NATIONAL LABORATORY, TENNESSEE
ORNL/PHUREG/TH-29 P. 131 PPS, TABS, FIGS, REFS, JULY 1970

A QUANTITY OF ZIRCONIUM TUBING (10.00 IN (25.40 CM) INSIDE DIAMETER BY 0.035 IN (0.89 CM) WALL THICKNESS) WAS PURCHASED SPECIFICALLY FOR USE IN A NUMBER OF RELATED FUEL CLEANING RESEARCH PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY COMMISSION. THE TUBING IS REPRESENTATIVE

115000 CONTINUED

OF CURRENT REACTOR TECHNOLOGY. THIS REPORT DOCUMENTS THE PROGRESS ACTIVITIES, PROVIDES A CONVENIENT REFERENCE SOURCE OF INFORMATION'S DATA AND FURTHER DISTRIBUTION TO THE VARIOUS USERS, AND PRESENTS SOME PRELIMINARY CHARACTERIZATION DATA.

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

PUBLICATION - PLANNING - REACTION, FUEL - MATERIAL - ZIRCALOY - SAFETY PROGRAM - R AND D PROGRAM - EQUIPMENT DESCRIPTION

115250

LORENZ RO • COLLINS JR • MANNING SM

MONTHLY PROGRESS REPORT ON FISSION PRODUCT RELEASE FROM LOW FUEL FOR THE PERIOD JANUARY-MARCH 1976
OAK RIDGE NATIONAL LABORATORY, TENNESSEE

ORNL/MEET/TM-30 • 29 PPS, 10 FIGS, 6 REFS, JULY 1976

DETAILED DATA ARE PRESENTED FOR IMPLANT TESTS 4-7, AND PRELIMINARY DATA ARE GIVEN FOR IMPLANT TEST 8. THE ZIRCALOY-4 CLADDING CONTINUES TO BE AN EFFECTIVE BARRIER TO THE TELLURIUM, EXCEPT IN THE CASE OF THE IMPLANT TEST 5, IN WHICH CONSIDERABLE TELLURIUM SPECIES APPEARS TO HAVE BEEN EJECTED WITH AND RUPTURE. JUNCTION OF THE FUEL AND SUBSEQUENT SWELLING OF THE FUEL WERE OBSERVED IN IMPLANT TEST 7, WHICH WAS CONDUCTED IN AN AIR ATMOSPHERE.

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

DESIGN - FISSION PRODUCT RELEASE - STEAM - ARGON - ORGANIC FLUIDS - TEMPERATURE GRADIENT - THERMAL CONDUCTION - THERMAL PROPERTY - SAFETY PROGRAM - R AND D PROGRAM - JUDICIAL - JUDICIAL

115301

PER 66

JUNE 1976 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PREPARED AT OAK RIDGE NATIONAL LABORATORY

OAK RIDGE NATIONAL LABORATORY, TENNESSEE

ORNL/MEET/TM-30 • 27 PPS, JULY 1976

THIS PROGRESS REPORT CONTAINS SHORT 11 TO 4 PAGES SUMMARIES OF THE ACCOMPLISHMENTS DURING THE MONTH ON THE FOLLOWING ONSITE PROJECTS: HEAVY SECT- STEEL TELLURIUM PROGRAM, FISSION PRODUCT DATA AND CORROSION RELEASE, LEAK RELEASE FROM LOW FUEL, MATERIALS TESTS, NUCLEAR SAFETY INFORMATION CENTER, FUEL OXIDATION HEAT TRANSFER-SAMPLE EFFECTS, ZIRCALOY FUEL CLADDING CORROSION STUDIES, ZIRCONIUM METAL-WATER OXIDATION KINETICS, OXIDATION RELEASE AND TRANSPORT FROM LOW FUEL, OTHER SAFETY ANALYSIS AND RESEARCH, AND DESIGN CRITERIA FOR PIPING AND NOZZLES.

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

ACCIDENT ANALYSIS - AEROSOL - DESIGN CRITERIA - FISSION PRODUCT RELEASE - FAILURE - FUEL ELEMENT - HEAT TRANSFER - ACCIDENT, LOSS OF COOLANT - METAL WATER REACTION - OXIDATION - R AND D PROGRAM - FISSION PRODUCT TRANSPORT - GROSS DATA - GROSS CORROSION - REACTION, LOW - FAILURE, CLADDING - ZIRCALOY - SAFETY PROGRAM - OXIDATION - NUC - ENERGY LEVEL - R AND D PROGRAM - REACTION, HIGH - REACTION, LOW - MOST - PIPES AND PIPE FITTINGS

120002

BRONSON DE • DIAL RE

AN APPARATUS FOR SPOT WELDING SHEATHED THERMOCOUPLES TO THE INSIDE OF SMALL-DIAMETER TUBES AT PRECISE LOCATIONS
OAK RIDGE NATIONAL LABORATORY, TENNESSEE

ORNL/MEET/TM-33 • 17 PPS, 9 FIGS, AUG. 1976

THIS REPORT DESCRIBES EQUIPMENT AND PROCEDURES USED TO SPOT WELD TANTALUM- OR STAINLESS-STEEL-SHEATHED THERMOCOUPLES TO THE INSIDE DIAMETER OF ZIRCALOY TUBING TO MEET THE REQUIREMENTS OF THE MULTISTEP WIRE TEST (MWT) PROGRAM AT ORNL. SPOT WELDING AND WIRE LEAVING TUBES WERE FABRICATED TO REMOVE THE OXIDE COATING ON THE ZIRCALOY TUBING AT LOCAL AREAS AND SPOT WELD FROM THERMOCOUPLES SEPARATED CIRCUMFERENTIALLY BY 90 DEGREES AT ANY ANGULAR DISTRIBUTION DESIRED. IT WAS FOUND NECESSARY TO APPLY A NICKEL COATING TO STAINLESS-STEEL-SHEATHED THERMOCOUPLES TO OBTAIN ACCEPTABLE WELDS. THE MATERIAL AND SHAPE OF THE INNER ELECTRODE AND RESISTANCE BETWEEN INNER AND OUTER ELECTRODES WERE FOUND TO BE CRITICAL PARAMETERS IN OBTAINING ACCEPTABLE WELDS.

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

R AND D PROGRAM - SAFETY PROGRAM - REACTION, LOW - STEEL, STAINLESS - TANTALUM - THERMOCOUPLE - STUDIES - WELDING - EQUIPMENT DESIGN - ZIRCALOY - TESTING - NICKEL - COATING - WELD

119505

CAHILLANT JV

MONTHLY PROGRESS REPORT ON THE ZIRCONIUM METAL-WATER OXIDATION KINETICS PROGRAM SPONSORED BY THE DIVISION OF REACTOR SAFETY RESEARCH FOR APRIL-JUNE 1976

OAK RIDGE NATIONAL LABORATORY, TENNESSEE

ORNL/MEET/TM-31 • 43 PPS, 15 FIGS, 16 REFS, AUG. 1976

EXCEPT FOR SCOPING TESTS OF THE INFLUENCE OF STEAM PRESSURE ON THE OXIDATION BEHAVIOR OF ZIRCALOY-4, THE EXPERIMENTAL PHASE OF THE REACTION RATE STUDIES WAS COMPLETED. KINETIC DATA FOR THE ISOTHERMAL STEAM-ZIRCALOY REACTION BETWEEN 900 AND 1500 °C (1652-2732 °F) ARE REPORTED, AND THE

[illegible]

1154

U.S. AGENCY - NATIONAL TECHNICAL INFORMATION SERVICE, 525 NORTH DULLES ROAD, ARLINGTON, VIRGINIA 22204

815043

ADDAALDITF - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, MA 01104

110937

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, NTIS, SPRINGFIELD, IL 62814

837627

THE OBJECTIVES OF THIS PROGRAM ARE TO DELIVER THE INFORMATION CONTAINED ON UNCLASSIFIED, ALREADY CLASSIFIED, OR UNCLASSIFIED BUT FUTURELY CLASSIFIED DOCUMENTS TO THE APPROPRIATE AGENCIES AND TO PROVIDE A DATA

119027

BASE TO A PRELIMINARY ASSESSMENT OF THE MAGNITUDE AND DISTRIBUTION OF GEOMETRIC CHANGES IN THE FUEL ROD CLADDING IN A MULTIPHASE AREA AND OF THE EXTENT OF FUEL CHANNEL DEGRADATION THAT MIGHT RESULT. THIS REPORT SUMMARIZES THE PROGRESS MADE DURING THE REPORT PERIOD IN DESIGN AND CONSTRUCTION OF COMPONENTS AND SYSTEMS, DEVELOPMENT OF TESTS AND EVALUATIONS, AND PRELIMINARY OF CONSTRUCTIVE ITEMS, AND DISCUSSES PRELIMINARY RESULTS OF STRUCTURAL TESTS.

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

FAILURE • CLADDING • MATERIALS • LOSS OF COHESION • TESTING • DEGRADATION • ZIRCALOY • FUEL CHANNEL • FUEL ROD • JACOBI

119375

DECKERS, J. • COLE, T. A. • MCNEEL, J. D.
FISSION PRODUCT DATA AND GAMMA ENERGY RELEASE QUARTERLY PROGRESS REPORT FOR APRIL-JUNE 1976
JAN. RIDGE NATIONAL LABORATORY, TENNESSEE
JNL/RUREG/TR-47 • 23 PAGES, 2 TABLES, 6 FIGURES, DEPT. 1976

SEVERAL IMPORTANT ASPECTS OF THE OVERALL PROGRAM TO DETERMINE FISSION-PRODUCT ENERGY RELEASE RATES USING SPECTROSCOPIC METHODS ARE PRESENTED. THESE ASPECTS INCLUDE: (1) STATUS OF CURRENT STATUS OF FISSION GASES AND THE IMPACT UPON THE MEASURED ENERGY RELEASE RATES; (2) IMPROVEMENT IN DATA TRANSFER FROM THE PVP-15 DATA-ACQUISITION COMPUTER TO THE PVP-13 DATA-ANALYSIS COMPUTER; AND (3) EFFICIENT CALCULATION OF THE PVP-13 OR GASES DETECTION. IN ADDITION, THE FINAL DATA-TAKING FOR PVP-15 DATA ENERGY RELEASE WAS COMPLETED.

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

FISSION GAS RELEASE • FISSION PRODUCT RELEASE • ANALYTICAL TECHNIQUE • SPECTROSCOPY • GAMMA • PVP-13

119477

PAERL, R. E.
ZIRCONIUM METAL-WATER OXIDATION KINETICS. III - OXYGEN DIFFUSION IN WATER AND ALPHA ZIRCONIUM PHASE
JAN. RIDGE NATIONAL LABORATORY, TENNESSEE
JNL/RUREG-5 • 25 PAGES, FIGURE 1, 1976

THE REACTION OF ZIRCONIUM IN STEAM AT ELEVATED TEMPERATURE INVOLVES TWO MAIN PHASES OF OXIDATION: (1) OXIDE AND (2) ZIRCONIUM METAL. ALPHA ZIRCONIUM FROM THE PARENT METAL PHASE. IN THIS PAPER, EQUATIONS WILL BE DEVELOPED THAT EXPRESS THE OXYGEN DIFFUSION COEFFICIENTS IN THE OXIDE AND ALPHA PHASES IN TERMS OF THE REACTION RATE CONSTANTS AND EQUILIBRIUM SOLUBILITY VALUES. THESE EQUATIONS WERE APPLIED TO RECENT EXPERIMENTAL KINETIC DATA ON THE STEAM OXIDATION OF ZIRCONIUM-90 TO DETERMINE THE EFFECTIVE OXYGEN DIFFUSION COEFFICIENTS IN THESE PHASES OVER THE TEMPERATURE RANGE 1500-1800 K.

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

DIFFUSION • MEASUREMENT • METAL WATER REACTION • OXIDATION • OXYGEN • ZIRCONIUM • OXIDATION KINETICS • REACTION • LAW • THERMAL PROPERTY • ZIRCONIUM • SAFETY PROGRAM • PVP-13 • PVP-15 • PVP-16 • TEMPERATURE • JACOBI

119500

MOORE, D. D.
QUARTERLY PROGRESS REPORT ON THE CREEP AND COLLAPSE OF ZIRCALOY FUEL CLADDING PROGRAM SPONSORED BY THE DIVISION OF REACTOR SAFETY RESEARCH FOR JANUARY-MARCH 1976
JAN. RIDGE NATIONAL LABORATORY, TENNESSEE
JNL/RUREG/TR-51 • 9 PAGES, 6 FIGURES, OCT. 1976

THE CREEP AND COLLAPSE STUDY ON ZIRCALOY FUEL CLADDING IS CONCERNED WITH THE DEFORMATION BEHAVIOR OF CLADDING UNDER NORMAL AND NEAR-NORMAL REACTOR OPERATING CONDITIONS. EFFORTS THIS PAST QUARTER HAVE CONTINUED IN TWO PHASES OF THE PROGRAM: (1) COMPLETION OF COLLAPSE TESTING AND (2) DEVELOPMENT AND FABRICATION OF DEFORMATION-MINIMIZING EQUIPMENT. SUFFICIENT COLLAPSE TESTING HAS BEEN DONE TO COMPLETE THE ANALYSIS OF THE TEST DATA FOR NEARLY PERFECT TUBING. IT IS APPARENT THAT ALL THREE OF THE INDEPENDENT VARIABLES USED IN THIS STUDY (PELLET-TO-PELLET GAP, PELLET-TO-CLADDING GAP, AND TEMPERATURE) PLAY MAJOR ROLES IN THE COLLAPSE PHENOMENON. A COLLAPSE MODEL IS PRESENTED THAT DESCRIBES THE COLLAPSE PRESSURE AS A FUNCTION OF TEMPERATURE, ROUGH TEMPERATURE, MINOR-SCALE HARDNESS, AND PELLET-TO-PELLET GAP.

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

FAILURE • CLADDING • REACTION • LAW • ZIRCALOY • SAFETY PROGRAM • PVP-13 • PVP-15 • PVP-16 • FULL ROD • JACOBI

119571

MOORE, D. D.
QUARTERLY PROGRESS REPORT ON THE CREEP AND COLLAPSE OF ZIRCALOY FUEL CLADDING PROGRAM SPONSORED BY THE DIVISION OF REACTOR SAFETY RESEARCH FOR APRIL-JUNE 1976
JAN. RIDGE NATIONAL LABORATORY, TENNESSEE
JNL/RUREG/TR-52 • 9 PAGES, 6 FIGURES, OCT. 1976

WORK HAS CONTINUED THIS PAST QUARTER ON THE DEFORMATION BEHAVIOR OF ZIRCALOY CLADDING UNDER NORMAL EXTERNAL PRESSURE. THE EXPERIMENTAL COLLAPSE MODEL PRESENTED LAST QUARTER WAS STATISTICALLY REFINED TO PROVIDE A BETTER FIT TO THE DATA. INSTABILITIES IN THE ENVIRONMENT INSTRUMENT RELATED TO

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, 505 BRIDGEMAN ST. SPRINGFIELD, VA. 22104

FAIRFORD • CLARK • HILL • LEE • MILLER • ROBERTSON • WALKER • YOUNG •

117072

FEE \$6

AUGUST 1970 MONTHLY RENTALS FOR OFFICE OF "WILLARD" IN CHICAGO, ILLINOIS, AS ONE OF THE "WILLARD"

ONE RICE BUILDING LABORATORY, STAND 202E

DECLASSIFIED BY: 6032
DATE: 10/26/2001

PRESENTS HIGHLIGHTS OF THE FOLLOWING PROGRAMS AND ACTIVITIES: HEAVY METAL FROM TECHNOLOGY; FISSION-PRODUCT WASTE AND WASTE ENERGY RELEASE; LOCAL RELEASE FROM LOW FUEL; MATERIALS QUALITY TESTS; NUCLEAR SAFETY INFORMATION LEADERS FOR SHUT DOWN HEAT TRANSFER-DEPENDENT EFFECTS; GLOBAL FUEL CLASSIFICATION STUDIES; ZIRCONIUM METAL-ORGANIC CHEMISTRY KINETICS; REMEDIATION RELEASE AND TRANSPORT FROM LOW FUEL; NUCLEAR SAFETY ANALYSES AND RESEARCH; DESIGN CRITERIA FOR PIPING AND REACTORS; AND NEW CONVENTIONAL SAFETY FOR EVALUATION OF NUCLEAR REACTORS.

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

ANALYSIS • DESIGN CRITERIA • LOADING • FISHING PRODUCT RELEASE • METAL DATA REDUCTION • NOC-2 • DUNE
CALCULATION • INFORMATION • WAVE • NOC-3 • REMOTE • NOC-3 • NOC • 2 AND 3 JOURNAL • REACTION REPORT • NOC-6
AGENCY • NOC • NOC-7 • NOF • OREIL AND PIPE FITTING

120001

DISPATCH ON - INVESTIG 10 - CLOSING

LEARNER INSPECTION AND VALIDATION OF FULL-PIN SIMULATIONS

305 HICKS RD/10000 LAUREL AVE, GREENSBORO

1984/1985/85-86 : 04 223 20 2163, 227, 1270

AN INFRARED SYSTEM FOR THE RAPID EVALUATION OF PULP PINK SIMULATIONS IS DESCRIBED. DIGITAL AS WELL AS ANALOG DATA PROCESSING CAPABILITIES PERMITS COMPARISON ANALYSIS OF DESIGN. THE VARIOUS ELEMENTS OF THE SYSTEM AND THE ERRORS CONTINGENT OF EACH ARE DISCUSSED. AND THE PROCEDURES DEVELOPED TO FACILITATE DESIGN ARE PRESENTED. A DETAILED DESCRIPTION OF EQUIPMENT REQUIRED EQUIPMENT IS ALSO GIVEN. A HIGH TEMPERATURE TEST RUNNER IS DESCRIBED, AND IT IS SHOWN THAT SUCH EVALUATION MAY REVEAL ANOMALIES IN THE PHYSICAL BEHAVIOR OF THE SIMULATION NOT REVEALED BY THE USUAL PROCEDURES.

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

4 AND 5 PROGRAM = ARMY PROGRAM = MORTON, PHIL = SIMULATION = OTHER ELEMENTS = TESTS = STRID.
UNRESTRICTIVE = UN-1 = UN-2

120320

FREE 66

SEPTEMBER 1970 APPROXIMATELY 10 MILLIONS PER OFFICE OF NUCLEAR REGULATORY RESEARCH PROVIDED AT JAN 1968 REVISION

THE UNITED STATES DEPARTMENT OF THE INTERIOR

[illegible]

THIS PROJECT'S REPORT CONTAINS SHORT (1 TO 4 PAGE) SUMMARIES OF THE ACCOMPLISHMENTS DURING THE MONTH IN THE FOLLOWING CORE PROJECTS: REACT SECTION STEEL TREATMENT PROGRAM, PISONI PROJECT DATA AND WASTE TREAT RELEASE, LOCA RELEASE FROM LOW PWR, RADIATION AND TESTS, NUCLEAR SAFETY INFORMATION CENTER, NON-NUCLEAR HEAT TRANSFER-SERVATE EFFECTS, LIQUID FUEL CANNING WALL PSE STUDIES, CIRCULAR METAL-ROOF ORIENTATION KINETICS, AGRISON RELEASE AND TRANSPORT FROM LOW PWR, NUC SAFETY ANALYSIS, AND RESEARCH, AND DESIGN CRITERIA FOR PIPING AND MODELS.

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

REACTION, LOW + MOST + ENERGY LEVEL + FISSION PRODUCT RELEASE + JUDGES DATA + JUDGES JUDGES + FAILURE, FULL
ELEMENT + MSIL + UNCOMMON + HEAT TRANSPORT + ILL-CALCUL + FAILURE, CLADDING + RETAIL + OTHER REACTION + INFORMATION
ORIGIN + REACTION, LOW + 4 + ACTION + FISSION PRODUCT MANAGEMENT + REACTION, MSIL + ACCIDENT ANALYSIS + ORIGIN
CRITERIA + PLOTS AND PLOTS FITTING + ON AND OFF PLOTS + GASEOUS PHASE + MSIL-2 + MSIL-3 + MSIL-5 + MSIL-7 + MSIL-8

1203 JV

FREE 66

[illegible]

JAMES GIBBS WALLACE COMPANY, INC., LANSING, MI

REF/WH46/IN-60 9, 23 PPS, NOVEMBER 1976

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 2216.

120300

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

12-0307

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. GOVERNMENT PRINTING OFFICE, SPRINGFIELD, VA. 22161

120446

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

120552

[illegible]

120052 (CONTINUED)

THE TESTS IN THE LUG LUNAR TEST SERIES WERE PERFORMED IN A STEAM ATMOSPHERE. THE TEST FUEL CAPSULES WERE IRRADIATED TO APPROXIMATELY 1000 MW/M² IN THE GENERAL ELECTRIC TEST REACTOR (GTR) IN OCTOBER 1967. IN A 2 HR TEST AT 700 DEG C THE PERCENT OF TOTAL INVENTORIES RELEASED WAS AS: 11.7% I₂, 11.7% I₁, AND CS-137, 0.0008%. IN A 2 HR TEST AT 700 DEG C THE PERCENT RELEASED WAS AS: 20.2% I₂, 19.0% I₁, AND CS-137, 0.12%. THE I₂ AND I₁ NUMBERS WERE FROM FISSION GAS RELEASE IN THE GAP SPACE DURING WITH IRRADIATIONS. THE CS-137 WAS RELEASED FROM THE CAPSULE DURING THE HEAT TEMPERATURE TEST.

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

A AND U PROGRAM • SAFETY PROGRAM • REACTOR, LUN • DESIGN • FISSION PRODUCT RELEASE • FISSION GAS RELEASE • FISSION • DESIGN • GTR (1H) • NUC-3

120130

PER 66

NOVEMBER 1976 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT ORNL. NATIONAL LABORATORY
ORNL RICHIE NATIONAL LABORATORY, TENNESSEE
ORNL/NUREG-76 • 23 PAGES, DEC. 1976

THIS PROGRAM REPORT CONTAINS SHORT (1 TO 6 PAGES) SUMMARIES OF THE ACCOMPLISHMENTS DURING THE MONTH ON THE FOLLOWING CORE PROJECTS: HEAVY SECTION STEEL TECHNOLOGY PROGRAM, FISSION PRODUCT BETA AND GAMMA ENERGY RELEASE, LOCAL RELEASE FROM LUN FUEL, MULTIPHASE FLOW TESTS, NUCLEAR SAFETY INFORMATION CENTER, FOR DESIGNING HEAT TRANSFER-SEPARATE EFFECTS, ZIRCALOY FUEL CLADDING COLLAPSE STUDIES, ZIRCALOY METAL-WATER OXIDATION KINETICS, AEROSOL RELEASE AND TRANSPORT FROM LUN FUEL, HIGH SAFETY ANALYSIS AND RESEARCH, AND DESIGN CRITERIA FOR PIPING AND NOZZLES.

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

REACTOR, LUN • NSSI • ENERGY LEVEL • FISSION PRODUCT RELEASE • GROSS BETA • GROSS GAMMA • FUEL, FULL ELEMENT • NSIC • OXIDATION • HEAT TRANSFER • ZIRCALOY • FAILURE, CLADDING • METAL WATER REACTION • ZIRCALOY • OXIDATION • REACTION, LUN • DESIGN • FISSION PRODUCT TRANSPORT • HEAT, HIGH • ACCIDENT ANALYSIS • DESIGN CRITERIA • PIPES AND PIPE FITTINGS • A AND U PROGRAM • SAFETY PROGRAM • NUC-2 • NUC-3 • NUC-5 • NUC-7 • NUC-8

117003

NUREG-0076

TRANSIENT NATURAL CONVECTION IN AN INTERNALLY HEATED FLUID LAYER TOPICAL REPORT, JUNE 1975-JUNE 1976
THE OHIO STATE UNIVERSITY, COLUMBUS
NUREG-0076 • 62 PAGES, 22 FIGS, 20 PAGES, AUG. 1976

THE TIME REQUIRED FOR THE DEVELOPMENT OF THE MAXIMUM TEMPERATURE DIFFERENCE IN THE LAYER WITH INTERNAL GENERATION IS CORRELATED WITH THE NAYLEIGH NUMBER BY $FOUR U = 11.577 NAYLEIGH^{-0.133}$. THE TIME REQUIRED FOR THE COMPLETE DECAY OF THE MAXIMUM TEMPERATURE DIFFERENCE IN STEADY CONVECTION AT A GIVEN NAYLEIGH NUMBER WHEN INTERNAL ENERGY GENERATION IS SUDDENLY STOPPED IS GIVEN BY $FOUR U = 11.577 NAYLEIGH^{-0.215}$. IN BOTH OF THESE EQUATIONS, $FOUR U$ IS THE PARNELL NUMBER FOR THE LAYER, $ALPHA T_L$ IS, AND HA IS THE NAYLEIGH NUMBER BASED ON THE CONDUCTION TEMPERATURE DIFFERENCE WITH INTERNAL ENERGY GENERATION. THESE EQUATIONS WILL FIND GENERAL APPLICATION IN MOST ACCIDENT REMOVAL PROBLEMS IN HEAT EXCHANGER HEATINGS AND PARTICULARLY IN THE ANALYSIS OF THE SMALL-TIME THERMAL RESPONSE OF IN-VESSEL AND EX-VESSSEL HEAT EXCHANGER DEVICES.

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

REACTOR, LUN • HEAT TRANSFER, NATURAL CONVECTION • HEAT GENERATION, INTERNAL • EXPERIMENT • SAFETY PROGRAM • NUC-3 • A AND U PROGRAM • UNSTEADY STATE • THERMAL TRANSIENT • JACOBS

119007

SPINRAD 21

EVALUATION OF FISSION PRODUCT AFTER-HEAT, QUARTERLY REPORT JANUARY 1-MARCH 31, 1976
OREGON STATE UNIVERSITY, CUMVALLIS
NUREG-0010-2 • 37 PAGES, 7 FIGS, 9 PAGES, AUG. 1976

REPORTED HERE ARE STUDIES ON: ESTIMATION OF FISSION-PRODUCT GAMMA SPECTRA; COMPARISONS OF THEORETICAL PREDICTIONS OF FISSION-PRODUCT DECAY POWER WITH EXISTING EXPERIMENTAL PROGRAMS; AND A PRELIMINARY ESTIMATION OF DECAY ENERGY ESTIMATES FOR FISSION PRODUCTS WHICH HAVE NOT BEEN EXPERIMENTALLY DETERMINED. IN ADDITION, BRIEF PROGRAM STATEMENTS ARE ENTERED FOR CONTINUING ITEMS OF WORK WITH PENTANA TO THE GOAL OF THIS PROJECT AT OSU: TO PUT THE UNDERSTANDING OF SHORT-TERM DECAY POWER FROM FISSION PRODUCTS AND OTHER PROBLEMS INTO PRACTICE SPECIFICALLY IN A SHORT-TERM REACTION OF REALISTIC OPERATING HISTORY ON A WELL-DOCUMENTED, SCIENTIFIC BASIS.

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

DECAY HEAT • COMPARISONS, THEORY AND EXPERIENCE • FISSION PRODUCT ACTIVITY, GROSS • REACTOR, LUN • SAFETY PROGRAM • NUC-3 • A AND U PROGRAM • JACOBS

119020

ORNL/NUREG-76

119624 - CONTINUED
 LWR SAFETY RESEARCH PROGRAM QUARTERLY REPORT, JAN-MARCH 1/76
 SANDIA LABS., ALBUQUERQUE, NEW MEXICO
 SAND-76-0369 - NUCLEAR POWER - 177 PPS, 22 TABS, 71 FIGS, 57 REFS, DEPT. 1976

THIS WORK WAS PERFORMED BY THE NUCLEAR FUEL CYCLE SAFETY RESEARCH DEPARTMENT AT SANDIA LABORATORIES. THERE ARE 2 MAJOR PHASES OF THE RESEARCH INVOLVING WATER-ROCK-CONCRETE INTERACTIONS AND STEAM EXPLOSION PHENOMENA. THE EXPERIMENTAL ELEMENTS OF THE MULTIPHASE CONCRETE INTERACTION STUDY ARE (1) DEPOSITION OF LONG-TERM WEATHERING CONCRETES, (2) KINETICS AND STOICHIOMETRY OF THE THERMAL DECOMPOSITION OF CONCRETES, (3) MECHANISMS OF CONCRETE THERMAL HEAT FLUXES AT URM SURFACES, AND (4) SIMULATION EXPERIMENTS EXAMINING THE INTERFACE BETWEEN A HOT AND A DECOMPOSING SOLID. IN THE STEAM EXPLOSION PHENOMENA PHASE, THIS QUARTER HAS DEVOTED TO ASSEMBLING, DEBUBBLING, AND SWIRLING DOWN THE VARIOUS COMPONENTS OF THE WATER-ROCK-CONCRETE SYSTEM. THREE MULTINODE STAINLESS STEEL/STEAM INTERACTIONS WERE PERFORMED TO TEST THE SAFETY AND INTEGRITY OF THE SYSTEM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, ILL. 62756
 REACTION, RRM - CONCRETE - EXPLOSION - METAL-WATER REACTION - REACTION, RRM - STEAM - MULTINODE - CONCRETE
 MELTDOWN - REACTION, CON - NUC-3 - R AND D PROGRAM - JACOBI

4. NRC-4 WATER REACTOR SAFETY RESEARCH, ANALYSIS DEVELOPMENT

120300
 RANSOM JR + TRAMP JA
 CHARACTERISTICS, STABILITY, AND SHORT-LENGTH PHENOMENA IN TWO-PHASE EQUATION SYSTEMS
 IDIUM NATIONAL ENGINEERING LAB., IDIUM FALLS
 AEC-1272 +. 22 PPS, 1 FIG, 17 TABS, FEB. 1970

THE OCCURRENCE AND SIGNIFICANCE OF COMPLEX CHARACTERISTICS IN TWO-PHASE FLOW EQUATION SYSTEMS ARE CLARIFIED BY A DETAILED ANALYSIS OF SEPARATED TWO-PHASE FLOW BETWEEN TWO PARALLEL PLATES. THE BASIC SYSTEM OF ONE-DIMENSIONAL, TWO-PHASE FLOW EQUATIONS FOR THIS PROBLEM POSSESSES COMPLEX CHARACTERISTICS, EXHIBITS UNDESIRABLE INSTABILITIES IN THE SHORT-LENGTH LIMIT, AND CONSTITUTES AN IMPROPERLY POSED INITIAL VALUE PROBLEM. THESE DIFFICULTIES HAVE LED SOME USERS TO PROPOSE MAJOR MODIFICATIONS TO THE BASIC EQUATION SYSTEM. THE RELATIVELY MINOR MODIFICATION OF INTRODUCING SURFACE TENSION IS SHOWN TO BE SUFFICIENT TO REMOVE THE CHARACTERISTIC INST. TO STABILIZE SHORT-LENGTH DISTURBANCES, AND TO FORMULATE A PROPERLY POSED PROBLEM. FOR A GIVEN VALUE OF THE SURFACE TENSION, THE BASIC EQUATION SYSTEM THIS MODIFIED IS SHOWN TO PREDICT CORRECTLY THE EVOLUTION OF SMALL-AMPLITUDE DISTURBANCES HAVING WAVELENGTHS LONG COMPARED TO THE PLATE SPACING.

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

A AND B PROGRAM + SAFETY PROGRAM + REACTOR, LWR + FLOW, TWO PHASE + THERMAL ANALYSIS + ANALYTICAL MODEL + FLOW STABILITY + NRC-4

110000
 RELAP/MODS - A COMPUTER PROGRAM FOR TRANSIENT THERMAL-HYDRAULIC ANALYSIS OF NUCLEAR REACTORS AND RELATED SYSTEMS USER'S MANUAL. VOLUME I - RELAP/MODS DESCRIPTION
 AERJET NUCLEAR COMPANY, IDIUM FALLS
 AEC-1272-1334 (VOL. 1) +. 330 PPS, 17 TABS, 30 FIGS, SEPT. 1970

RELAP MODELS SYSTEM FLOW CONDITIONS INCLUDING FLOW, PRESSURE, MASS INVENTORIES, FLUID QUALITY, AND HEAT TRANSFER. A SUBROUTINE PROVIDES BATH PROPERTY TABLES. COMPONENT THERMAL CONDITIONS AND ENERGY TRANSFERS ARE MODELED. THE REACTOR SYSTEM IS SUBDIVIDED INTO DISCRETE VOLUMES WHICH WITH INTERCONNECTING JUNCTIONS (FLOW PATHS), ARE TREATED AS ONE DIMENSIONAL HOMOGENEOUS ELEMENTS. RELAP SOLVES AN INTEGRAL FORM OF FLUID CONSERVATION AND STATE EQUATIONS FOR EACH USER DEFINED VOLUME AND GENERATES A TIME HISTORY OF SYSTEM CONDITIONS. DATA ARE RECORDED FOR VOLUME FLUID, COMPONENT HEAT, AND JUNCTION FLOW CHARACTERISTICS. THE OUTPUT IS IN THE FORM OF PRINTED TABLES DIGITAL DATA.

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

REACTOR, BWR + COMPUTER PROGRAM + HEAT TRANSFER + REACTOR, PWR + REACTOR KINETICS + REACTOR, LWR + FLOW + TRANSIENT + NRC-4 + THERMAL HYDRAULIC ANALYSIS

110009
 RELAP/MODS - A COMPUTER PROGRAM FOR TRANSIENT THERMAL-HYDRAULIC ANALYSIS OF NUCLEAR REACTORS AND RELATED SYSTEMS USER'S MANUAL. VOLUME II - PROGRAM IMPLEMENTATION
 AERJET NUCLEAR COMPANY, IDIUM FALLS
 AEC-1272-1334 (VOL. 2) +. 470 PPS, 20 TABS, 20 FIGS, SEPT. 1970

THIS PORTION OF THE RELAP/MODS USER'S MANUAL PRESENTS THE DETAILS OF SETTING UP AND ENTERING THE REACTOR MODEL TO BE EVALUATED. THE INPUT CARD FORMAT AND ARRANGEMENT IS PRESENTED IN DETAIL, INCLUDING NOT ONLY CARDS FOR DATA BUT ALSO THOSE FOR EDITING AND RESTARTING. PROGRAM INITIALIZATION INCLUDING PRESSURE DISTRIBUTION AND ENERGY BALANCE IS DISCUSSED. A SECTION ENTITLED 'USER GUIDELINES' IS INCLUDED TO PROVIDE RUNNING RECOMMENDATIONS, ANALYSIS AND VERIFICATION TECHNIQUES, AND COMPUTATIONAL DIFFICULTY RESOLUTION. THE SECTION IS CONCLUDED WITH A DISCUSSION OF THE COMPUTER OUTPUT FORM AND FORMAT.

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

COMPUTER PROGRAM + HEAT TRANSFER + REACTOR KINETICS + REACTOR, LWR + FLOW + PRESSURE TRANSIENT + DISTRIBUTION + TRANSIENT + NRC-4 + THERMAL HYDRAULIC ANALYSIS

110090
 RELAP/MODS - A COMPUTER PROGRAM FOR TRANSIENT THERMAL-HYDRAULIC ANALYSIS OF NUCLEAR REACTORS AND RELATED SYSTEMS USER'S MANUAL. VOLUME III - CHECKOUT APPLICATIONS
 AERJET NUCLEAR COMPANY, IDIUM FALLS
 AEC-1272-1334 (VOL. 3) +. 265 PPS, 34 TABS, 127 FIGS, SEPT. 1970

PROVIDES A DESCRIPTION OF THE CHECKOUT PROCESS AND THE SPECIFIC PROBLEMS MODELED TO DEMONSTRATE THE UTILITY OF THE CODE. IT SHOULD BE NOTED THAT THE CHECKOUT PROBLEMS REPORTED IN THIS PUBLICATION WERE RUN USING UPDATE 1 OF RELAP/MODS. SIX MONTHS OF THE USE OF UPDATE 1 LED TO THE IDENTIFICATION OF A NUMBER OF MINOR ERRORS IN CODING AS WELL AS SOME LOGIC CHANGES. ADDITIONAL EVALUATION MODEL CHANGES AND CODE MODEL REVISIONS. THESE CHANGES ELIMINATED IN THE DEVELOPMENT OF AN IMPROVED CODE IDENTIFIED AS RELAP/MODS, UPDATE 2. THE DIFFERENCES BETWEEN UPDATES 1 AND 2 ARE OUTLINED IN APPENDIX 1. THE CHECKOUT OF UPDATE 2 IS ALSO INCLUDED IN APPENDIX 2.

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

COMPUTER PROGRAM + HEAT TRANSFER + REACTOR KINETICS + REACTOR, LWR + FLOW + TRANSIENT + NRC-4 + THERMAL HYDRAULIC ANALYSIS

120062
 NORDEN RE • NORTONSEN GA • NORDEN L • NORDEN JM
 NUMERICAL METHODS FOR SOLVING THE COUPLING CONDITIONS FOR A SEPARATED CONTINUUM
 IOWA NATIONAL ENGINEERING LAB., IOWA FALLS
 ANND-7006-1300 • 37 PPS, 11 FIGS, 11 REFS, SEPT. 1976

A DESIRE TO MORE ACCURATELY PREDICT THE BEHAVIOR OF TRANSIENT TWO-PHASE FLOWS HAS RESULTED IN AN INVESTIGATION OF THE FEASIBILITY OF COMPUTING UNIFORM PHASE VELOCITIES AND UNIFORM PHASE TEMPERATURES. THE FINAL DIFFERENCE FORMS OF A SET OF EQUATIONS GOVERNING A SEPARATED CONTINUUM ARE PRESENTED ALONG WITH THE METHODS DEVELOPED FOR SOLVING THE RESULTING SYSTEMS OF SIMULTANEOUS NONLINEAR EQUATIONS. RESULTS FROM A ONE-DIMENSIONAL COMPUTER CODE ARE PRESENTED TO ILLUSTRATE THE CAPABILITIES OF ONE OF THE SOLUTION METHODS.

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

ACCIDENT, LOSS OF COOLANT • FLOW, TWO PHASE • NUMERICAL METHOD • THERMAL HYDRAULIC ANALYSIS • NUC-4

120061
 DIETZ RA
 QUARTERLY TECHNICAL PROGRESS REPORT ON WATER REACTOR SAFETY PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH APRIL-JUNE 1976
 IOWA NATIONAL ENGINEERING LAB., IOWA FALLS
 TREC-7006-1000 • 117 PPS, 75 FIGS, 57 REFS, OCT. 1976

RESULTS OF A REFLUX HEAT TRANSFER TEST SERIES WERE ANALYZED TO DETERMINE THE REFLUX HEAT TRANSFER CHARACTERISTICS OF THE SEMISCALE NED-1 ELECTRICALLY HEATED CORE. A SPECIAL TEST TO INVESTIGATE THE SENSITIVITY OF DEPARTURE FROM NUCLEATE BOILING TO A CHANGE IN CORE GEOMETRY WAS CONDUCTED WITH THE SEMISCALE NED-1 FACILITY. TWO LOSS-OF-COOLANT EXPERIMENTS, INCLUDING AN INVESTIGATION OF HOT-SPOT DELAY EFFECTS, WERE CONDUCTED IN LEFT. FUEL BEHAVIOR VERIFICATION EFFORTS CONCENTRATED ON A WGR AND MORE-DETAILED SEALING MODEL, IMPLEMENTED IN THE EXPRESSIONS FOR CLADDING EXPANSION AND SPECIFIC HEAT. MODIFICATION OF THE ZIRCALOY PLASTIC DEFORMATION MODEL, AND ANALYSIS OF RESULTS FROM STEADY STATE AXIAL GAS FLOW EXPERIMENTS UTILIZING POR RINGS WITH ARROW AND MELTIN CASES.

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

REACTOR, LWR • ACCIDENT, LOSS OF COOLANT • CORE REFLUXING • FUEL BEHAVIOR • HEAT EXCH • ENGINEERING SAFETY FEATURE • COMPUTER PROGRAM • BWR • B AND D PROGRAM • NUC-2 • NUC-3 • NUC-4

114002
 LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT OCTOBER-DECEMBER 1975
 ARGONNE NATIONAL LABORATORY, ILLINOIS
 ANL-76-15 • 76 PPS, 7 TABS, 70 FIGS, 12 REFS, 1976

THIS PROGRESS REPORT SUMMARIZES THE ARGONNE NATIONAL LABORATORY WORK PERFORMED DURING OCTOBER, NOVEMBER, AND DECEMBER 1975 ON WATER-REACTOR-REACTOR SAFETY HEAT-TRANSFER AND FLOW PROBLEMS. THE FOLLOWING RESEARCH AND DEVELOPMENT AREAS ARE COVERED: (1) LOSS-OF-COOLANT ACCIDENT RESEARCH; HEAT TRANSFER AND FLOW DYNAMICS; (2) TRANSIENT FUEL BEHAVIOR AND FISSION-PRODUCT RELEASE PROGRAMS; AND (3) MECHANICAL PROPERTIES OF ZIRCALOY CONTAINING OXYGEN.

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

REACTOR, BWR • FISSION PRODUCT RELEASE • HEAT TRANSFER • ACCIDENT, LOSS OF COOLANT • BATHEN • REACTOR, PHN • ANALYTICAL MODEL • DYNAMICS, NONLINEAR • ZIRCALOY • BLOOMER • JACOBS

110917
 LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT JAN-MAR, 1976
 ARGONNE NATIONAL LABORATORY, ILLINOIS
 ANL-76-09 • 71 PPS, 12 TABS, 40 FIGS, MARCH 1976

THIS PROGRESS REPORT SUMMARIZES THE ARGONNE NATIONAL LABORATORY WORK PERFORMED DURING THE PERIOD JANUARY-MARCH 1976 ON WATER REACTOR SAFETY AND HEAT TRANSFER PROBLEMS. THE WORK IN LOSS OF COOLANT ACCIDENT RESEARCH INCLUDES SHIFT-FLUX MODELING, NITROGEN-OXYGEN BLOOMER TESTS AND TRANSIENT CRITICAL HEAT FLUX TESTS. WORK IS ALSO REPORTED ON MODELING OF FUEL-FISSION PRODUCT BEHAVIOR, MECHANICAL PROPERTIES OF ZIRCALOY CONTAINING OXYGEN, AND STEAM EXPLOSION STUDIES.

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

ACCIDENT, LOSS OF COOLANT • ACCIDENT, LOSS OF PRESSURE • REACTOR, SAFETY RESEARCH • OUT OF PILE EXPERIMENT • REACTOR, LWR • ZIRCALOY • NUC-3 • BLOOMER • HEAT FLUX, CRITICAL • NUC-4 • THERMAL HYDRAULIC ANALYSIS • JACOBS

120201
 LIGHT WATER REACTOR SAFETY RESEARCH PROGRAM QUARTERLY PROGRESS REPORT APRIL-JUNE 1976
 ARGONNE NATIONAL LABORATORY, ILLINOIS
 ANL-76-07 • 60 PPS, 2 TABS, 50 FIGS, 1976

SUMMARIZES THE ARGONNE NATIONAL LABORATORY WORK PERFORMED DURING APRIL, MAY, AND JUNE 1976 ON WATER-REACTOR-REACTOR SAFETY HEAT-TRANSFER AND FLOW PROBLEMS. THE FOLLOWING RESEARCH AND DEVELOPMENT

120001 CONTINUED

AREAS ARE COVERED: (1) LOSS-OF-COOLANT ACCIDENT RESEARCH: HEAT TRANSFER AND FLUID DYNAMICS; (2) TRANSIENT FUEL RESPONSE AND FISSION-PRODUCT RELEASE PHENOMENA; (3) MECHANICAL PROPERTIES OF ZIRCALOY CONTAINING UTHER, AND (4) STEAM-EXPLOSION STUDIES.

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

REACTOR, LOU + ACCIDENT, LOSS OF COOLANT + ZIRCALOY + UNIT OF FUEL ELEMENTS + COMPUTER PROGRAM + HEAT FLUX + CRITICAL + HEAT TRANSFER ANALYSIS + DISCUSSION + FISSION PRODUCT RELEASE + FLEXIBLE PROPERTY + NUC-3 + NUC-4

117511

PROJECT DESCRIPTION FOR FISSION PRODUCT TRANSPORT ANALYSIS

WATKINS, COLUMBIA LAB., UNIV

22 PPS, 2 FIGS, REC. NO. 70-405, JUNE 4, 1976

STUDIES ARE CURRENTLY BEING INITIATED TO PROVIDE DATA WHICH SHOULD SIGNIFICANTLY IMPROVE THE DESCRIPTION OF FISSION PRODUCT RELEASE FROM FUEL RODS FOR THE TERMINATED LOCA. IN ADDITION, EXTENSIVE THERMAL-HYDRAULIC ANALYSES OF EMERGENCY CORE COOLING SYSTEM (ECCS) OPERATION HAVE PROVIDED CONSIDERABLE DATA ON FLUID FLOW AND OTHER CONDITIONS THROUGHOUT THE PRIMARY SYSTEM LOOPS FOR THE TERMINATED LOCA. THEREFORE, IT IS APPROPRIATE AT THIS TIME TO DEVELOP A PRIMARY SYSTEM FISSION PRODUCT TRANSPORT MODEL WHICH IS CONSISTENT WITH THE CURRENTLY AVAILABLE DATA AND THAT WOULD REALISTICALLY CALCULATE THE RATE AND MAGNITUDE OF FISSION PRODUCT RELEASE TO THE CONTAINMENT CAN BE PERFORMED.

AVAILABILITY - NAC PUBLIC LIBRARY ROOM, 1717 M STREET, WASHINGTON, D. C. 20005 100 CENTS/PAGE -- MINIMUM CHARGE \$2.00

REACTOR, LOU + CHEMICAL REACTION + DEPOSITION + FUEL MELTOWN + ACCIDENT, LOSS OF COOLANT + REACTOR, PWR + SHIPPING CONTAINER + FISSION PRODUCT TRANSPORT + NUC-2 + THERMAL HYDRAULIC ANALYSIS

110000

GEORGE JA + BARTON P + LERCHER AS + JORDAN H + WATSON JR
FISSION PRODUCT TRANSPORT ANALYSIS

WATKINS, COLUMBIA LAB., UNIV

WIL-MUREC-1900 + 37 PPS, 4 TABS, 4 FIGS, 10 REFS, SEPT. 7, 1976

ANALYSIS METHODS AND COMPUTER MODELS FOR PREDICTING FISSION PRODUCT TRANSPORT AND DEPOSITION UNDER POSTULATED LOSS OF COOLANT ACCIDENT CONDITIONS WITHIN WATER-LOADED REACTOR PRIMARY SYSTEMS AND SHIPPING CASKS ARE BEING DEVELOPED. EXISTING THERMAL-HYDRAULIC ANALYSES (RELAP-5) FROM THE BASIS FOR DEFINING THE CONDITIONS WITHIN THE PRIMARY SYSTEMS AND THE TRANSPORT FLOW FROM WHICH FISSION PRODUCTS DEPOSIT. THE ANALYSES TO BE DEVELOPED IN THIS STUDY WILL PROVIDE METHODS FOR MORE REALISTICALLY CALCULATING THE RATE AND MAGNITUDE OF FISSION PRODUCT RELEASE TO THE CONTAINMENT.

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

REACTOR, LOU + DEPOSITION + ACCIDENT, LOSS OF COOLANT + REACTOR, PWR + ANALYTICAL MODEL + FISSION PRODUCT TRANSPORT + NUC-3 + NUC-4 + THERMAL HYDRAULIC ANALYSIS + JALOUS

110304

WELF D + GROSS A + MENZIES JR + WARDEN B

DEVELOPMENT OF A COMPUTER CODE FOR THERMAL HYDRAULICS OF REACTORS (THUR) FIFTH QUARTERLY PROGRESS REPORT
OCTOBER-DECEMBER 1975

ORGANIZATION NATIONAL LAB., WATKINS, APT.

WIL-MUREC-50530 + 34 PPS, 1 FIG, 10 REFS, JULY 1976

THIS IS THE FIFTH REPORT OF A SERIES. CRITICAL TWO-PHASE FLOW OF A TWO-EQUIVALENT, IMMISCIBLE MIXTURE OF LIQUID WATER AND VAPOR WAS COMPUTED BY THE MODIFIED TUNNEY AND IMPLICIT EULER METHODS WHICH ARE INTENDED FOR USE IN THE SYSTEMS CODE. ASSUMPTIONS FOR 2D THERMAL-HYDRAULIC AND TRANSPORT PROPERTIES OF WATER, FUEL AND CLAD MATERIALS WERE COMPLETED AND VERIFIED. A MODEL FOR THE CENTRIFUGAL PUMP WAS USED AND A SCHEME FOR IMPLICIT COMPUTING OF COMPONENTS IN A GENERAL HYDRAULIC SYSTEM WAS DEVELOPED.

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

COMPUTER PROGRAM + REACTOR, LOU + FLOW, TWO PHASE + TRANSPORT PROPERTY + W AND V PROGRAM + TRANSPORT + NUC-4 + THERMAL HYDRAULIC ANALYSIS

110001

LATHROP AC

REACTOR SAFETY AND TECHNOLOGY QUARTERLY PROGRESS REPORT, APRIL 1-JUNE 30, 1976

LOS ALAMOS SCIENTIFIC LABORATORY, NEW MEXICO

LA-MUREC-6007-PH + 70 PPS, 17 TABS, 55 FIGS, 137 REFS, AUG. 1976

THE NEUTRONIC MODEL DESIGN OF EQUIPMENT FOR FUEL HEAT-UP EXPERIMENTS WAS COMPLETED. A TWO-DIMENSIONAL CORE TEMPERATURE MODEL WAS DEVELOPED FOR THE LANC CODE, AND A ONE-DIMENSIONAL ANALYTICAL DIFFUSION MODEL WAS FORMULATED. A MATH MODELING WAS HEAVILY IN THE LOU SAFETY RESEARCH PROGRAM WITH THE CREATION OF THE FIRST VERSION OF THE TRAC CODE. THE VERSION CONSISTS OF ONE-DIMENSIONAL COMPONENT MODELS, SERVICE MANUALS, AND CONTROL MODELS AND IS CAPABLE OF

119001 (CONTINUED)

RUNNING VERY GENERAL LOOP CONFIGURATIONS. IN LUPON SAFETY RESEARCH, A NEW PHASE-TRANSITION MODEL WAS ADDED TO SIMMEN-1. THE NEW CONDUCTION-LIMITED MODEL USES A MORE IMPLICIT MATHEMATICAL SOLUTION AND PROVIDES WITH A MORE PHYSICAL AND A MORE STABLE RESULT THAN THE PREVIOUS EXPLICIT SOLUTION USING THE NEGATIVE MODEL.

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

ACCIDENT ANALYSIS • COMPUTER PROGRAM • CONTAINMENT ANALYSIS • FISSION PRODUCT RELEASE • EMISSION • ANALYTICAL MODEL • COOLANT CHEMISTRY • HELIUM • REACTION, LUP • SAFETY PROGRAM • ALUMINUM • H AND O PROGRAM • REACTION, HYDRO • REACTION, LUPON • SEISMIC DESIGN • SAFEGUARDS, NUCLEAR MATERIAL • RUC-6 • RUC-7 • RUC-8 • THERMAL HYDRAULIC ANALYSIS • JACOBS • ACCIDENT, CORE DISRUPTIVE

119004

GENO RG • GAMES CI • LANTEN RL

COMPARE: A COMPUTER PROGRAM FOR THE TRANSIENT CALCULATION OF A SYSTEM OF VOLUMES CONNECTED BY PLUMBING VENT. LOS ALAMOS SCIENTIFIC LABORATORY, NEW MEXICO
LA-MUREG-0400-NS • 72 PPS, 10 TABS, 20 FIGS, 10 REFS, SEPT. 1970

THIS PROGRAM PERFORMS TRANSIENT SUBCOMPARTMENT PRESSURE RESPONSE ANALYSIS OF NUCLEAR POWER PLANTS. THE VOLUME THERMODYNAMICS AND FLOW EQUATIONS ARE FOR A HOMOGENEOUS MEDIUM ASSUMED TO BE IN THERMODYNAMIC EQUILIBRIUM, CONSISTING OF ANY ONE OR ANY COMBINATION OF THE FOLLOWING: (1) STEAM, (2) TWO-PHASE WATER AND (3) ANY THREE PERFECT GASES SUCH AS AIR, HELIUM, ETC.

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

COMPUTER PROGRAM • CONTAINMENT • STEAM • THERMODYNAMICS • FLOW, TWO PHASE • WATER VAPOR • PRESSURE TRANSIENT • COMPARTMENT • RUC-6

120039

REAY RG • CONOVER RJ • WHITERAN DE

REPORT ON THE APPLICATION OF STATISTICAL TECHNIQUES TO THE ANALYSIS OF COMPUTER LUP
LOS ALAMOS SCIENTIFIC LABORATORY, NEW MEXICO
LA-MUREG-0520-NS • 15 PPS, 14 FIGS, 5 REFS, OCT. 1970

TWO TYPES OF SAMPLING PLANS ARE PRESENTED AS ALTERNATIVES TO SIMPLE RANDOM SAMPLING IN MONTE CARLO STUDIES. THESE PLANS ARE SHOWN TO BE IMPROVEMENTS OVER SIMPLE RANDOM SAMPLING WITH RESPECT TO THE VARIANCE OF A CLASS OF ESTIMATORS WHICH INCLUDES THE SAMPLE MEAN AND THE SAMPLE CUMULATIVE DISTRIBUTION FUNCTION. THE PARTIAL CORRELATION COEFFICIENT IS PRESENTED AS A MEASURE OF SENSITIVITY. EXAMPLES AND TEST CASES ARE INCLUDED.

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

STATISTICAL ANALYSIS • SAMPLING • MONTE CARLO • COMPUTER PROGRAM • RUC-6 • SENSITIVITY ANALYSIS

127000

LATHROP RG

REACTOR SAFETY AND TECHNOLOGY QUARTERLY PROGRESS REPORT, JULY 1-SEPTEMBER 30, 1970

LOS ALAMOS SCIENTIFIC LAB., NEW MEXICO

LA-MUREG-0570-PR • 90 PPS, 17 TABS, 80 FIGS, 85 REFS, NOV. 1970

IN THE FISSION PRODUCT TRANSPORT TASK OF THE PWR SAFETY PROGRAM, THE LANC-1 CODE HAS DEVELOPED. EQUATIONS WERE FORMULATED FOR THE MORE POWERFUL CODES LANC-2 AND LANC-3, AND THE LANC-1 CODE WAS WRITTEN TO PREVENT FISSION PRODUCT LEAKAGE FROM THE CONTAINMENT BUILDING. SEVERAL IMPORTANT ADVANCES WERE ACHIEVED IN THE LUP SAFETY PROGRAM. AN ANALYSIS OF NON TEST PROBLEM NO. 2 WAS PERFORMED USING THE THAC CODE, AND CALCULATED RESULTS WERE IN EXCELLENT AGREEMENT WITH EXPERIMENTAL VALUES. IN LUPON SAFETY RESEARCH, TRANSITION PHASE MODEL-CUMS AND MODEL-SWASSERMAN STUDIES WERE PERFORMED WITH THE SIMMEN-1 CODE.

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

ACCIDENT ANALYSIS • COMPUTER PROGRAM • CONTAINMENT ANALYSIS • FISSION PRODUCT RELEASE • EMISSION • ANALYTICAL MODEL • COOLANT CHEMISTRY • HELIUM • REACTION, LUP • SAFETY PROGRAM • ALUMINUM • H AND O PROGRAM • REACTION, HYDRO • REACTION, LUPON • SEISMIC DESIGN • SAFEGUARDS, NUCLEAR MATERIAL • RUC-6 • RUC-7 • RUC-8 • THERMAL HYDRAULIC ANALYSIS • JACOBS • ACCIDENT, CORE DISRUPTIVE

117002

SMITH TA

HEAT TRANSFER AND CARRYOVER OF LUP PRESSURE WATER IN A HEATED VERTICAL TUBE

MASSACHUSETTS INSTITUTE OF TECHNOLOGY, CAMBRIDGE

MUREG-0105 • 110 PPS, 2 TABS, 29 FIGS, AUG. 1970

LOCAL HEAT TRANSFER COEFFICIENTS IN THE STABLE FILM BOILING AND DISPERSED FLOW REGIMES WERE STUDIED FOR THE UPWARD FLOW OF LUP PRESSURE WATER IN A HEATED VERTICAL TUBE. WALL TEMPERATURES WERE MAINTAINED CONSTANT WITH TIME AND ALONG THE TUBE SO THAT BOTH AXIAL AND TIME TEMPERATURE GRADIENTS APPROXIMATED ZERO. SEVERAL THEORIES FOR THE DIFFERENT POSSIBLE TYPES OF FLUX FLAMING OR TURBULENT, TUBE WALL FILM WERE COMPARED WITH THE EXPERIMENTAL DATA. THE CARRYOVER POINT FOR LUP FLOODING RATES (1 INCH/SEC OR LESS) WAS INFERRED FROM THESE COMPARISONS AND WERE WELL AGREEMENT WITH THE PLUMMER CRITICAL MASS CRITERION FOR LIQUID CARRY OVER.

117002 CONTINUOUS
 AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

HEAT TRANSFER • ACCIDENT • LOSS OF COOLANT • NUC-2 • STEEL • STAINLESS • REACTOR, LWR • OVERHEATING • SAFETY
 PROGRAM • S AND S PROGRAM • FILM BOILING • TEMPERATURE • NUC-3 • JACOBI

117070
 REPLENISHING
 REPLENISHING HEAT TRANSFER IN A LIGHT WATER REACTOR
 MASSACHUSETTS INSTITUTE OF TECHNOLOGY, CAMBRIDGE
 THREE-DIMENSIONAL, 1 L III •, 200 PPS, 50 PICS, 70 LOTS, AUG. 1970

THE HEAT TRANSFER PHENOMENA OCCURRING DURING THE REPLENISHING PHASE OF A HYPOTHETICAL LOSS OF COOLANT
 ACCIDENT ARE INVESTIGATED. IT WAS FOUND THAT PEAK CLADDING TEMPERATURE EXCURSIONS ARE TERMINATED
 BY THE PRESENCE OF ENTRAINED LIQUID DROPS IN THE VERTICAL PLUING VAPOR STREAM. THIS DISPERSED
 FILM FILM BOILING REGIME IS CREATED DUE TO A COMPLEX SET OF UNSTEADY THERMAL-HYDRAULIC
 CONDITIONS. NEUR PRESENTATION OF THE REPLENISHING OF A TWO PHASE MIXTURE PROVIDES A CALCULATIONAL METHOD
 WHICH DESCRIBES THE MANY PROCESSES OCCURRING IN THE JOURNAL DURING THE TRANSITION. A PHYSICALLY
 BASED CALCULATIONAL METHOD HAS BEEN DEVELOPED FOR PREDICTION OF REPLENISHING HEAT TRANSFER.

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

COMPUTER PROGRAM • CLADDING • HEAT TRANSFER • ACCIDENT • LOSS OF COOLANT • ANALYTICAL MODEL • NUC-2 •
 REACTOR, LWR • OVERHEATING • SAFETY PROGRAM • WATER VAPOR • S AND S PROGRAM • TEMPERATURE • NUC-3 • FILM
 REPLENISHING • JACOBI

5. NRC-S WATER REACTOR SAFETY RESEARCH, RETAILERED THE MATERIALS

110020
 RODDAMEN EC • MOORE SE
 PLASME JOINTS WITH CONTACT INSIDE THE BOLT CIRCLE - ASME PART 8 DESIGN RULES
 BATTLE COLUMBUS LABS., OHIO • CAN RIDGE NATIONAL LABORATORY, TENN.
 ORNL/SUB-2013/REPORT NO. 11 • 20 PPS, 12 TABS, 9 FIGS, MAY 1970

THIS REPORT (A) SUMMARIZES THE THEORY FOR PART 8 PLASME JOINTS, (B) PRESENTS EXAMPLES WHICH SHOW THE SIGNIFICANT DIFFERENCES BETWEEN PART 8 AND PART 8 PLASME JOINTS, (C) PRESENTS THE AVAILABLE TEST DATA RELEVANT TO THE CHARACTERISTICS OF PART 8 PLASME JOINTS, (D) LISTS LISTINGS OF TWO COMPUTER PROGRAMS WHICH CAN BE USED TO EVALUATE THE CHARACTERISTICS OF PART 8 PLASME JOINTS, AND (E) GIVES RECOMMENDATIONS FOR CODE REVISIONS AND OTHER ASPECTS OF PART 8 PLASME-JOINT DESIGN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 CODES AND STANDARDS • COMPUTER PROGRAM • DESIGN CRITERIA • NRC-S • PLASME • PIPES AND PIPE FITTINGS • JACOBS

117013
 RODDAMEN EC
 APPROPRIATE MINERAL STRESSES FOR USE WITH ASME CODE PRESSURE-LOADING STRESS INDICES FOR NUCLEAR
 BATTLE COLUMBUS LABS., OHIO
 ORNL/SUB-2013-2 • 30 PPS, 8 TABS, 8 FIGS, 13 REFS, JUNE 1970

THE STUDY OF MINERAL STRESSES AND STRESS INDICES DESCRIBED IN THIS REPORT IS PART OF A CONTINUING STUDY OF DESIGN RULES FOR NUCLEI IN PRESSURE VESSELS AS A COOPERATIVE EFFORT WITH INDUSTRY TO DEVELOP AND VERIFY ANALYTICAL METHODS FOR ASSESSING THE SAFETY OF NUCLEAR PRESSURE-VESSSEL AND PIPING-SYSTEM DESIGN. IN THE PRESENT REPORT, THE BACKGROUND OF THE STRESS INDEX METHOD IS BRIEFLY DISCUSSED. THE REINFORCEMENT REQUIREMENTS ARE CRUCIAL TO AN UNDERSTANDING OF THE APPLICABILITY OF THE STRESS INDICES AND ARE BRIEFLY REVIEWED. THE REPORT THEN PRESENTS DATA RELEVANT TO THE MAXIMUM STRESS INDEX AND A DISCUSSION OF THAT DATA. THE REPORT DEALS PRINCIPALLY WITH ONE ASPECT OF THE OVERALL PROBLEM: THE MAXIMUM STRESS FOR NUCLEI IN OPENING WITH NO LOCAL REINFORCEMENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 CODES AND STANDARDS • STRESS ANALYSIS • NRC-S • NUCLEI • PRESSURE VESSELS • PIPES AND PIPE FITTINGS • JACOBS

117307
 HORN GT • GEMLEN PC • HANGLAND RL • KENNEDY HP • POPELAN C • ROSENFIELD AN
 CRITICAL EXPERIMENTS, MEASUREMENTS AND ANALYSES TO ESTABLISH A CRACK ARREST METHODOLOGY FOR NUCLEAR PRESSURE
 VESSEL STEELS. TASK 02, SIXTH QUARTERLY PROGRESS REPORT JANUARY THROUGH MARCH 1970
 BATTLE COLUMBUS LABS., OHIO
 ORNL-NUREG-1951 • 62 PPS, 22 FIGS, JULY 1970

THE GOVERNING EQUATIONS FOR A COMPLETE TWO-DIMENSIONAL ANALYSIS OF CRACK FRACTURE AND CRACK ARREST - INCLUDING INERTIA AND THERMAL GRADIENTS ARE DERIVED. STRAIN ENERGY AND COMPLIANCE VALUES CALCULATED IN THIS WAY FOR AN OPEN SPECIMEN ARE IN EXCELLENT AGREEMENT WITH EXPERIMENTAL OBSERVATION. COMPLIANCE MEASUREMENTS HAVE BEEN CARRIED OUT ON THE RECTANGULAR OPEN SPECIMEN TO TEST THE ACCURACY OF THE COMPLIANCE VALUES CALCULATED WITH THE ONE-DIMENSIONAL BEAM-TURN-UP ASTIC-FOUNDATION MODEL. RESULTS SHOW THAT THE COMPLIANCE OF THE RECTANGULAR OPEN SPECIMEN IS NOT ACCURATELY REPRESENTED BY THE MODEL. CORRECTIONS TO BE APPLIED TO STRESS INTENSITY VALUES ARE IDENTIFIED. FINALLY, A NEW PROCEDURE FOR MEASURING K_{IC} AND K_{IS} VALUES IS DESCRIBED. THE PROCEDURE IS BASED ON A DYNAMIC ANALYSIS OF THE CRACK LENGTH AT ARREST AND DYNAMICALLY STEPPED BEAM LOADING AND ELIMINATES THE NEED FOR THE MORE DIFFICULT CRACK VELOCITY MEASUREMENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 STEEL • TEMPERATURE GRADIENT • NRC-S • MATHEMATICAL TREATMENT • CRACK • PRESSURE VESSELS • JACOBS

119215
 RODDAMEN EC • MOORE SE
 EVALUATION OF THE BOLTING AND FLANGES OF ANSI 800-S PLASME JOINTS - ASME PART 8 DESIGN RULES
 BATTLE COLUMBUS LABS., OHIO (FOR ORNL)
 ORNL/SUB-2013-3 • 170 PPS, 29 TABS, 20 FIGS, 20 REFS, SEPT. 30, 1970

A NUMBER OF AMBIGUITIES AND UNNECESSARY RESTRICTIONS ON THE USE OF 800-S PLASME JOINTS IN THE CODE RULES STILL REMAIN. THIS REPORT WAS PREPARED TO INVESTIGATE THE USE OF 800-S PLASME JOINTS IN CODE APPLICATIONS AND TO MAKE RECOMMENDATIONS FOR APPROPRIATE MODIFICATIONS OF THE CODE RULES. CALCULATIONS WERE MADE USING THE CODE PROCEDURES AND THE COMPUTER PROGRAM FLANGE. AVAILABLE TEST DATA ON PLASME JOINTS ARE PRESENTED AND COMPARED WITH RESULTS FROM FLANGE CALCULATIONS IN APPENDIX A. THESE COMPARISONS INDICATE THAT THE CALCULATION METHODS USED IN THE TEXT OF THE REPORT ARE CONSERVATIVE, BUT NOT NECESSARILY ACCURATE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 CODES AND STANDARDS • COMPUTER PROGRAM • TESTING • STRESS ANALYSIS • DESIGN STUDY • NRC-S • PLASME • COMPARISON • PIPES AND PIPE FITTINGS • JACOBS

120151

120151 CONTINUED
 MORRISON EC • CHARTREY MC
 ELASTIC STRESSES AT REINFORCED HOLES IN SPHERICAL SHELLS WITH PRESSURE AND MOMENT LOADING
 DAYVILLE COLUMBUS LABS., OHIO
 OMI-3006 • 4 PPS. 7 TABS. 11 FIGS. 9 REFS. SEPT. 20, 1970

CALCULATED ELASTIC STRESSES AT REINFORCED HOLES IN SPHERICAL SHELLS WITH PRESSURE AND MOMENT LOADING ARE PRESENTED. THE MODELS USED IN THE CALCULATIONS REPRESENT A WIDE VARIETY OF REINFORCED SHAPES; ALL HAVING CORE CONCENTRATIONS. THE RESULTS SHOW LOW STRESS INDICES FOR PRESSURE LOADING FOR HOLES WITH LOCAL REINFORCEMENT ARE ACCEPTABLE WITH SOME MODIFICATION IN COVERAGE. SIMPLE EQUATIONS FOR STRESS INDICES FOR MOMENT LOADING ARE DEVELOPED. POTENTIAL APPLICATION OF THE MOMENT-LOADING STRESS INDICES IS DISCUSSED. SEVERAL RECOMMENDATIONS FOR LOW STRESS INDICES ARE INCLUDED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 SPHERE • NOZZLE • STRESS ANALYSIS • CORES AND STANDARDS • MATHEMATICAL TREATMENT • HOLES • PRESSURE VESSELS

120097
 HYUNDAI OF • SCHWENKER UD • GROUP JJ
 RESIDUAL STRESSES AT WELDMENTS IN PIPES AND PRESSURE VESSELS. QUARTERLY PROGRESS REPORT, JULY 15, 1970 - OCTOBER 15, 1970
 DAYVILLE COLUMBUS LABS., OHIO
 OMI-3006-1950 • 4 PPS. 4 TABS. 13 FIGS. 10 REFS. OCT. 1970

PRESENTS A PROGRESS SUMMARY FOR A COMBINED EXPERIMENTAL AND ANALYTICAL STUDY TO DEVELOP MATHEMATICAL MODELS FOR PREDICTING RESIDUAL STRESSES IN WELD-JOINT DETAILS. THE PROGRAM IS DIVIDED INTO THREE TASKS: A CRITICAL LITERATURE REVIEW, LABORATORY TESTS, AND DEVELOPMENT OF ANALYTICAL MODELS. THE LITERATURE REVIEW HAS BEEN COMPLETED. SPECIFIC ANALYTICAL MODELS FOR PREDICTING TEMPERATURE DISTRIBUTIONS AND RESIDUAL STRESSES HAVE BEEN IDENTIFIED. PRELIMINARY CALCULATIONS WITH THE TEMPERATURE MODEL SHOWED GOOD CORRELATION WITH EXPERIMENTAL DATA. THE MATHEMATICAL REPRESENTATION FOR PREDICTING STRESS DISTRIBUTIONS IN WELDED JOINTS IS IN EARLY STAGES OF DEVELOPMENT. PRELIMINARY CHECKS ON THE STRESS ANALYSIS MODEL SUCCESSFULLY DEMONSTRATED ITS CAPABILITY TO HANDLE UNLOADING. THE FIRST BELTING EXPERIMENT HAS BEEN COMPLETED. DATA ARE STILL BEING OBTAINED FROM THE SPECIMENS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 WELDING • PIPES AND PIPE FITTINGS • PRESSURE VESSELS • ANALYTICAL MODEL • TESTING • HOLES • H AND J PROGRAM

120098
 HEND GT • GUYLER MC • HENDLAND MC
 CRITICAL EXPERIMENTS, MEASUREMENTS AND ANALYSES TO ESTABLISH A CRACK ARREST METHODOLOGY FOR NUCLEAR PRESSURE VESSEL STEELS. SECOND ANNUAL PROGRESS REPORT JULY 1, 1970 - JUNE 30, 1970
 DAYVILLE COLUMBUS LABS., OHIO
 OMI-3006-1950 • 150 PPS. TABS. FIGS. OCT. 1970

A TWO-DIMENSIONAL, DYNAMIC, FINITE DIFFERENCE ANALYSIS OF CRACK PROPAGATION AND ARREST, INCLUDING INERTIA FORCES AND THERMAL STRESSES IS DERIVED. CALCULATIONS BASED ON THE ONE-DIMENSIONAL ANALYSIS ARE COMPARED WITH DETAILED MEASUREMENTS OF SUB-ARREST IN TWO PLASTIC MATERIALS. THE AGREEMENTS BETWEEN CALCULATIONS AND EXPERIMENTS SUPPORT THE DYNAMIC CRACK PROPAGATION THEORY OF CRACK ARREST. CONSIDERATIONS ENTAILING THE DESIGN OF A CRACK ARREST PROPERTY MEASURING PROCEDURE, INCLUDING SPECIMEN SHAPE, SIZE, THICKNESS AND THE CRACK JUMP ARE DISCUSSED. THE INFLUENCE OF JOINT GEOMETRY ON THE PROPENSITY FOR CRACK BRANCHING IS EXAMINED. INTERACTIONS BETWEEN TEST PIECE AND THE LOADING SYSTEM ARE EXAMINED. A "TIE-JOINT" DEVICE FOR LIMITING THE INTERACTION IS APPLIED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 CRACK • PRESSURE VESSELS • STEEL • MATHEMATICAL TREATMENT • TESTING • COMPARISON • H AND J PROGRAM • HOLES

112200
 INTERGRANULAR STRESS CORRUSSION CRACKING OF SENSITIZED STAINLESS STEELS FIRST QUARTERLY PROGRESS REPORT JULY 1 - SEPTEMBER 30, 1970
 BROOKHAVEN NATIONAL LAB., UPTON, N.Y.
 OMI-3006 • 4 PPS. 3 FIGS. 2 REFS. OCT. 1970

GOAL IS TO PROVIDE A BETTER UNDERSTANDING OF THE CHEMICAL, METALLURGICAL AND ELECTROCHEMICAL FACTORS THAT CAUSE INTERGRANULAR STRESS CORRUSSION IN THE HEAT-AFFEC. ZONES OF BELTS IN STAINLESS STEELS EXPOSED TO REACTOR COOLANTS CONTAINING JAYGEN. TO PROVIDE AID WITH GUIDANCE IN DETERMINING WHICH COMBINATIONS OF MATERIAL CHEMISTRY AND WATER CHEMISTRY WILL RENDER THE MATERIAL SUSCEPTIBLE TO THIS TYPE OF ATTACK AND TO DEVELOP A RANGE OF TESTING FOR THE CORRUSION SUSCEPTIBILITY OF WELL-SENSITIZED STAINLESS STEELS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 JAYGEN • TESTING • REACTOR COOLANT • STEEL, STAINLESS • FUELING • CORROSION • MICROSTRUCTURE • JALOUS

117097

117497 CONTINUED
 GRI BA + BUSLER AD + PAPAGEORGIO BA
 RELIABILITY OF COOL PRIMARY PIPING: CRITIQUE OF STRESS-STRENGTH OVERLAY METHOD FOR CALCULATED INLET CONDITIONS
 BROOKHAVEN NATIONAL LAB., UPTON, N.Y.
 ORNL-NUREG-21002 + 25 PAGES, 3 FIGS, 10 REFS, APRIL 1976

DUE TO LACK OF OPERATING EXPERIENCE FOR LARGE LOOP CONDITIONS, RELIABILITY OF THE COOL PRIMARY PIPING MUST BE INFERRRED FROM INDIRECT ANALYSIS. THIS ANALYSIS INCLUDES MATERIALS STRESS ANALYSIS AND STRUCTURAL RELIABILITY METHODS. PARTICULAR CONCERN IS GIVEN TO THE POSSIBILITY OF A RUPTURE OF THE COOL-LEG CIRCUMFER SECTION OF THE PIPING NEAR THE HEAT EXCHANGER INLET. A CALCULATION (CUMULATIVE) PIPE RUPTURE FOLLOWED BY REACTOR SHUTDOWN LEADS TO JUDGING OF SCRAM IN THE HOT CHANNEL OF THE CORE. ORNL-NUREG-0127 ATTEMPTS TO ESTABLISH THAT LOSS OF PRIMARY PIPING SAFETY IS AN EXTREMELY RARE EVENT AND OF SUFFICIENTLY LOW PROBABILITY THAT IT NEED NOT BE INCLUDED AS PART OF THE DESIGN BASIS FOR THE CORE. THIS REPORT COVERS THE ASSESSMENT OF PIPING RELIABILITY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA 22101
 REACTOR COOLANT + STRESS ANALYSIS + ANALYSIS + RELIABILITY, SYSTEM + NUC-7 + CLINTON RIVER (LAWRENCE) + PIPES AND PIPE FITTINGS + JACOBS

117501
 INTERGRANULAR STRESS CORROSION CRACKING OF SENSITIZED STAINLESS STEELS FROM QUARTERLY PROGRESS REPORT JANUARY 1-MARCH 31, 1976
 BROOKHAVEN NAT LAB., UPTON, N.Y.
 ORNL-NUREG-50517 + 11 PAGES, 4 TABS, 7 FIGS, 3 REFS, JUNE 1976

REPORT DESCRIBES WORK CARRIED OUT IN PREVIOUS PERIODS AND PROGRESS DURING THIS REPORTING PERIOD. THE ELECTROCHEMICAL SCANNING DEVICE IS EXPLAINED AND EXPERIMENTAL RESULTS ARE PRESENTED. A PROBLEM BEING HANDLED IN IS DEVELOPMENT OF THE EQUIPMENT TO MEASURE GRAIN MATERIALS AS WELL AS THE CORROSION GRAIN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA 22101
 STEEL, STAINLESS + NUC-5 + STRESS CORROSION + R AND D PROGRAM + MICROSTRUCTURE + CHALK + JACOBS

119000
 INSPECTION OF NUCLEAR REACTOR BELONGING BY ACOUSTIC EMISSION DATA REPORT, NOVEMBER 1975-MAY 1976
 GARD INC., NILES, ILLINOIS
 NUREG-0035-2 + 26 PAGES, 5 TABS, 12 FIGS, SEPT. 1976

THE CAPABILITY OF ACOUSTIC EMISSION TO DETECT CHANGING IN NUCLEAR QUALITY PIPING BELONG IS DOCUMENTED IN THE FINAL REPORT FOR PHASE I OF THIS PROJECT (NUREG-0035-1). EXPERIMENTAL ACOUSTIC EMISSION DATA ON SIMULATED NUCLEAR PRESSURE VESSEL BELONG WHICH WERE DEVELOPED SINCE THE PREVIOUS REPORTING PERIODS ARE PRESENTED IN THIS REPORT. SOME ANALYSIS OF THIS DATA IS INCLUDED. HOWEVER, THE ANALYSIS IS INCOMPLETE DUE TO THE CURRENT UNAVAILABILITY OF CORRELATION AND METALLOGRAPHIC RESULTS. THE FIRST SIMULATED NUCLEAR PRESSURE VESSEL BELONG TEST WAS CONDUCTED AT BAWLUR AND WILCOX, BARBERTON, OHIO. THE TEST BELONG WAS A MULTIPASS (APPROX. 10 PASSES) SEMI-AUTOMATIC SUBMERGED ARC WELD (SAW). THE SECOND SIMULATED NUCLEAR PRESSURE VESSEL BELONG TEST WAS CONDUCTED AT COMBUSTION ENGINEERING, CHATTANOOGA, TENNESSEE. PLATE MATERIAL WAS A-533 STEEL AND AGAIN TYPICAL NUCLEAR BELONG PROCEDURES AND FITTING WERE FOLLOWED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA 22101
 STEEL + TESTING + BELONG + NUC-5 + REACTOR, LOR + SAFETY PROGRAM + R AND D PROGRAM + ACOUSTICS + CHALK + PRESSURE VESSELS + JACOBS + INSERVICE INSPECTION

116030
 REACTOR PRIMARY COOLANT SYSTEM PIPE RUPTURE STUDY PROGRESS REPORT NO. 33 JANUARY-JUNE 1975
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 GEAP-10207-33 + 113 PAGES, 23 TABS, 77 FIGS, OCT. 1975

STUDY IS DESIGNED TO EXPAND THE UNDERSTANDING OF FAILURE-CAUSING MECHANISMS AND TO PROVIDE IMPROVED CAPABILITY FOR EVALUATING REACTOR PIPING SYSTEMS TO MINIMIZE THE PROBABILITY OF FAILURE. ANALYTICAL AND EXPERIMENTAL EFFORTS (PHASE I) WERE STARTED IN 1963. THIS REPORT SUMMARIZES: (1) BASIC FATIGUE CRACK GROWTH RATE STUDIES ON LOW PRESSURE PIPING MATERIALS IN A SIMULATED BOR PHENANT COOLANT ENVIRONMENT, (2) AT-REACTOR TESTS OF THE EFFECT OF PHENANT COOLANT ENVIRONMENT ON THE FATIGUE BEHAVIOR OF PIPING STEELS, (3) STUDIES DIRECTED AT QUANTIFYING SCC SENSITIZATION IN TYPE 304 STAINLESS STEEL, (4) SUPPORT STUDIES TO CHARACTERIZE THE ELECTROCHEMICAL POTENTIAL BEHAVIOR OF A TYPICAL BOR PHENANT WATER ENVIRONMENT AND (5) SPECIAL TESTS TO SIMULATE FRACTURE SURFACE CHARACTERISTICS OF 2205 FILL FRACTURES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA 22101
 REACTOR, BOR + FAILURE + BELONG + REACTOR COOLANT + STEEL, STAINLESS + NUC-5 + WEIGHT/WEIGHTMENT + FATIGUE + CRACK + PIPES AND PIPE FITTINGS + JACOBS

117726
 REACTOR PRIMARY COOLANT SYSTEM PIPE RUPTURE STUDY PROGRESS REPORT NO. 33 JANUARY-MARCH 1976
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.

117726 CONTINUED
 ORAD-10207-25 P. 20 PPS, 3 TABS, 10 FIGS, MAY 1970

A FATIGUE CRACK INITIATION TEST WITH 12 SPECIMENS BEING RUN IN 500 PSI AIR WAS COMPLETED. THE DATA ARE BEING PUT INTO A TYPICAL REPORT ON FATIGUE CRACK INITIATION. AN ENVIRONMENTAL FATIGUE CRACK GROWTH TEST, BEGUN IN THE PREVIOUS REPORT PERIOD, WAS COMPLETED. THE RESULTANT FATIGUE CRACK GROWTH DATA ARE COMPARED TO ASME SECTION 81 DATA AND COMPARABLE DATA FROM OTHER INVESTIGATORS. THE ORDER OF INITIATION OF CRACKS IN TYPE-304 STAINLESS STEEL WAS EVALUATED ELECTROCHEMICALLY. SAMPLES FROM OUTF WELDS ARE BEING PREPARED FOR STRESS CORROSION CRACKING EVALUATION.

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

REACTOR, ORAD - FAILURE, PIPE - STEEL, STAINLESS - NUC-S - HEATING, LWR - FUEL, FATIGUE - SAFETY PROGRAM - STRESS CORROSION - ORAD - PROGRAM - CORROSION - PIPES AND PIPE FITTINGS - JACOBS

117710
 DALLAS: FJ
 THREE-DIMENSIONAL ELASTIC-PLASTIC STRESS AND STRAIN ANALYSIS FOR FRACTURE MECHANICS: COMPLEX GEOMETRIES
 NUCLEAR RESEARCH CORP., POLE ALTO, CALIF.
 ORAD-10207-25 P. 27 PPS, 10 FIGS, 9 REFS, NOV. 1970

A COMPUTER PROGRAM, NUC-30, HAS BEEN COMPLETED AND WAS USED TO ANALYZE A CYLINDRICAL PRESSURE VESSEL WITH A NOZZLE ASSEMBLY. A METHOD FOR GENERATING CRACK TIP ELEMENTS WAS DEVELOPED AND A MODEL WAS CREATED FROM A CYLINDRICAL PRESSURE VESSEL WITH A NOZZLE AND AN INJECTED PLAN AT THE INSIDE NOZZLE CORNER. THE NUC-30 PROGRAM WAS AGAIN USED TO ANALYZE THIS PLANNED MODEL. DOCUMENTATION FOR THE USE OF THE NUC-30 COMPUTER PROGRAM HAS BEEN INCLUDED AS AN APPENDIX.

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

COMPUTER PROGRAM - STRESS ANALYSIS - PRESSURE VESSELS - CYLINDER - FRACTURE TOUGHNESS - MODEL - PLAN - NUC-30

115700
 LOSS FJ
 FRACTURE TOUGHNESS OF PRESSURE VESSEL MATERIALS FOR LIGHT WATER REACTOR, PROGRESS REPORT ENDING 29 FEBRUARY 1970
 NAVAL RESEARCH LAB., WASHINGTON, D.C.
 ORAD-10207-25 P. 33 PPS, 5 TABS, 11 FIGS, 13 REFS, MAY 1970

THE KIC (K_{1c}) VS TEMPERATURE TRENDS OF THE PROGRAM MATERIALS HAVE BEEN CHARACTERIZED WITH PRECRACKED CARRY AND 1-IN. BOND SPECIMENS LOADED AT THE SAME RATE (17). WHILE THE RESULTS FROM BOTH SPECIMENS AGREE AT LOW TEMPERATURES, THE PRECRACKED CARRY TEST APPEARS TO OVERESTIMATE THE TOUGHNESS IN THE BRITTLE/DUCTILE TRANSITION REGION. AN INCONSISTENCY FOR THE 1-IN. BOND SPECIMEN HAS BEEN NOTED BETWEEN KIC VALUES COMPUTED FROM: (A) THE FRACTURE LOAD AND (B) A J-INTEGRAL PROGRAM USING SPECIFIC ENERGY TO THE TIME OF FRACTURE. AN APPROXIMATE THEORETICAL ANALYSIS OF THE BONDING MOMENTS AND FORCES DEVELOPED DURING IMPACT OF A NOTCHED BEAM IN THREE-POINT BENDING HAS BEEN COMPLETED.

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

NOT DATA - STEEL - TESTING - REACTOR, LWR - SAFETY PROGRAM - R AND D PROGRAM - FRACTURE TOUGHNESS - POST - PRESSURE VESSELS - JACOBS

115573
 LOSS FJ
 STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS, PROGRESS REPORT ENDING 29 FEBRUARY 1970
 NAVAL RESEARCH LAB., WASHINGTON, D.C.
 ORAD-10207-25 P. 49 PPS, 8 TABS, 24 FIGS, 12 REFS, AUG. 20, 1970

DESCRIBES PROGRESS IN THE FOLLOWING AREAS: (A) FATIGUE CRACK PROPAGATION IN REACTOR PRESSURE VESSEL STEELS IN AIR ENVIRONMENT, (B) DYNAMIC FRACTURE TOUGHNESS OF 1-IN. (25-MM) AND PRECRACKED CARRY-7 BOND SPECIMENS UNDER IMPACT LOADING, (C) POSTINITIATION CRACK GROWTH AND PROPERTIES RECOVERY IN REACTOR VESSEL STEELS, (D) FRACTURE CONTINUITY, (E) VARIABLE RESISTANCE OF STRUCTURAL STEELS TO NAVIGATION EMBEZZLEMENT, AND (F) THE INITIAL PROGRAM PLAN TO INVESTIGATE THE PHENOMENA OF BOND PRESTRESS AND PLASTIC NET LIGAMENT IN SUPPORT OF THERMAL STUDIES.

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

NOT DATA - REACTOR, LWR - STEEL - TESTING - NUC-S - IRRADIATION TESTING - SAFETY PROGRAM - R AND D PROGRAM - GROWTH/DEVELOPMENT - FATIGUE - CRACK - PRESSURE VESSELS - JACOBS

119000
 LOSS FJ
 STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS, PROGRESS REPORT ENDING 31 MAY 1970
 NAVAL RESEARCH LAB., WASHINGTON, D.C.
 ORAD-10207-25 P. 46 PPS, 3 TABS, 10 FIGS, 20 REFS, SEPT. 1970

DESCRIBES PROGRESS IN THE FOLLOWING AREAS: (A) BOND PRESTRESS INVESTIGATIONS RELATED TO THERMAL SHOCK OF A VESSEL; (B) FRACTURE TOUGHNESS DEGRADATION OF COMMERCIAL WELD DEPOSITS FOLLOWING LOW

110600 CONTINUED

PLASMA IRRADIATION: (C) AERIAL SPECTROGRAPHY STUDY ON THE MECHANISM OF POLYMERIZATION CONTINUOUS TO STEADY IRRADIATION SENSITIVITY; (D) ELASTIC ANALYSIS OF THREE-POINT BEND SPECIMENS SUBJECT TO IMPACT LOADING; AND (E) - IMPACT OF REACTOR MATERIALS RESEARCH UNDERWAY AT THE FUTURE LABORATORIES.

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

NOT DATA • IRRADIATION DAMAGE • STEEL • IMPACT SHOCK • NUC-S • REACTOR, LOW • SAFETY PROGRAM • A AND B PROGRAM • FRACTURE TOUGHNESS • THERMAL TRANSIENT • PRESSURE VESSELS • JACOBS

110721

MEMORANDUM

ANALYTICAL RELATIONS BETWEEN ELASTIC-PLASTIC FRACTURE CRITERIA

OAK RIDGE NATIONAL LABORATORY, TENNESSEE

OAK RIDGE/IN-27 • 15 PPS, 2 FIGS, 10 REFS, JULY 1970

THE SOLUTION OF THE HERRMANN CRACK OPENING DISPLACEMENT (COD) DESIGN CURVE RECENTLY PROPOSED IN THE U.S. AS A BASIS FOR DETERMINING ALLOWABLE CRACK SIZES IS DERIVED FROM THE EQUIVALENT STRAIN APPROXIMATION FOR THE J INTEGRAL. IT IS ALSO SHOWN THAT ANOTHER APPROXIMATION FOR THE J INTEGRAL RECENTLY PROPOSED BY WESTERMARK IS MATHEMATICALLY EQUIVALENT TO THE HERRMANN COD APPROXIMATION.

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

DESIGN CRITERIA • EQUATION • STEEL • ELASTICITY • NUC-S • MATHEMATICAL STUDY • PLASTICITY • A AND B PROGRAM • FRACTURE TOUGHNESS • FLOW • LOFTING • CRACK

117500

MEMORANDUM

HEAVY-SECTION STEEL TECHNOLOGY PROGRAM QUARTERLY PROGRESS REPORT FOR JANUARY-MARCH 1970

OAK RIDGE NATIONAL LABORATORY, TENNESSEE

OAK RIDGE/IN-20 • 75 PPS, FIGS, REFS, JULY 1970

HEST IS AN ENGINEERING RESEARCH ACTIVITY DEVOTED TO EXTENDING AND DEVELOPING THE TECHNOLOGY FOR ASSESSING THE RISK OF SAFETY AGAINST FRACTURE OF THE THICK-WALLED STEEL PRESSURE VESSELS USED IN LWR'S. THE PRINCIPAL AREA OF RESEARCH IS THE DETERMINATION AND STRUCTURAL INTEGRITY OF VESSELS CONTAINING CRACKLINE FLAWS. CURRENT WORK COVERS FRACTURE MECHANICS ANALYSIS, EFFECT OF HIGH-TEMPERATURE PRIMARY AND/OR WATER ON CRACK GROWTH, INVESTIGATIONS OF IRRADIATED MATERIALS, PRESSURE VESSEL INVESTIGATIONS, THERMAL SHOCK STUDIES, AND HELD HEAT-AFFECTED-ZONE CHARACTERIZATION IN MATHEMATICAL CALCULATIONS, SPECIMEN PREPARATION, AND TESTING IS DESCRIBED.

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

STEEL • REACTOR COOLANT • WELDING • HIGH TEMPERATURE • NUC-S • IRRADIATION TESTING • STRUCTURAL INTEGRITY • A AND B PROGRAM • FRACTURE TOUGHNESS • GROWTH/DEVELOPMENT • CRACK • HEAT • PRESSURE VESSELS • JACOBS

115201

PER 66

JUNE 1970 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT OAK RIDGE NATIONAL

LABORATORY

OAK RIDGE NATIONAL LABORATORY, TENNESSEE

OAK RIDGE/IN-00 • 27 PPS, JULY 1970

THIS PROGRESS REPORT CONTAINS SHORT (1 TO 4 PAGES) SUMMARIES OF THE ACCOMPLISHMENTS DURING THE MONTH ON THE FOLLOWING WORK PROJECTS: HEAVY SECTION STEEL TECHNOLOGY PROGRAM, FISSION PRODUCT DATA AND LOSS ENERGY RELEASE, LOSS RELEASE FROM LOW FUEL, RADIATION WIND TESTS, NUCLEAR SAFETY INFORMATION CENTER, FOR OIL/GAS HEAT TRANSFER-SEPARATE EFFECTS, ZIRCALOY FUEL CLADDING CORROSION STUDIES, ZIRCALOY METAL-WATER CORROSION KINETICS, NUCLEAR RELEASE AND TRANSPORT FROM LOW FUEL, NUCLEAR SAFETY ANALYSIS AND RESEARCH, AND DESIGN CRITERIA FOR PIPING AND NOZZLES.

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

ACCIDENT ANALYSIS • AEROSOL • DESIGN CRITERIA • FISSION PRODUCT RELEASE • FUEL, PWR ELEMENT • HEAT TRANSFER • ACCIDENT, LOSS OF COOLANT • METAL WATER REACTION • OXIDATION • ZIRCALOY • FISSION PRODUCT TRANSPORT • GROSS DATA • GROSS LOSS • REACTOR, LOW • FAILURE, LOADING • ZIRCALOY • SAFETY PROGRAM • OIL/GAS • NUC • ENERGY LEVEL • OR AND B PROGRAM • REACTION, METAL • REACTION, LOW • HEAT • PIPES AND PIPE FITTINGS

110214

MEMORANDUM • ROBINSON GC • HOLD PV

TEST OF 6-IN.-THICK PRESSURE VESSELS, SERIES 2: INTERMEDIATE TEST VESSEL V-7

OAK RIDGE NATIONAL LABORATORY, TENNESSEE

OAK RIDGE/IN-1 • 117 PPS, FIGS, REFS, AUG. 1970

THIS WAS A CRACK-INITIATION FRACTURE TEST. THE VESSEL WAS HEATED TO 515 DEG C AND PREHEATED HYDRAULICALLY UNTIL LEAKAGE THROUGH THE FLAW TERMINATED THE TEST AT A PEAK PRESSURE OF 107 MPa (15,300 PSI). FRACTURE TOUGHNESS DATA OBTAINED BY TESTING FRACTURE MECHANICS AND COMPACT-TENSION SPECIMENS MACHINED FROM A PROLONGATION OF THE HYDRAULIC TEST SHALL BE USED IN POSTTEST ANALYSES OF THE FLAWED VESSEL. THE VESSEL, AS EXPECTED, DID NOT LEAK. UPON DEPRESSURIZATION, THE RUPTURED LIGAMENT CLOSED SO AS TO MAINTAIN STATIC PRESSURE WITHIN LEAKAGE AT ABOUT 120 MPa.

110216 OCEANICUS
110-700-0511-

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

TESTING - NRC-5 - C AND D PROGRAM - FRACTURE MECHANICS - LEAK - CORROSION - WELD - PRESSURE VESSELS - JACOBS

110063
CALLAHAN JR. - DIVISION OF
STRESS ANALYSIS OF PERFORATED FLAT PLATES UNDER IN-PLANE LOADING
OAK RIDGE NATIONAL LABORATORY, TENNESSEE
ORNL/RRR/62-2 P. 63 PPS. 60 PAGES. 4 REFS. AUG. 1970

SPRINT-CAGE FLAT PLATES HAVING ONE, TWO, THREE, AND FIVE EQUALLY SPACED HOLES WERE TESTED UNDER UNIFORM AND 1:1 DIAGONAL LOADING CONDITIONS. EXPERIMENTAL RESULTS WERE COMPARED WITH CORRESPONDING CALCULATED VALUES OBTAINED USING THE CO-ORDINATE METHOD TABLES. THE STUDY WAS CONDUCTED (1) TO OBTAIN RELIABLE EXPERIMENTAL DATA FOR DESIGN OF PERFORATED PLATES IN VESSELS UNDER LARGE MODELS OF CORRUPTION, (2) TO EVALUATE CAPABILITY OF THE CO-ORDINATE METHOD, (3) DEVELOPMENT OF A METHOD OF ANALYSIS FOR CLUSTERS OF HOLES ATTACHED TO PRESSURE VESSELS, AND (4) TO INVESTIGATE THE STRESS STATES IN REGIONS OF CLOSELY SPACED HOLES. THE COMPARISONS OF CALCULATED AND EXPERIMENTAL STRESSES WERE IN GOOD AGREEMENT.

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

STRESS ANALYSIS - PLATE - PIPES AND PIPE FITTINGS - NOZZLE - PRESSURE VESSELS - C AND D PROGRAM - COMPUTED PROGRAM - TESTING - CORRUPTION

110037
PER 64
JULY 1970 MONTHLY MEETING FOR OFFICE OF NUCLEAR REGULATION RESEARCH PROGRAMS AT OAK RIDGE NATIONAL LABORATORY
OAK RIDGE NATIONAL LABORATORY, TENNESSEE
ORNL/RRR/60-00 P. 25 PPS. AUG. 1970

THIS PROGRESS REPORT CONTAINS SUMMARY OF 100 PAGES SUMMARIES OF THE ACCIDENTS DURING THE MONTH OF THE FOLLOWING CASES: PROJECTS: LEAK SECTION STEEL TECHNOLOGY PROGRAM, FISSILE PRODUCT WASTE AND WASTE RELEASE, LEAK RELEASE FROM LOW FUEL, NUCLEAR WASTE TESTS, NUCLEAR SAFETY INFORMATION CENTER, FUEL ELEMENTS HEAT TRANSPORT-SEPARATE EFFECTS, ZIRCALOY FUEL CLADDING CORROSION STUDIES, ZIRCALOY FUEL-WATER OXIDATION KINETICS, NUCLEAR WASTE AND TRANSPORT FROM LOW FUEL, NUCLEAR SAFETY ANALYSIS AND RESEARCH AND DESIGN CRITERIA FOR PIPING AND NOZZLES.

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

ACCIDENT ANALYSIS - WASTE - DESIGN CRITERIA - FISSILE PRODUCT RELEASE - FAILURE, FUEL ELEMENT - HEAT TRANSPORT - METAL WASTE RELEASE - OXIDATION - ZIRCALOY - FISSILE PRODUCT TRANSPORT - NRC-2 - WASTE WASTE - GROSS CORROSION - NRC-5 - REACTION, LEAK - FAILURE, CLADDING - ZIRCALOY - SAFETY PROGRAM - NRC-3 - CORROSION - NRC - ENERGY LEVEL - C AND D PROGRAM - REACTION, WASTE - REACTION, LEAK - NRC-6 - NRC-7 - WASTE - PIPES AND PIPE FITTINGS

110623
CONVERTER NO
PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO A FUEL-CLADDING INTERFACIAL STRESS EXPERIMENT, T-1 AND T-2
OAK RIDGE NATIONAL LABORATORY, TENNESSEE
ORNL/RRR/60-01 P. 203 PPS. 1161. REFS. SEPT. 1970

STUDIES THIS FAR HAVE INCLUDED FRACTURE MECHANICS ANALYSIS OF TYPICAL WELDS, THE DESIGN AND CONSTRUCTION OF A FUEL-CLADDING TEST FACILITY, DETERMINATION OF MATERIAL PROPERTIES FOR TEST SPECIMENS, AND TWO FUEL-CLADDING EXPERIMENTS. CALCULATIONS INDICATED THAT UNDER SOME CIRCUMSTANCES CRACK PROPAGATION COULD BE EXPECTED AND THAT EXPERIMENTS SHOULD BE CONDUCTED. IN THE FIRST EXPERIMENT, INITIATION WAS NOT EXPECTED. CRACK PROPAGATION DID INITIALLY TAKE PLACE IN THE SECOND EXPERIMENT AS EXPECTED. THESE RESULTS TEND TO VALIDATE THE METHODS OF ANALYSIS. ADDITIONAL EXPERIMENT, WILL BE REQUIRED BEFORE THE VALIDITY CAN BE DETERMINED WITH REASONABLE ACCURACY.

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

ACCIDENT, LOSS OF COOLANT - TEST, PRESSURE VESSEL - FLEXIBLE PROPERTY - THERMAL ANALYSIS - THERMAL CORRUPTION - NRC-5 - REACTION, LEAK - FREQUENCY LOADING - FUEL-CLADDING - CORROSION - PRESSURE VESSELS - JACOBS

110000
OCEANICUS
HEAT-SEALING STEEL TECHNOLOGY PROGRAM MONTHLY PROGRESS REPORT FOR APRIL-JUNE 1970
OAK RIDGE NATIONAL LABORATORY, TENNESSEE
ORNL/RRR/60-00 P. 105 PPS. 1061. REFS. JUL. 1970

HST IS AN ENGINEERING RESEARCH ACTIVITY DEVOTED TO EXTENDING AND DEVELOPING THE TECHNOLOGY FOR ASSESSING THE RISK OF SAFETY CRITICAL FRACTURE OF THE THICK-WALLED STEEL PRESSURE VESSELS USED IN LWRs. THE PRINCIPAL AREA OF RESEARCH IS THE DESIGN AND STRUCTURAL INTEGRITY OF VESSELS CONTAINING LOCALIZED FLAWS. CURRENT WORK INCLUDES FRACTURE MECHANICS ANALYSIS, EFFECT OF HIGH-

117900

TEMPERATURE PROTECTANT POLYMER MARKER ON CRACK SURFACES INDICATING THE PRESENCE OF IRRADIATED MATERIALS. PRESENTATION OF THE INFORMATION, AND THERMAL ANALYSIS. ANALYSIS-INDICATED FACTORS OF MODEL CRACK LENGTH AND CRACK MECHANISM IN PLASTIC. MARKER IS USED TO INDICATE CRACK MECHANISM. THERMAL ANALYSIS. POLYMER MARKER ON CRACK SURFACES INDICATING THE PRESENCE OF IRRADIATED MATERIALS. PRESENTATION OF THE INFORMATION, AND THERMAL ANALYSIS. ANALYSIS-INDICATED FACTORS OF MODEL CRACK LENGTH AND CRACK MECHANISM IN PLASTIC. MARKER IS USED TO INDICATE CRACK MECHANISM.

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

NOTE - A AND B PROGRAMS - POLYMER MARKER ON CRACK SURFACES INDICATING THE PRESENCE OF IRRADIATED MATERIALS. PRESENTATION OF THE INFORMATION, AND THERMAL ANALYSIS. ANALYSIS-INDICATED FACTORS OF MODEL CRACK LENGTH AND CRACK MECHANISM IN PLASTIC. MARKER IS USED TO INDICATE CRACK MECHANISM.

117972

FEB 66

SEPTEMBER 1976 MONTHLY MEMORANDUM FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT THE NUCLEAR NATIONAL LABORATORY

THE NUCLEAR NATIONAL LABORATORY, TENNESSEE
 DATE: SEPTEMBER 1976

PRESENTATION OF THE INFORMATION, AND THERMAL ANALYSIS. ANALYSIS-INDICATED FACTORS OF MODEL CRACK LENGTH AND CRACK MECHANISM IN PLASTIC. MARKER IS USED TO INDICATE CRACK MECHANISM. THERMAL ANALYSIS. POLYMER MARKER ON CRACK SURFACES INDICATING THE PRESENCE OF IRRADIATED MATERIALS. PRESENTATION OF THE INFORMATION, AND THERMAL ANALYSIS. ANALYSIS-INDICATED FACTORS OF MODEL CRACK LENGTH AND CRACK MECHANISM IN PLASTIC. MARKER IS USED TO INDICATE CRACK MECHANISM.

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

NOTE - A AND B PROGRAMS - POLYMER MARKER ON CRACK SURFACES INDICATING THE PRESENCE OF IRRADIATED MATERIALS. PRESENTATION OF THE INFORMATION, AND THERMAL ANALYSIS. ANALYSIS-INDICATED FACTORS OF MODEL CRACK LENGTH AND CRACK MECHANISM IN PLASTIC. MARKER IS USED TO INDICATE CRACK MECHANISM.

120200

FEB 66

SEPTEMBER 1976 MONTHLY MEMORANDUM FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT THE NUCLEAR NATIONAL LABORATORY

THE NUCLEAR NATIONAL LABORATORY, TENNESSEE
 DATE: SEPTEMBER 1976

PRESENTATION OF THE INFORMATION, AND THERMAL ANALYSIS. ANALYSIS-INDICATED FACTORS OF MODEL CRACK LENGTH AND CRACK MECHANISM IN PLASTIC. MARKER IS USED TO INDICATE CRACK MECHANISM. THERMAL ANALYSIS. POLYMER MARKER ON CRACK SURFACES INDICATING THE PRESENCE OF IRRADIATED MATERIALS. PRESENTATION OF THE INFORMATION, AND THERMAL ANALYSIS. ANALYSIS-INDICATED FACTORS OF MODEL CRACK LENGTH AND CRACK MECHANISM IN PLASTIC. MARKER IS USED TO INDICATE CRACK MECHANISM.

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

NOTE - A AND B PROGRAMS - POLYMER MARKER ON CRACK SURFACES INDICATING THE PRESENCE OF IRRADIATED MATERIALS. PRESENTATION OF THE INFORMATION, AND THERMAL ANALYSIS. ANALYSIS-INDICATED FACTORS OF MODEL CRACK LENGTH AND CRACK MECHANISM IN PLASTIC. MARKER IS USED TO INDICATE CRACK MECHANISM.

120300

FEB 66

SEPTEMBER 1976 MONTHLY MEMORANDUM FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT THE NUCLEAR NATIONAL LABORATORY

THE NUCLEAR NATIONAL LABORATORY, TENNESSEE
 DATE: SEPTEMBER 1976

PRESENTATION OF THE INFORMATION, AND THERMAL ANALYSIS. ANALYSIS-INDICATED FACTORS OF MODEL CRACK LENGTH AND CRACK MECHANISM IN PLASTIC. MARKER IS USED TO INDICATE CRACK MECHANISM. THERMAL ANALYSIS. POLYMER MARKER ON CRACK SURFACES INDICATING THE PRESENCE OF IRRADIATED MATERIALS. PRESENTATION OF THE INFORMATION, AND THERMAL ANALYSIS. ANALYSIS-INDICATED FACTORS OF MODEL CRACK LENGTH AND CRACK MECHANISM IN PLASTIC. MARKER IS USED TO INDICATE CRACK MECHANISM.

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

NOTE - A AND B PROGRAMS - POLYMER MARKER ON CRACK SURFACES INDICATING THE PRESENCE OF IRRADIATED MATERIALS. PRESENTATION OF THE INFORMATION, AND THERMAL ANALYSIS. ANALYSIS-INDICATED FACTORS OF MODEL CRACK LENGTH AND CRACK MECHANISM IN PLASTIC. MARKER IS USED TO INDICATE CRACK MECHANISM.

120300

FEB 66

SEPTEMBER 1976 MONTHLY MEMORANDUM FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT THE NUCLEAR NATIONAL LABORATORY

120130 CONTINUOUS
LABORATORY
OAK RIDGE NATIONAL LABORATORY, TENNESSEE
ORNL/TN-46/70-76 v. 25 PPS. DEC. 1970

THIS PROJECT REPORT CONTAINS SHORT (1 TO 4 PAGES) SUMMARIES OF THE ACCOMPLISHMENTS DURING THE MONTH IN THE FOLLOWING ORNL PROJECTS: HEAVY SECTION STEEL TECHNOLOGY PROGRAM, FISSION PRODUCT BETA AND GAMMA ENERGY RELEASE, LOCA RELEASE FROM LHM PNL, MULTIJET GUNST TESTS, NUCLEAR SAFETY INFORMATION CENTER, PWR HELEDOX HEAT TRANSFER-SEPARATE EFFECTS, ZIRCALOY FUEL CLADDING CORROSION STUDIES, ZINCUMINUM METAL-WATER OXIDATION KINETICS, AIRBORNE RELEASE AND TRANSPORT FROM LHM PNL, NRC SAFETY ANALYSIS AND RESEARCH, AND DESIGN CRITERIA FOR PIPING AND NOZZLES.

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

REACTOR, LHM • MSST • ENERGY LEVEL • FISSION PRODUCT BETA • GROSS BETA • GROSS GAMMA • FAILURE • FUEL ELEMENT • NSIC • OXIDATION • HEAT TRANSFER • ZIRCALOY • FAILURE, CLADDING • METAL WATER REACTION • ZINCUMINUM • OXIDATION • REACTOR, LHM • AEROSOL • FISSION PRODUCT TRANSPORT • NEUTRON, HIGH • ACCIDENT ANALYSIS • DESIGN CRITERIA • PIPES AND PIPE FITTINGS • OR AND B PROGRAM • OSARET PROGRAM • RMC-2 • RMC-3 • RMC-5 • RMC-7 • TAC-8

117033
MAXWELL RL • MULLANO RB • KARIANTHRA JM
EXPERIMENTAL STRESS ANALYSIS OF THE ATTACHMENT REGION OF HEMISPHERICAL NOZZLE WITH ATTACHED NOZZLES. PART 2. RADIAL NOZZLE 7.875 IN. O.D.-7.500 IN. I.D. ZERO PENETRATION
UNIVERSITY OF TENNESSEE, KNOXVILLE
ORNL/SUB/2203-v. 145 PPS. 60 TABS. 30 FIGS. JULY 1970

PART OF A COOPERATIVE EFFORT WITH INDUSTRY TO CONFIRM THE ADEQUACY OF PRESENT CODES AND STANDARDS, ANALYTICAL METHODS AND DESIGN CRITERIA, AND/OR THE DEVELOPMENT OF NEW RULES AND METHODS FOR ASSESSING THE SAFETY OF NUCLEAR PRESSURE VESSEL AND PIPING-SYSTEM DESIGN. THIS REPORT PRESENTS RESULTS OF INVESTIGATIONS CONDUCTED ON A RADIAL ATTACHED NOZZLE 7.875 IN. O.D., 7.5 IN. I.D. AND FINISHED FLUSH WITH THE INNER SURFACE OF THE HEMISPHERE. STRESS VALUES WERE DETERMINED FOR THE FOLLOWING LOADINGS: (1) INTERNAL PRESSURE APPLIED TO THE HEMISPHERE AND NOZZLE ASSEMBLY, (2) AN AXIAL LOAD APPLIED COLLINEAR WITH NOZZLE, (3) A PURE TORSION APPLIED IN THE RADIAL PLANE OF THE NOZZLE, AND (4) A PURE BENDING MOMENT, OR AXIAL CUPLE, APPLIED TO THE NOZZLE.

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

CODES AND STANDARDS • DESIGN CRITERIA • STRESS ANALYSIS • ANALYTICAL TECHNIQUE • RMC-5 • NOZZLE • PRESSURE VESSELS • PIPES AND PIPE FITTINGS • JACOBS

6. NRC-6 WATER REACTOR SAFETY RESEARCH, SITE SAFETY RESEARCH

100000
 PROCEEDURES FOR EVALUATION OF VIBRATORY GROUND MOTIONS OF SOIL DEPOSITS AT NUCLEAR POWER PLANT SITES
 SHANNON & BILSON INC. • ARCADIAN ASSOCIATES, SEATTLE
 NUREG-75/072 • 107 PPS, 33 FIGS, JUNE 1975

INDICATES HOW LOCAL SOIL CONDITIONS CAN BE CONSIDERED IN THE DEVELOPMENT OF SEISMIC INPUT CRITERIA THAT ARE CONSISTENT WITH USUAL CRITERIA. CONTAINS AN IN-DEPTH DISCUSSION OF BACKGROUND INFORMATION AND CURRENT TECHNOLOGY FOR ESTABLISHING EARTHQUAKE GROUND MOTION CRITERIA USING SITE-INDEPENDENT AND SITE-DEPENDENT TECHNIQUES. IN ADDITION, GEOTECHNICAL INVESTIGATIONS REQUIRED AS A BASIS FOR ESTABLISHING SUCH CRITERIA ARE BRIEFLY SUMMARIZED. A RECOMMENDED PROCEDURE FOR DETERMINING SEISMIC INPUT CRITERIA FOR NUCLEAR PLANTS IS THEN PROVIDED AND IS ILLUSTRATED WITH AN EXAMPLE APPLICATION.

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

DESIGN CRITERIA • DESIGN • SITING • SOIL PROPERTY • EARTHQUAKE ENGINEERING • GROUND MOTION • ELECTRONICS • SEISMIC DESIGN • AGENCY: NRC • JACQUES

115181
 OVERVIEW OF NUCLEAR FACILITIES. SECTIONS FROM "NUCLEAR SAFETY"
 OAK RIDGE NATIONAL LABORATORY, TENNESSEE. NUCLEAR SAFETY INFORMATION CENTER
 NUREG/NSIC-110 • 207 PPS, 100S, FIGS, REFS, JULY 1976

THIS REPORT CAPTURES SITING POLICY AND PRACTICE FOR NUCLEAR POWER PLANTS AS DEVELOPED IN THE U.S. AND ABROAD. TWENTY-FOUR ARTICLES FROM NUCLEAR SAFETY ON THIS GENERAL TOPIC ARE REPRINTED SINCE THEY PROVIDE A VALUABLE REFERENCE SOURCE. THE APPENDICES ALSO INCLUDE REPRINTS OF SOME RELEVANT REGULATORY RULES AND GUIDES ON SITING. ADVANTAGES AND DISADVANTAGES OF SEVERAL SITING CONCEPTS SUCH AS UNDERGROUND CONTAINMENT, OFFSHORE SITING, AND NUCLEAR ENERGY PARKS ARE ADDRESSED. OTHER TOPICS INCLUDING SITE CRITERIA, RISK CRITERIA, AND NUCLEAR SMOKE CRITERIA.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA. 22161
 (89-75/FULL-SIZE COPY, \$2.25/PP)

CONTAINMENT, UNDERGROUND • DESIGN • SITING, REACTOR • REACTOR, POWER • SITING, MULTIPLE • POPULATION DISTRIBTION • SITING, OFF SHORE • SITE CRITERIA • SITING, UNDER • JACQUES

7. NRC-7 ADVANCED REACTOR SAFETY RESEARCH. FAST REACTORS

117079

ROBERT, JR. • PIZZICA PA • HALLIDAY
PHYSICS AND PUMP COASTDOWN CALCULATIONS FOR A MODEL OF A 430J ONE SHUT-TWOED LUMP
ARGONNE NATIONAL LABORATORY, ILLINOIS
NRL-70-77 • 21 PPS, 0 TABS, 4 FIGS, 11 REFS, JUNE 1976

PUMP COASTDOWN CALCULATIONS FOR A MODEL OF A 430J ONE LUMP SIMILAR TO ONE STUDIED BY MCGHEE ET AL GAVE SODIUM BOILING HEATIVITY RAMP RATES OF ABOUT 200°C/SEC INSTEAD OF UP TO 400°C/SEC OBTAINED PREVIOUSLY. THIS DISCREPANCY HAS NOT BEEN SATISFACTORILY EXPLAINED. IF HYDRAULIC COUPLING SODIUM CHANNELS IS NEGLECTED, AS WAS THE CASE IN THE CALCULATIONS OF MCGHEE ET AL, RAMP RATES UP TO 400°C/SEC ARE FOUND IN SOME CASES.

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

SODIUM • ANALYTICAL MODEL • N AND D PROGRAM • ONEACTIVITY EFFECT, VOID • REACTOR, LUMP • SODIUM • NRC-7 • JACOBS

117080

CYRULIS P • ALLWEIER JR • LAMM JR • BERNING RS • RITCHIE AL
SOME INSIGHTS ON THE EFFECT OF ENGINEERED SAFETY FEATURES IN THE CASE OF HYPOTHETICAL LUMP ACCIDENTS
BATTILLE CALOROUS LAB., ILLINOIS
NRL-70-78 • 04 PPS, 0 TABS, 14 FIGS, 01 REFS, JULY 21, 1976

ANALYSES HAVE BEEN PERFORMED TO EVALUATE THE EFFECT OF CERTAIN ENGINEERED SAFETY FEATURES ON REDUCING THE RISK OF HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS IN LUMPS. CONSIDERATION HAS BEEN LIMITED TO ACCIDENTS WHICH ARE ASSUMED TO RESULT IN THE CORE MELTDOWN OF THE REACTOR VESSEL. THE CONSEQUENCES OF SUCH ACCIDENTS HAVE BEEN COMPARED FOR TWO PRINCIPAL PLANT DESIGNS: ONE WHICH DOES NOT INCLUDE ANY SPECIFIC PROVISIONS TO ACCOMMODATE SUCH AN ACCIDENT, AND THE OTHER WHICH INCLUDES AN EX-VESSSEL CORE CATCHER AND A REACTOR CAVITY LEAK SYSTEM. THE CONSEQUENCES OF THE VARIOUS EVENT SEQUENCES CONSIDERED WERE EVALUATED IN TERMS OF THE FRACTION OF THE REACTOR INVENTORY OF ACTIVE MATERIALS RELEASED TO THE ENVIRONMENT AS A FUNCTION OF TIME.

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

ACCIDENT ANALYSIS • FISSILE PRODUCT RELEASE • CORE MELTDOWN • REINFORCED SAFETY FEATURE • REACTOR, LUMP • CORE CATCHER • BENEFIT VS RISK • NRC-7 • JACOBS • ACCIDENT, CORE DISRUPTIVE

116343

YANG JR
AN ANALYSIS OF TRANSIENT THERMAL RESPONSE IN THE UPPER OUTLET PLENUM OF AN LUMP
ORNL-70-79 • 04 PPS, 0 TABS, 10 FIGS, 01 REFS, JUNE 1976

A TWO-ZONE MIXING MODEL BASED ON THE LUMPED-PARAMETER APPROACH WAS DEVELOPED FOR THE ANALYSIS OF TRANSIENT THERMAL RESPONSE IN THE UPPER OUTLET PLENUM OF AN LUMP. THE RESULTS INDICATE THAT EFFECTS OF FLOW STRATIFICATION, CLOSER HEIGHT, INITIAL HEAT CAPACITY AND BYPASS PLENUM ARE IMPORTANT FOR TRANSIENT SODIUM TEMPERATURE CALCULATION. THERMAL EXPANSION OF SODIUM AND HEAT TRANSFER WITH THE COVER GAS DO NOT PLAY ANY SIGNIFICANT ROLE IN SODIUM TEMPERATURE. THIS TWO-ZONE MIXING MODEL WILL BE A PART OF THE THERMODYNAMIC TRANSIENT CODE SSC.

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

ACCIDENT ANALYSIS • SODIUM • THERMAL ANALYSIS • ANALYTICAL MODEL • PLENUM, HEATING • REACTOR, LUMP • PLENUM • NRC-7 • JACOBS

116049

KAZIMI RS
THERMODYNAMIC LUMP SAFETY EXPERIMENTS QUARTERLY REPORT JANUARY-MARCH 1976
ORNL-70-80 • 04 PPS, 0 TABS, 10 FIGS, 01 REFS, JULY 1976

REPORTS PROGRESS MADE DURING THE PERIOD IN THE AREA OF FAST ACCIDENT THERMODYNAMIC PHENOMENA IN THE LUMP. INCLUDED IS A REVIEW OF THE SCOPE OF THE PHENOMENA, AND SOME ON HEAT GENERATING MIXING POOL AND PLUNGING FLUID STREAMING/FREEZING PHENOMENA.

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

POWELL MELTDOWN • OUT OF PILE EXPERIMENT • NRC-2 • POOL BOILING • SAFETY ANALYSIS • REACTOR, LUMP • SODIUM • EMERGENCY COOLING SYSTEM • JACOBS • ACCIDENT, CORE DISRUPTIVE

116345

AGRAWAL AM
SSC AN ADVANCED THERMODYNAMIC TRANSIENT CODE FOR LUMPS QUARTERLY PROGRESS REPORT JANUARY-MARCH 31, 1976
ORNL-70-81 • 04 PPS, 0 TABS, 10 FIGS, 01 REFS, JULY 1976

THIS IS THE THIRD QUARTERLY PROGRESS REPORT DESCRIBING THE RESEARCH PERFORMED. PROGRESS MADE DURING THIS QUARTER IS REPORTED IN THE FOLLOWING SEVEN CATEGORIES: (1) VESSEL MODELING, (2)

110305 CONTINUOUS

PRIMARY LOOP MODELING, (2) INTERMEDIATE LOOP MODELING, (3) TERTIARY LOOP MODELING, (4) INITIALIZATION CALCULATIONS, (5) CONSTITUTIVE LAWS AND CORRELATIONS, AND (7) CODE DEVELOPMENT. THE SSC WILL BE DESIGNED TO SIMULATE TWO MAJOR ACCIDENTS, (1) PIPE BREAKS UP TO AND INCLUDING A DOUBLE-ENDED JUNCTION IN THE PRIMARY HEAT TRANSPORT SYSTEM AND (2) THE LONG TERM AFTER HEAT REMOVAL IN THE ABSENCE OF FORCED PUMPING POWER.

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

COMPUTER PROGRAM • DELAY HEAT • ACCIDENT, LOSS OF CURRENT • CORRELATION • R AND D PROGRAM • REACTION, LUMP • TRANSIENT • NRC-7 • THERMAL HYDRAULIC ANALYSIS

110001

LAURENCE

SSC - AN ADVANCED THERMOHYDRAULIC TRANSIENT CODE FOR LWR'S - QUARTERLY PROGRESS REPORT APRIL 1-JUNE 30, 1976
ORNL-RM-6-30343 • 40 PPS, 11 FIGS, 12 REFS, SEPT. 1976

REPORTS PREPARED DURING THE QUARTER. THE OBJECTIVES AND CURRENT ACCOMPLISHMENTS OF THE PROGRAM AND RESEARCH, INCLUDING WORK ON IN-VESSA MODELING, PRIMARY, INTERMEDIATE AND TERTIARY LOOP MODELING, INITIALIZATION CALCULATIONS, AND A PIPE BREAK MODEL.

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

REACTION, LUMP • ACCIDENT, LOSS OF COOLANT • COMPUTER PROGRAM • ANALYTICAL MODEL • FLOW, LUMP • FLOW, TURBULENT • THERMAL HYDRAULIC ANALYSIS • HEAT TRANSFER, NATURAL CONVECTION • HEAT TRANSFER ANALYSIS • FAILURE, PIPE • NRC-7 • EMERGENCY COOLING SYSTEM

110002

LAURENCE

THERMOHYDRAULIC LWR SAFETY EXPERIMENTS QUARTERLY REPORT APRIL-JUNE 1976
ORNL-RM-6-30344 • 23 PPS, 6 FIGS, 18 REFS, SEPT. 1976

PROGRESS MADE DURING THE PERIOD ON POST ACCIDENT THERMOHYDRAULIC PHENOMENA IN THE LWR CORE IS REPORTED. INCLUDED IS WORK ON HEAT GENERATING BOILING POOL PHENOMENA, FLOWING FLUID STREAMING/HEATING PHENOMENA, AND THE DOWNWARD PROPAGATION RATE OF A MELTER HEAT GENERATING PWR.

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

REACTION, LUMP • ACCIDENT, LOSS OF COOLANT • ANALYTICAL MODEL • THERMAL HYDRAULIC ANALYSIS • HEAT TRANSFER ANALYSIS • CORE MELTDOWN • VCIU FRACTION • FLOW BLOCKAGE • NRC-7 • EMERGENCY COOLING SYSTEM

110074

LAURENCE

CROSS-SECTION METHODOLOGY IN "SINER"

LOS ALAMOS SCIENTIFIC LABORATORY, NEW MEXICO

LA-RM-6-0336-MS • 10 PPS, 2 TABS, 9 REFS, MAY 1976

THE CROSS-SECTION METHODOLOGY INCORPORATED IN THE SINER CODE IS DESCRIBED. DATA BASE FOR ALL CROSS SECTIONS IS THE ENDF/B SYSTEM WITH VARIOUS PROCESSING COMPUTER CODES TO GROUP COLLAPSE AND MODIFY THE GROUP CONSTANTS WHICH ARE USED IN SINER. EITHER INFINITELY DILUTE CROSS SECTIONS OR THE DOPAMINER FORMULAS CAN BE USED IN SINER. PRESENTLY ONLY A MICROSCOPIC TREATMENT IS CONSIDERED, BUT PRELIMINARY MACROSCOPIC ALGORITHMS HAVE BEEN INVESTIGATED.

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

COMPUTER PROGRAM • THEORETICAL INVESTIGATION • SAFETY PROGRAM • CROSS SECTION • R AND D PROGRAM

110001

LAURENCE

REACTOR SAFETY AND TECHNOLOGY QUARTERLY PROGRESS REPORT, APRIL 1-JUNE 30, 1976

LOS ALAMOS SCIENTIFIC LABORATORY, NEW MEXICO

LA-RM-6-0007-PH • 70 PPS, 17 TABS, 55 FIGS, 107 REFS, AUG. 1976

THE NIGHTMARE MODEL DESIGN OF EQUIPMENT FOR FUEL HEAT-UP EXPERIMENTS WAS COMPLETED. A TWO-DIMENSIONAL CORE TEMPERATURE MODEL WAS DEVELOPED FOR THE LANC CODE, AND A ONE-DIMENSIONAL ANALYTIC DIFFUSION MODEL WAS FORMULATED. A MAJOR MILESTONE WAS REACHED IN THE LWR SAFETY RESEARCH PROGRAM WITH THE CREATION OF THE FIRST VERSION OF THE TIAL CODE. THE VERSION CONSISTS OF ONE-DIMENSIONAL COMPONENT MODELS, SERVICE MODULES, AND CONTROL MODULES AND IS CAPABLE OF RUNNING VERY GENERAL LOOP CONFIGURATIONS. IN LWR SAFETY RESEARCH, A NEW PHASE-TRANSITION MODEL WAS ADDED TO SINER-1. THE NEW CONDUCTION-LIMITED MODEL USES A MORE IMPLICIT NUMERICAL SOLUTION AND PROVIDES BOTH A MORE PHYSICAL AND A MORE STABLE METHOD THAN THE PREVIOUS EXPLICIT SOLUTION USING THE NIGHTMARE MODEL.

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

ACCIDENT ANALYSIS • COMPUTER PROGRAM • CONTAINMENT ANALYSIS • FISSION PRODUCT RELEASE • KRYPTON • ANALYTICAL MODEL • COOLANT CHEMISTRY • HELIUM • REACTOR, LWR • SAFETY PROGRAM • ALGORITHM • R AND D PROGRAM • REACTION

120471

RESEARCH

THE RESULTS OF A BRIEF PRELIMINARY DESIGN STUDY OF A FACILITY FOR TRANSIENT NUCLEAR TESTS, OF FAST SODIUM REACTOR POOL AND REACTORS. THE STUDY IS BASED ON THE USE OF A REACTOR ON HEATING ORIGINALLY BUILT FOR THE UTILITY REACTOR, AND THE USE OF SOME REACTION INFORMATION AND REACTOR DESIGN AND FORTIFICATION TECHNOLOGY REMAINING FROM THE PUEBLO-2 REACTOR. FACILITY PERFORMANCE IN CONNECTION WITH A REFERENCE 127-PW. PER EXPERIMENT IS DISCUSSED.

REACTOR. LIFER • REACTOR TRANSIENT • REACTOR. SPEEDER • UNIT-1 (GCH) • PERFORMANCE • FULL ELEMENTS • MC-7 •
 P-1-100

125000

LAYER 10

IN THE FISSION PRODUCT TRANSPORT PART OF THE NRC SAFETY PROGRAM, THE LANC-1 CODE WAS DEVELOPED. RESULTS WERE FORMULATED FOR THE MORE COMPLEX CODES LANC-2 AND LANC-3. AND THE LANC CODE WAS DESIGNED TO PREDICT FISSION PRODUCT LEAKAGE FROM THE CONTAINMENT BUILDING. SEVERAL IMPORTANT ADVANCES WERE ACHIEVED IN THE LNC SAFETY PROGRAM. AN ANALYSIS OF NRC TEST NUMBER NO. 2 WAS PERFORMED USING THE TRAC CODE, AND CALCULATED RESULTS WERE IN EXCELLENT AGREEMENT WITH EXPERIMENTAL VALUES. IN LANC SAFETY RESEARCH, TRANSITION FROM SINGLE-LINE AND SINGLE-SUBSEQUENT STUDIES WERE PERFORMED WITH THE LANC-1 CODE.

ACCIDENT ANALYSIS • COMPUTER PROGRAM • CONTAINMENT ANALYSIS • FISSION PRODUCT RELEASE • KRYPTON • ANALYTICAL MODEL • COOLANT CHEMISTRY • HELIUM • REACTOR, LWR • SAFETY PROGRAM • CALCOMP • R AND R PROGRAM • REACTION, NEUTRON • REACTOR, LWR • SEISMIC DESIGN • SUPPLEMENTS, NUCLEAR MATERIAL • NRC-6 • NRC-7 • NRC-8 • THERMAL HYDRAULIC ANALYSIS • JACOBS • ACCIDENT, CORE OVERHEATING

LIST

FILE 66

THIS PROGRESS REPORT CONTAINS SHORT (1 TO 4 PAGE) SUMMARIES OF THE ACCOMPLISHMENTS DURING THE MONTH ON THE FOLLOWING SUB-PROJECTS: HEAVY SECTION STEEL TECHNOLOGY PROGRAM, FISSION PRODUCT GSTA AND GAMMA ENERGY RELEASE, LOCA RELEASE FROM LOW PULL, PULSED AND TESTS, NUCLEAR SAFETY INFORMATION CENTER, TWO DIMENSIONAL HEAT TRANSFER-SEPARATE EFFECTS, ZIRCALOY PWR CLADDING COLLAPSE STUDIES, ZIRCONIUM METAL-WATER OXIDATION KINETICS, AIRBORNE RELEASE AND TRANSPORT FROM LOW PULL, NTR SAFETY ANALYSIS AND RESEARCH, AND DESIGN CRITERIA FOR PIPING AND NUCLES.

ACCIDENT ANALYSIS • ASSESS • DESIGN CRITERIA • FISSION PRODUCT RELEASE • FAILURE MODE ELIMINATE • FUEL
TRANSFER • ACCIDENT, LOSS OF COOLANT • HEAT EXCHANGER • ORIENTATION • ZIRCALOY • FISSION PRODUCT
TRANSPORT • CROSS GATA • CROSS GAMMA • REACTOR, LOW • FAILURE, CLADDING • ZIRCALOY • SAFETY PROGRAM •
BLOODING • MUSIC • ENERGY LEVEL • ON AND OFF PROGRAM • REACTOR, HIGH • REACTOR, LUMPEN • NOSE • PEOPLE AND PIPE
FITTINGS

114539

CONTAINS 80 • ADAPTS 81 • DRESS 75 • PANTS 70

THE DEVELOPMENT OF A CAPACITOR DISCHARGE VAPORIZATION TECHNIQUE FOR PUTTING LIQUID FUEL INTO THE ATMOSPHERE STATES THAT TYPEF THREE PROJECTS FOR AN HCBR WAS CONTINUED. THE RESULTS AND ANALYSES OF THESE TESTS, AS REPORTED, ACTIVITIES INVOLVING THE REACTIVATION OF, AND MODIFICATION TO, THE NUCLEAR SAFETY PILOT PLANT ARE CONTINUING. TITLE II DESIGN OF THE FUEL AEROSOL SIMULATION TEST CONTINUED. MAJOR EFFORTS WERE CONCENTRATED ON COMPLETING THE DESIGN OF MAJOR COMPONENTS AND LONG-TERM PROCUREMENT ITEMS. ANALYTICAL REQUIREMENTS FOR THE AIR PNEUMATIC ARE IDENTIFIED.

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

FREE 66

NAME: William, Anthony, Kristina

DATE: 11/04/1979
TIME: 10:00 AM

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

[illegible]

DATE: 10/14/14

1. NAME OF THE PARTY _____

THE BUREAU OF THE ARMY AND THE ARMY OF THE UNITED STATES OF AMERICA

1970

DISCUSSES THE CURRENT STATUS OF LIGNIN-CELLULOSE FAST-EXCHANGE REACTION (FER) DEVELOPMENT AND USE OF THE PRINCIPAL SUPPORT ISSUES. A MATHEMATICAL COMPARATIVE ANALYSIS (MCA) OF ECONOMIC INFORMATION ON AVAILABLE LIGNINS RELATIVE TO THE DEVELOPMENT OF THE SUGAR REACTOR AS A SAFE SOURCE OF NUCLEAR FUEL IS PRESENTED FOR THE PERIOD 1973 THROUGH 1974. THE DISCUSSANTY CONSIDERS OF APPROPRIATELY TOOK ABSTRACTS COVERING EARLY RESEARCH AND DEVELOPMENT AND OPERATIONAL EXPERIENCES LEADING UP TO THE PRESENT DESIGN PRACTICES THAT ARE NECESSARY FOR THE LICENSING OF SUGAR REACTORS. KEY-WORDS: ALCOHOL AND PHENOL; D-TYPE LIGNINS ARE INCLUDED FOR COMPLETENESS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA. 22101
(813) 345 1516, 22/AF)

REACTOR, LEAD - PREVIEW - POLYMERIZATION - SIZING - REACTION, ORDER - REACTION, FAST - SODIUM - N² O
PROGRAM - QUALITY ASSURANCE - OPERATING EXPERIENCE - DESIGN STUDY - ACCIDENT, HYDROGENIC - MERCURY

SEE ME

ADJUT 1976 MONTHLY REPORTS FOR OFFICE OF BUREAU REGULATORY AND SAFETY PROGRAMS, AS CAN BE SEEN FROM THE FOLLOWING:

301 RICE BUILDING, COLUMBIA, TENNESSEE

電話/傳真/Fax: 31 435 601 1839

PRESENTS HIGHLIGHTS OF THE FOLLOWING PROGRAMS AND ACTIVITIES: PLANT SECTION STRA TECHNOLOGY; FISSION PRODUCT LCA AND GAMMA ENERGY RELEASE; LCA RELEASE FROM LOI FULL; MULTIMODAL TESTS; NUCLEAR SAFETY INFORMATION CENTER; PWR LOW VOLTAGE HEAT TRANSFER-DEGRADATION EFFECTS; STEAMER FUEL CLOGGING CORROSION STUDIES; ZIRCONIUM METAL-WATER OXIDATION KINETICS; ALUMINA RELEASE AND TRANSPORT FROM LWRs; FULL-SCALE SAFETY ANALYSES AND NUCLEAR; DESIGN CRITERIA FOR PIPING AND PRESSURE; AND THE CRACKING FACTORS FOR EVALUATION OF FUEL ELEMENTS.

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

AMERSON + DILUTION CHITINIA + CLAMPING + FISSURE PRODUCT RELEASE + DENTAL AFTER REACTION + PVC-2 + UNSUB
CALCULATION, INTERFERENCE + LUNA + PVC-3 + NOZZLE + PVC-3 + RAIL + H. AND J. HANCOCK + REACTION, LAMPSON + PVC-5 +
A-ALLOY, PVC + PVC-7 + H-2 + PILES AND PIPE FITTINGS

DEC 66

DEPT. March 1976. Multiple highlights on OFFICE OF NUCLEAR REGULATORY COMMISSION documents at JAR. Please notify JAR.

LABORATORY

DATE SUBMITTED: 02-21-2018 BY: [REDACTED]

IN 5 PHASES: (1) REPORT DATES: SHORT (1 TO 4 PAGE) SUMMARIES OF THE ACCOMPLISHMENTS DURING THE MONTH IN THE FOLLOWING CORE PROJECTS: PLANT SECTION STEEL THERMODYNAMIC ANALYSIS, FISSION PRODUCT DATA AND CORRELATION RELEASE, LUNA RELEASE FROM LUN FUEL, PLUTONIUM BURN TESTS, NUCLEAR SAFETY INFORMATION, LUNF - NON-ILLUMINUM HEAT TRANSFER-JAPANESE EFFECTS, CIRCULAT FUEL CLADDING COLLAPSE STUDIES, ZINCUMUM AL-ALLOY-DEUTERON IONIZATION KINETICS, ANCHOR RELEASE AND THERMODYNAMIC FROM LUN FUEL, PWR SAFETY ANALYSIS AND RESEARCH, AND DESIGN CRITERIA FOR PWR AND BWRs.

120330 CONTINUING

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

REACTOR, LWR • MIST • ENERGY LEVEL • FISSION PRODUCT RELEASE • WOODS DATA • WOODS DATA • FAILURE, FUEL
ELEMENT • MELT • MELTDOWN • HEAT TRANSFER • ZIRCALOY • FAILURE, CLADDING • METAL WATER REACTION • ZIRCONIUM •
OXIDATION • REACTOR, LWR • AEROSOL • FISSION PRODUCT TRANSPORT • REACTION, NEUT • ACCIDENT ANALYSIS • DESIGN
CRITERIA • PIPES AND PIPE FITTINGS • OR AND B PROGRAM • SAFETY PROGRAM • RUC-2 • RUC-3 • RUC-5 • RUC-7 • RUC-

120330

PGE 66

NOVEMBER 1976 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROVIDED AT OAK RIDGE NATIONAL
LABORATORY

OAK RIDGE NATIONAL LABORATORY, TENNESSEE
ORNL/NUREG/TH-68 P. 23 PPS, NOVEMBER 1976

THIS PROGRESS REPORT CONTAINS SHORT (1 TO 4 PAGES) SUMMARIES OF THE ACCOMPLISHMENTS DURING THE
MONTH ON THE FOLLOWING ONSITE PROJECTS: HEAVY SECTION STEEL TECHNOLOGY PROGRAM, FISSION PRODUCT
DATA AND GASES ENERGY RELEASE, LEAK RELEASE FROM LWR FULL, MULTIPLE ASSET TESTS, NUCLEAR SAFETY
INFORMATION CENTER, FOR LWR FULL HEAT TRANSFER-SEPARATE EFFECTS, LWR FULL CLADDING CORROSION
STUDIES, ZIRCONIUM METAL-WATER OXIDATION KINETICS, AEROSOL RELEASE AND TRANSPORT FROM LWR FULL,
WATER SAFETY ANALYSIS AND RESEARCH, AND DESIGN CRITERIA FOR PIPES AND NOZZLES.

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

REACTOR, LWR • MIST • ENERGY LEVEL • FISSION PRODUCT RELEASE • WOODS DATA • WOODS DATA • FAILURE, FUEL
ELEMENT • MELT • MELTDOWN • HEAT TRANSFER • ZIRCALOY • FAILURE, CLADDING • METAL WATER REACTION • ZIRCONIUM •
OXIDATION • REACTOR, LWR • AEROSOL • FISSION PRODUCT TRANSPORT • REACTION, NEUT • ACCIDENT ANALYSIS • DESIGN
CRITERIA • PIPES AND PIPE FITTINGS • OR AND B PROGRAM • SAFETY PROGRAM • RUC-2 • RUC-3 • RUC-5 • RUC-7 • RUC-

117005

DISCUSSION ON • RESUME/TC ON

LWR SAFETY, I. - REVIEW OF CURRENT ISSUES AND BIBLIOGRAPHY OF LITERATURE 1963-1969

OAK RIDGE NATIONAL LABORATORY, TERN. NUCLEAR SAFETY INFORMATION CENTER
ORNL/NUREG/NSIC-123 VOL. 11 P. 510 PPS, 2 TABS, 37 REFS, SEPT. 1976

DISCUSSES THE CURRENT STATUS OF LIQUID-METAL FAST BREEDER DEVELOPMENT AND ONE OF THE PRINCIPAL
SAFETY ISSUES, A HYPOTHETICAL CORE-DISRUPTIVE ACCIDENT. BIBLIOGRAPHIC INFORMATION ON DOMESTIC
LWRs RELATIVE TO THE DEVELOPMENT OF THE BREEDER PLANT AS A SAFE SOURCE OF NUCLEAR POWER IS
PRESENTED FOR THE PERIOD 1960 THROUGH 1969. THE BIBLIOGRAPHY CUMULATES UP 1503 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. 22101
(013/COPY)

BIBLIOGRAPHY • FISSION PRODUCT RELEASE • HEAT TRANSFER • REACTION, LWR • HEAT-UP TRANSIENT • DESIGN •
TRANSPORTATION AND MODELING • ELECTRIC POWER • OPERATING EXPERIENCE • REACTOR, LWR • RUC-7 • INSTRUMENTS,
RISC • ACCIDENT, CORE DISRUPTIVE • JACOBS

120645

PONTANA RM • RESS 15

LWR AEROSOL RELEASE AND TRANSPORT PROGRAM QUARTERLY PROGRESS REPORT FOR APRIL-JUNE 1976

OAK RIDGE NATIONAL LABORATORY, TENNESSEE
ORNL/NUREG/TH-59 P. 49 PPS, TABS, FIGS, 10 REFS, DEC. 1976

THE PROGRAM OF CAPACITOR DISCHARGE VAPORIZATION (CDV) EXPERIMENTS CONTINUES. ALL TESTS PROVIDED
MEASURABLE PERSISTENT AEROSOL VIDEOS AND PROVIDED ADDITIONAL INFORMATION FOR USE IN FUTURE
TRANSIENT RELEASE EXPERIMENTS. DESIGN AND INSTALLATION OF CDV EQUIPMENT AT JNL CONTINUES. A
PREHEAT-CYCLE EXPERIMENTAL STUDY HAS CONFIRMED THE CALCULATION METHODS USED TO ESTIMATE THE
FRACTION OF PELLETS MELTED UNDER UNIFORM CONDITIONS. EXPERIMENTS USING THE EXISTING METALLIC
URANIUM OXIDATION AEROSOL GENERATION TECHNIQUE IN THE LWR-11 VESSEL ARE REPORTED. RESULTS OF
MEASUREMENTS OF PARTICLE-SIZE DISTRIBUTION BY VARIOUS TECHNIQUES ARE DISCUSSED. IN THE LWR-11
VESSEL WORK, PROGRESS IN THE ACTIVATION OF THE NUCLEAR SAFETY PILOT PLANT INPP CONTINUES
BOTH IN DESIGN AND CONSTRUCTION.

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

AEROSOL • FISSION PRODUCT RELEASE • PLUTONIUM • FISSION PRODUCT TRANSPORT • ACCIDENT, HYPOTHETICAL • REACTION,
LWR • RADIOISOTOPE TRANSPORT • RUC-7 • NSPP

120130

PGE 66

NOVEMBER 1976 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROVIDED AT OAK RIDGE NATIONAL
LABORATORY

OAK RIDGE NATIONAL LABORATORY, TENNESSEE
ORNL/NUREG/TH-79 P. 25 PPS, DEC. 1976

Availability - National Technical Information Service, U. S. Department of Commerce, Springfield, VA. 22161

11/20/2016

200001A L.A. 300 22-100001A 100001A

During this period the output of the series of experiments with sodium was initiated. These experiments are conducted with peak energy variations ranging from 1500 to 2700 eV. In the first four experiments, the chlorine was not all used, in run -2, the chlorine flow rate was increased to obtain a more uniform and complete conversion of the chlorine. Run -3, the rate of experiment actually was used in a previous experiment; hence, the peak was modified somewhat as compared with a previous run.

[illegible]

110725

DAVID A. LADD, ALUMINUM WORKS, A.D.

PROGRESS IN THE ALPHA JOURNAL PROJECT IS DEVELOPED. ANALYSIS REVIEWED INCLUDE THE SAFETY ANALYSIS REPORT; LOW MULLION DESIGN; CANNON DEVELOPMENT; PLUMMER DESIGN; PULP ELEMENT DESIGN; AIRWAY FUEL MATERIAL STUDIES; SECONDARY FUEL MATERIAL STUDIES; JOURNAL CORE FUEL ELEMENT; AND THE BIOGRAPHIC SYSTEM.

REACTION, PAIR + MULTIPLICATION + REACTION, PULSED + SAFETY PROGRAM + N AND O PROGRAM + JMC-7 + JAC-703

120170

SANDIA LABOR. - ALBUQUERQUE, N. M. MEXICO

NUMBER 1775. JONAS LANGEHAUGERES CONDUCTED IN EXPERIMENTAL FAST-NEUTRON SAFETY RESEARCH POSITION FURNISH BY THE ACQUISITION OF REACTION SAFETY RESEARCH. A PORTION OF THIS RESEARCH INVOLVED A FEASIBILITY STUDY FOR PERFORMANCE IMPROVEMENT OF THE ANNUAL CYCLE PULSE REACTOR (ACP) UPGRADE. THE OBJECT OF THE ACP UPGRADE IS TO ARRIVE AT A REACTOR CONFIGURATION WHICH WILL PROVIDE AN INCREASED PULSE NEUTRON FLUENCE IN THE IRRADIATION CAVITY WITHOUT AN INCREASED DEGRADATION OF THE PULSE NEUTRONICS. THE UPGRADED REACTOR WILL ALSO HAVE AN INCREASED STEADY-STATE NEUTRON FLUX. THE APPROACH TO THE UPGRADE MODIFICATION INVOLVED A THERMAL-ION CURRENT, THE INNER NEUTRON SHIELDING OF THE IRRADIATION CAVITY, CONSISTS OF A MINOR-SCALE CAPACITY PULS ORIGIN WILL SUSTAIN A LARGE PULSED NEUTRON FLUENCE. THE OUTER NEUTRON SHIELDING OF A MINOR-SCALE NEUTRON FLUENCE PULS SIMILAR TO THE PRESENT ACP PULS. THIS REACTOR MODIFICATION WILL MAKE USE OF THE HANDBOOK OF THE CRITICAL REACTOR STRUCTURE AND CAN BE ACCOMPLISHED IN A RELATIVELY SHORT TIME.

REACTION, RESISTANCE, REACTIVITY, MECHANISM, COMPLICATION, PLUS DISTRIBUTION, UNUSUAL MECHANISM, DISCUSSION, REACTION MECHANISM, N-6-7

119157

ANNUAL REPORT ON LAF-1000 DISRUPTIVE ACCIDENTS ANNUAL REPORT, JUNE 1, 1973-JUNE 30, 1976

UNIVERSITY OF VIRGINIA, CHARLOTTESVILLE

THE WORK REPORTS HEREIN IS PART OF THE AEROSOL RELEASE AND TRANSPORT PROGRAM FOR LOW-BOY SAFETY ASSIGNED TO THE REACTION SAFETY RESEARCH DIVISION OF THE U.S. WILDERNESS MANAGEMENT COMMISSION. SINCE THE AT VARIOUS STAGES OF INVESTIGATION DURING THIS REPORTIVE PERIOD, A STUDY OF AEROSOLIZATION AND TRANSPORT DURING PUL EXPANSION AND A STUDY OF THE DYNAMICS OF PUL EXPANSION

119107 CONTINUED

AND THE SEVERAL PDA WERE COMPLETED. STUDIES ARE UNDERWAY ON CONDENSATION ON ADHESIVE
STRUCTURES AND ON GENERATION OF AEROSOLS FROM CONDENSATION. STUDIES ARE INITIATED ON SMALL-
PARTICLE GENERATION FROM HYPERBOLIC FRAGMENTATION ON PARTICLE/IMPACTS AND ON PARTICLE-
SURFACE INTERACTIONS.

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

ACCIDENT ANALYSIS • REACTOR, BREIDEN • REACTOR, FAST • FUEL MELTDOWN • JETLIN • AEROSOL PRODUCTION •
CONDENSATION • EVAPORATION • PARTICLE SIZE DISTRIBUTION • DUST • HEAT TRANSFER, TWO PHASE • REACTOR, LWR •
• FUEL CORE • INTERACTION • HCL-7 • ACCIDENT, CORE DISRUPTION • JALUS

THE CONSTRUCTION OF THE APPROPRIATE PPM MEASUREMENTS FROM OBSERVATION DATA WAS COMPLETED USING A MODIFIED EFFICIENT NEAREST-NEIGHBOR TECHNIQUE. THIS COMPLETE AND FUNCTIONAL CALCULATION USING EFFICIENT CUSTOM BUILT SOFTWARE WAS USED TO ASSESS THE RELIABILITY OF THE APPROPRIATE CALCULATIONS AND CONTRIBUTING TO STUDY VARIATION. DESIGN PROPOSED DATA-OBSERVABLE COMPARISONS, OR CORRELATIONS, WERE USED TO ASSESS THAT THE MODEL RESULTS OF INTEREST WERE IN LINE WITH THE DATA. TO CORROBORATE THE FINDINGS OF THE MODEL, RECEIVED AND OBSERVED DATA WERE COMPARED. THE ANALYSIS WAS USED TO ASSESS THE EFFECTS OF ALLIGANT INCREASE IN AIR-WATER RATIO INTO THE FURTHER CORRELATION OF THE MODEL AND OBSERVED CALCULATIONS. THIS WAS MADE IN THE PPM CORRELATION AND PPM ANALYSIS. EACH DATA AND PPM VALUES SHOWING AN APPROPRIATE MATCHING WAS ALSO STUDIED.

[illegible][illegible]

ACQUANT ANALYSIS • LINGUISTICS • FISSION PHENOMENON • REACTION • VIBRATION • JOURNALISM • ANALYTICAL MODEL •
FISSION PHENOMENON • TRANSPORT • COMPLEXITY • POSTMODERNISM • INFORMATION • COMPLEXITY • COMPLEXITY • STRUCTURAL
INTEGRITY • DEPTH EVALUATION • DATA LOGGING SYSTEM • A AND B • COMPLEXITY • REACTION • INFORMATION • MODEL •

A THERMODYNAMIC EQUILIBRIUM APPROACH HAS BEEN USED TO EXAMINE THE LIMITS OF IMPURITY COMPOSITION OF METAL LIQUID IN A HIGH-TEMPERATURE GAS-SOLVED METAL SYSTEM. EQUILIBRIUM CONCENTRATIONS WOULD OCCUR, WHICH ASSUME THERMODYNAMIC EQUILIBRIUM AND ARE THE SAME FOR DIFFERENT TREATMENTS. MORE DETAILED LEVELS OF CALCULATED LIQUANT COMPOSITIONS ALLOWING VARIOUS AMOUNTS OF STEAM, HYDROGEN, AND STEAM (NITROGEN) IN THE PRESENCE OF OTHER IMPURITIES. IN GENERAL, THE RESULTS INDICATE THAT THE MOST HEAVY METALS, H , C , AND O , DO NOT DOMINATE THE COMPOSITION OF THE LIQUANT. ANALYTICAL TREATMENTS BASED ON THIS CONCLUSION ARE USED TO GIVE OTHER ANALYTICAL DATA AND CONCLUSIONS. THE DATA HAVE PROVEN USEFUL FOR DEFINING LIMITS OF LIQUANT IMPURITY ANALYSIS AND FOR PREVENTING POSSIBLY IMPORTANT ANALYTICAL ERRORS IN THE METAL LIQUANT STATE.

[illegible]

NEVERHOOD OF AN ATOM ACCIDENT SIMULATION PROGRAM IS IN PREPARATION FOR THE PRODUCTION OF THE
USING AN ATOM PLANT THROUGHOUT RESPONSE TO VARIOUS INITIATION EVENTS. THE STATUS OF THE USUAL
COMPLEX MODELING THAT (COMPOSITE) WITH ANALYSIS PROGRAMS AS OF JUNE 13, 1979, IS GIVEN THE
PRODUCTION STRUCTURE AND CAPABILITIES OF THE WORK CASE AND DISCLOSED INFORMATION
RESEARCHING AND WORKING WITH THE COMPOSITE THAT HAVE BEEN AVAILABLE TO THE USUAL MODEL
VALUATION AND EVALUATION USING ANALYSIS ANALYSIS AND ALSO DISCLOSED

ACCIDENT ANALYSIS • LITERATURE SURVEY • ANALYTICAL SKILL • FUNCTIONAL DECOMPOSITION • DATA CHAINING SYSTEM •
 STEAM GENERATION • NUCLEAR REACTOR • MILLING • MODEL • TRANSDUCER • MFC • POLYESTER

DESIGN CRITERIA + MATERIALS + CONSTRUCTION + LABOR COSTING + PLANT SCHEDULE + DESIGN TEAM + VENDOR

117090
AMERICAN CA • DRIVE IN • NUMBER 86
A PROPOSAL FOR A BUSINESS FACILITY FOR RESEARCH AND DEVELOPMENT OF SCIENCE
AND ARTS SUBJECTS IN CALIFORNIA. NEW ORLEANS
CALIFORNIA-6-10-68 P. 17 P. 12 PAGES. 13 SEP 5. JULY 1970

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

VIBRATION • NOISE • FLUXION • PLASTIC DESIGN • WELDING PLANT • DESIGN • TESTING • JOINTS

110041
LATVIAN RD
HEALTH SAFETY AND TECHNOLOGY QUARTERLY ADDRESS REPORT, APRIL 1-JUNE 10, 1970
LOS ALAMOS SCIENTIFIC LABORATORY, NEW MEXICO
LA-MRSG-0007-PW P. 70 PWS. 17 TABS. 23 FIGS. 107 PAGES, APR. 1970

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

ACCIDENT ANALYSIS • COMPUTER PROGRAM • CONFINEMENT ANALYSIS • FISSION PRODUCT RELEASE • KINETICS • MATERIALS
MODEL • CORROSANT CHEMISTRY • MELT • REACTION, LOW • SAFETY PROGRAM • MANAGEMENT • HAZARD • PERFORMANCE • RADIATION,
HIGH • REACTION, LOW • SAFETY DESIGN • SAFEGUARDING • RADIATION PROTECTION • RADIATION • RADIATION • RADIATION • RADIATION
HYPERBARIC ANALYSIS • JACOBI • ACCIDENT, CORRE MINIMIZING

119092
CONT 5L P 6L JM
CORE HEATER AND PISTON PLUGS RELEASE FROM AN MTG CORE IN AN LCPA ACCIDENT
LOS ALAMOS SCIENTIFIC LABORATORY, NEW MEXICO
LA-4496-6-69-25 0. 05 PPS. 10 FIGS. 12 REFS. 54BT. 1970

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

COMPUTER PROGRAM • FISSION PRODUCT RELEASE • GRAPHITE • ACQUENT, LADS UP PLUG • ANALYTICAL MODEL • CORE
MELTDOWN • REACTOR, NUC • MIX-E

120448
LAWRENCE
REACTOR SAFETY AND TECHNOLOGY QUARTERLY PROGRESS REPORT, JULY 1-DECEMBER 30, 1969

AVAILABILITY - National Technical Information Service, U. S. Government Printing Office, Springfield, VA. 22161

110-20
MEMO TO
FROM: SAC, NEW YORK (100-8769)
SUBJECT: [REDACTED]
DATE: 1/10/70

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115301
 PEE 66
 JUNE 1970 MONTHLY MISSIONARY PUBL OFFICE OF AMERICAN REGULATION OF TRADE AND COMMERCE
 AMERICAN
 ON AICE NATIONAL LAWYERS. F. 1970588
 1970/1970/1970-66 P. 27 P. 27, MAY 1970

AVAILABILITY - POLITICAL TECHNOLOGY INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, BUREAU OF ECONOMIC ANALYSIS, WASHINGTON, D. C. 20540

117510
 MESSAGE DA P LEEFLEAD IN
 SUBJECT: A DUTY OF THE: IN PLEASANT FOR THE APPROXIMATE SIMILARITY OF THE NEW TEMPERATURE AND CLOUD NEARLY
 DECREASED-STEADY-STATE-STATE
 AND THE NEW MATHEMATICAL ANALYSIS OF THE TEMPERATURE
 FROM THE NEW MATHEMATICAL ANALYSIS OF THE TEMPERATURE
 FROM THE NEW MATHEMATICAL ANALYSIS OF THE TEMPERATURE

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

COMPUTER PROGRAMS • ANALYSIS OF DATA • HEAT TRANSFER ANALYSIS • CORRELATION • FLOW STABILITY • P.T. ST. DESIGN
(MIXED) • SYSTEMS CORRELATION • CONDENSATION HEAT • FRICTION FACTORS • INTERNAL TRANSFER • MORE...

THIS DOCUMENT REPORT CONTAINS INFORMATION OF A CONFIDENTIAL NATURE AND IS NOT TO BE RELEASED TO THE PUBLIC OR TO ANY OTHER PERSON OR ORGANIZATION WITHOUT THE WRITTEN AUTHORIZATION OF THE NATIONAL AERONAUTICS AND SPACE ADMINISTRATION. IT IS THE POLICY OF THE NATIONAL AERONAUTICS AND SPACE ADMINISTRATION TO PROTECT THE CONFIDENTIALITY OF INFORMATION CONTAINED HEREIN AND TO PREVENT THE UNLAWFUL DISCLOSURE OF SUCH INFORMATION. IT IS THE POLICY OF THE NATIONAL AERONAUTICS AND SPACE ADMINISTRATION TO PROTECT THE CONFIDENTIALITY OF INFORMATION CONTAINED HEREIN AND TO PREVENT THE UNLAWFUL DISCLOSURE OF SUCH INFORMATION.

[illegible]

WORK THIS QUARTER WAS CONCENTRATED ON FURTHER DEVELOPMENT OF THE TOP CODE AND ADAPTING IT TO SIMULATIONS OF SPECIFIC POST 37. WORKER REACTION PROBABILITIES. A COMPUTER FORC CONTAINS THE MOST RECENT RELEASE OF THE TOP CODE AND RELATED DOCUMENTATION AND INFORMATION FROM DISSEMINATION NATIONAL LABORATORY DURING THE QUARTER. WORK HAS BEEN INITIATED TO MAKE THIS VERSION OPERATIONAL ON THE JCS JOE SYSTEM. PROJECT PERSONNEL ATTENDED A WORKSHOP REVIEW OF NEW RELEASED DISSEMINATION AND ANALYSIS TASKS IN CHARLOTTE, NC.

ACCIDENT ANALYSIS • CAUSE/FIN PROBABILITIES • REACTION TIME/SEQUENT • ANALYTICAL PHASE • REACT TIME/REACT ANALYSIS • P.T. 27.
UNSAFE (P.T. 28) • STEAM GENERATION • REACTION, REACT • REACT • REACT

PRESENTS HIGHLIGHTS OF THE FOLLOWING PROGRAMS AND ACTIVITIES: REACTOR SECTION STEEL TECHNOLOGY; FISSILE PRODUCT WASTE AND WASTE ENERGY RELEASE; COCA RELEASE FROM CAND FUELS; MULTIPHASE REACTOR TESTS; NUCLEAR SAFETY INFORMATION CENTER; PWR BLOW DOWN HEAT TRANSFER-PRIMARY EFFECTS; (BLOWDOWN) FUEL CLADDING COLLAPSE STUDIES; ZIRCONIUM METAL-OXIDE OXIDATION KINETICS; NUCLEAR RELEASE AND TRANSPORT FROM LIQUID FUEL; NUCLEAR SAFETY ANALYSIS AND MANAGEMENT; CATASTROPHIC PIPING AND RELEASES; AND OVER-TEMPERATURE FAILURES FOR INNOVATION OF DEVELOPMENT.

DESIGN • DESIGN CRITERIA • LOADING • FISSURE PRODUCT RELEASE • RAIL HALL ANALYSIS • NRC-2 • WSL
CALCULATION, INTERNAL • LWA • NRC-3 • REZALE • NRC-3 • RAIL - 3 AND W. PROGRAM • REACTION, LEAK • NRC-6 •
AGENCY, NRC • NRC-7 • WSL • PIPES AND PIPE FITTINGS

THIS PROGRESS REPORT CONTAINS ABOUT 11 TWO-PAGE SUMMARIES OF THE ACCOMPLISHMENTS DURING THE MONTH ON THE FOLLOWING CARL PROJECTS: HEAVY ACTION STEEL TECHNOLOGY PROGRAM, FIBER OPTIC DATA AND GAMMA RAY DET RELEASE, LOCAL RELEASE FROM LOW PELL, MULTIPLE AND/ OR FEEDS, NUCLEAR SAFETY INFORMATION CENTER, FOR OCEANIC HEAT TRANSFER-SEPARATE EFFECTS, CLIMATE OF FUEL GASOLINE CATALYST STUDIES, ZINCUMIUM METAL-STEEL OXIDATION KINETICS, METAL RELEASE AND TRANSPORT FROM LOW PELL, NUCLEAR SAFETY ANALYSIS AND RESEARCH, AND DESIGN CRITERIA FOR PIPING AND TUBES.

REACTION, LAR + WEST + ENERGY LEVEL + FISSION PRODUCT RELEASE + WOUND DATA + WOUND WOUND + FAILURE, FULL
ELEMENT + NSIC + GROUND + HEAT TRANSFER + ZIRCALOY + FAILURE, CLADDING + HEAT WOUND REACTION + DISPOSITION
ORIENTATION + REACTION, LAR + AEROSOL + FISSION PRODUCT TRANSPORT + REACTION, NUC + ACCIDENT ANALYSIS + WOUND

 

A. B. 300

4. Содержание
 1. Введение
 2. Основы теории
 3. Практические задания
 4. Заключение
 5. Список литературы

[illegible]

DECLASSIFICATION - WILLIAM L. JAMES, JR. INFORMATION SECURITY, U. S. DEPARTMENT OF COMMERCE, WASHINGTON, DC. 20511

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1944 年 12 月 25 日 星期日 12 月 25 日

[illegible]

When this computerized data storage system is installed, the processing of records will be simplified and a reduction of the need for space between the two computer systems will be achieved. The computer will now also be used to analyze a new format of data communication system when the time between the accident initiation to the time of the communication is initiated at a point within the two minute window after computer time results in a change of status from the current system at a data terminal to an automatic response delay with a 1000-2000 time duration.

APPROXIMATELY - 200,000 INDIANS, INCLUDING 100,000 MEN, WOMEN AND CHILDREN, WERE KILLED BY THE SPANISH CONQUERORS.

[illegible]

REF 66

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5. NRC-12 TRANSPORTATION SAFETY RESEARCH

100100
 MCMASTER D • DRUGS RD • STUMP JR
 FINAL REPORT ON SPECIAL TESTS OF PLUTONIUM URIDE SHIPPING CONTAINERS TO FAA FLIGHT RECONSTRUCTION SURVIVABILITY STANDARDS
 SANDIA LABS., ALBUQUERQUE, NEW MEXICO
 SAND-75-0000 • NUREG-SAND 75-0000 • 172 PPS, 170 FIGS, SEPT. 1975

THE PURPOSE OF THE TEST SERIES WAS TO EVALUATE THE LEAK TIGHTNESS OF TYPICAL PLUTONIUM URIDE SHIPPING CONTAINERS WHEN SUBJECTED TO THE SAME TEST MECHANISMS TO WHICH THE FAA SUBJECTS THEIR FLIGHT RECORDERS. THE CONTAINERS CHOSEN FOR EVALUATION WERE THE LLU-1 PACKAGE MADE BY UAC AND DESIGNED BY UACONT. THE ROCKY FLATS MODEL 1510 CONT-SPEC ONE CONTAINER, AND THE NUCLEAR CONTAINERS, INC., MODIFIED PL-10 CONTAINER MODIFIED TO 55-GALLON URIDE SIZE.

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

SHIPPING CONTAINER • TEST, DESTRUCTIVE • TRANSPORTATION AND HANDLING • PLUTONIUM URIDE • SAFETY ANALYSIS • SPECIAL NUCLEAR MATERIAL

110070
 SCHWAB CL • BIERKOWSKI D • NUTHE RE
 REFERENCE CRITICAL EXPERIMENTS PROGRESS REPORT FOR PERIOD APRIL 1-JUNE 30, 1976
 ATOMICS INTERNATIONAL, GLENDALE, COLORADO
 NPP-NUREG-2535 • 27 PPS, 7 TABS, 4 FIGS, AUG. 1976

THE GREATER PART OF THE EFFORT DURING THE QUARTER WAS THE SET-UP AND PERFORMANCE OF TWENTY-FOUR CRITICAL EXPERIMENTS ON FINELY REFLECTED TANKS CONTAINING URANIL NITRATE SOLUTION AT AN ENRICHMENT OF 93% AND A RANGE OF CONCENTRATIONS. A DETAILED SECURITY DESCRIPTION OF THE APPARATUS HAS BEEN WRITTEN AND USED TO GENERATE DATA FOR COMPARISON WITH THE EXPERIMENTAL RESULTS. PREPARATION FOR THE REFLECTED SINGLE AND MULTIPLE CYLINDER EXPERIMENTS CONTINUED. TWO METHODS OF PREPARING THE LEO-ENRICHED U235 AT THE DESIRED DENSITY AND MWU ARE BEING UNDERTAKEN. AND THE RESULTS OF THESE STUDIES ARE GIVEN.

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

CRITICAL ASSEMBLY • URANIL • NRC-12 • CRITICALITY EXPERIMENT • NITRATE • SOLUTION

ALPHABETIC INDEX

A COLLECTION OF KEYWORDS IS USED TO DENOTE THE MAIN SUBJECT RELATED POINTS COVERED IN EACH ARTICLE. THE FOLLOWING INDEX IS AN ALPHABETICAL LISTING OF THE KEYWORDS GIVING REFERENCES TO EACH ARTICLE WHICH HAS REFERRED TO IT.

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ONE EVALUATION OF EEC PENETRATION DATA USING TWO SCALING
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TABLE 1. STEAM-WATER HEATING PROGRAM AND SYSTEM
TABLE 2. STEAM-WATER HEATING PROGRAM AND SYSTEM
TABLE 3. SIXTH QUARTERLY PROGRESS REPORT JANUARY THROUGH
TECHNIQUES TO THE ANALYSIS OF COMPUTER CODES

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