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

J. R. Buchanan

NUCLEAR SAFETY INFORMATION CENTER

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CONTENTS

Page

FOREWORD ................................................................. v
PARTS AND METHOD OF 'INDEXING ABSTRACTS ....................... vii
ABSTRACT ................................................................... ix
INTRODUCTION ............................................................. ix
WATER REACTOR SAFETY RESEARCH ............................... x
  Basic (R1) ................................................................. x
  Systems Engineering (R2) ........................................... x
  Fuel Behavior (R3) ..................................................... xi
  Analysis Development (R4) ......................................... xi
  Metallurgy and Materials (R5) .................................... xi
  Site Safety Research (R6) ........................................... xi
ADVANCED REACTOR SAFETY RESEARCH ...................... xii
  Fast Reactors (R7) ..................................................... xii
  Gas-Cooled Reactors (R8) .......................................... xii
OTHER RESEARCH ........................................................ xiii
  Health Safety Research (RH) ........................................ xiii
  Environmental Research (RE) ..................................... xiii
  Transportation Safety Research (RT) ......................... xiii
  Safeguards Research (RS) .......................................... xiii
  Effluent and Radiation Field Source
     Terms in Operating Reactors (RR) ......................... xiv
     Criticality (RC) .................................................... xiv
ORGANIZATION OF BIBLIOGRAPHY ................................ xlv
PRICES FOR DOCUMENTS ABSTRACTED IN THIS REPORT ....... xvi
BIBLIOGRAPHY .............................................................. xvii

1. R1 — Water Reactor Safety Research, Basic .......................... 1
2. R2 — Water Reactor Safety Research, Systems Engineering .......... 4
4. R4 — Water Reactor Safety Research, Analysis Development .......... 26
5. R5 — Water Reactor Safety Research, Metallurgy and Materials ....... 31
6. R6 — Water Reactor Safety Research, Site Safety Research .......... 37
7. R7 — Advanced Reactor Safety Research,
   Fast Reactors.................................................. 38
8. R8 — Advanced Reactor Safety Research,
   Gas-Cooled Reactors........................................... 46
9. RH — Health Safety Research.................................. 49
10. RE — Environmental Research................................ 50
11. RT — Transportation Safety Research........................ 53
12. RS — Safeguards Research..................................... 54
13. RR — Effluent and Radiation Field Source
    Terms in Operating Reactors................................... 59
14. RC — Criticality.................................................. 60

KEYWORD INDEX...................................................... 61
AUTHOR INDEX.......................................................... 69
PERMUTED-TITLE INDEX............................................... 73
The Nuclear Safety Information Center (NSIC), which was established in March 1963 at Oak Ridge National Laboratory, is principally supported by the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research. Support is also provided by the Division of Reactor Development and Demonstration of the Department of Energy. NSIC is a focal point for the collection, storage, evaluation, and dissemination of safety information to aid those concerned with the analysis, design, and operation of nuclear facilities. Although the most widely known product of NSIC is the technical progress review Nuclear Safety, the Center prepares reports and bibliographies as listed on the inside covers of this document. The Center has also developed a system of key words to index the information which it catalogs. The title, author, installation, abstract, and key words for each document reviewed are recorded at the central computing facility in Oak Ridge. The references are cataloged according to the following categories:

1. General Safety Criteria
2. Siting of Nuclear Facilities
3. Transportation and Handling of Radioactive Materials
5. Heat Transfer and Thermal Hydraulics
7. Fission Product Release, Transport, and Removal
8. Sources of Energy Release under Accident Conditions
9. Nuclear Instrumentation, Control, and Safety Systems
10. Electrical Power Systems
11. Containment of Nuclear Facilities
12. Plant Safety Features — Reactor
13. Plant Safety Features — Nonreactor
14. Radionuclide Release, Disposal, Treatment, and Management (inactive September 1973)
15. Environmental Surveys, Monitoring, and Radiation Dose Measurements (inactive September 1973)
16. Meteorological Considerations
17. Operational Safety and Experience
18. Design, Construction and Licensing
19. Internal Exposure Effects on Humans Due to Radioactivity in the Environment (inactive September 1973)
22. Safeguards of Nuclear Materials

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

J. R. Buchanan, Assistant Director
Nuclear Safety Information Center
P.O. Box Y
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

Telephone 615-483-8611, Ext. 3-7253
FTS number is 850-7253
PREMATURE FAILURES WERE OBSERVED IN ONE 0.008-IN.-WALL ROD AND IN ONE 0.012-IN.-WALL ROD OF THE MARK-III EXPERIMENTAL FUEL. CLADDING IS INCOLOY-800. FAILURES WERE IN REGION OF MAXIMUM POWER IN PEAK-POWER ROD. MOST REASONABLE EXPLANATION IS LOW-CYCLE FATIGUE, ACCELERATED BY HIGHER-THAN-DESIGN CLADDING TEMPERATURES. NO SIGNS OF RAPID CORROSION, SUCH AS HAVE BEEN ASSOCIATED WITH TYPE-304 SS, WERE FOUND.


*FAILURE, FUEL ELEMENT • CLAD • FAILURE, FATIGUE • INCONEL • REACTOR, INTERNAL SUPERHEAT • THERMAL MECHANICAL EFFECT • VESR (ISR)
BIBLIOGRAPHY OF REPORTS ON RESEARCH SPONSORED BY
THE NRC OFFICE OF NUCLEAR REGULATORY RESEARCH
JULY–DECEMBER 1977

J. R. Buchanan

ABSTRACT

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

INTRODUCTION

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

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

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

WATER REACTOR SAFETY RESEARCH

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

Basic (R1, formerly NRC-1)

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

Systems Engineering (R2, formerly NRC-2)

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

Fuel Behavior (R3, formerly NRC-3)

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

**Analysis Development (RA, formerly NRC-4)**

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

**Metallurgy and Materials (R5, formerly NRC-5)**

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

**Site Safety Research (R6, formerly NRC-6)**

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

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

Fast Reactors (R7, formerly NRC-7)

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

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

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

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

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

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

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

Health Safety Research (RH, formerly NRC-9)

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

Environmental Research (RE, formerly NRC-10)

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

Transportation Safety Research (RT, formerly NRC-12)

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

Safeguards Research (RS, formerly NRC-13)

This category includes information on methods and data, as well as devices and techniques, which support the NRC functional capabilities in safeguards rule-making, licensing, and inspection. In particular, there is information on (1) the effectiveness of evaluation methods which aid
in the assessment of physical protection and material control subsystems at fixed sites and in transport; (2) analysis of safeguards information requirements; (3) physical protection equipment and its evaluation; (4) topics of broad safeguards interest relevant to future directions, including estimation of consequences of malevolent nuclear acts, safeguards vulnerability to white-collar crime, and impacts of performance-oriented regulations, among others.

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

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

Criticality (RC)

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

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

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

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

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5285 Port Royal Road
Springfield, Virginia 22161
BIBLIOGRAPHY
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THIS BIBLIOGRAPHY CONTAINS 204-ABSTRACTS OF REPORTS TO THE U.S. NUCLEAR REGULATORY COMMISSION CONCERNING OPERATIONAL EVENTS THAT OCCURRED AT ONE POWER PLANTS IN 1974. THE REPORT INCLUDES 189 ABSTRACTS THAT DESCRIBE INCIDENTS, FAILURES, AND DESIGN OR CONSTRUCTION DEFICIENCIES THAT WERE EXPERIENCED AT THE FACILITIES. THEY ARE ARRANGED ALPHABETICALLY BY REACTOR NAME AND THEN CHRONOLOGICALLY FOR EACH REACTOR. KEY-WORDS AND PUBLISHED-TITLE INDEXES ARE PROVIDED TO FACILITATE LOCATION OF THE SUBJECTS OF INTEREST. AND TABLES THAT SUMMARIZE THE INFORMATION CONTAINED IN THE BIBLIOGRAPHY ARE PROVIDED. THE INFORMATION LISTED IN THE TABLE INCLUDES INSTRUMENT FAILURES, EQUIPMENT FAILURES, SYSTEM FAILURES, CAUSES OF FAILURES, DEFICIENCIES NOTED, AND THE TIME OF OCCURRENCE T.I.E., DURING REACTOR, OPERATION, TESTING, OR CONSTRUCTION.

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

A REACTOR, MAJOR INCIDENT COMPILATION & BIBLIOGRAPHY & OPERATING EXPERIENCE & DATA COLLECTION & NRC-1 & REACTOR POWER & REACTOR POWER & JACOBS

BIBLIOGRAPHY OF REPORTS DISTRIBUTED UNDER THE NRC LIGHT-WATER REACTOR SAFETY TECHNICAL EXCHANGE Vol. III (JANUARY-JUNE 1977)

A 200-PAGE NATIONAL LEND, TOYНА.

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BIBLIOGRAPHY OF REPORTS ON RESEARCH SPONSORED BY THE NRC OFFICE OF NUCLEAR REGULATORY RESEARCH JANUARY-JUNE 1977

A 90-PAGE NATIONAL LEND, TOYНА.

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A NUCLEAR DIAGNOSTICS FOR SAFETY ASSESSMENT QUARTERLY PROGRESS REPORT FOR APRIL-JUNE 1977

A 115-PAGE NATIONAL LEND, TOYНА.

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A NUCLEAR DIAGNOSTICS FOR SAFETY ASSESSMENT QUARTERLY PROGRESS REPORT FOR JULY-SEPTEMBER 1977

A 70-PAGE NATIONAL LEND, TOYНА.
122000 CONTINUED
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
A AND B PROGRAM - SAFETY PROGRAM - NRC-2 - REACTOR. OUR - BLOWDOWN - HEAT TRANSFER - COMPUTER PROGRAM - EMERGENCY COOLING SYSTEM - OUT OF PILE EXPERIMENT

122000

EXPERIMENT DATA REPORT FOR SEMISCALE MOD-1 TESTS S-05-0 AND S-05-7 (ALTERNATE EGC INJECTION TESTS)
EDWARDS NATIONAL ENGINEERING LAB., IDAHO FALLS
REEI-122000-1, 205 PGS. 11 TABS. 356 FIGS. 5 REP., JUNE 1977

RECORDED TEST DATA ARE PRESENTED FOR TESTS S-05-0 AND S-05-7 OF THE SEMISCALE MOD-1 SIMULATED CORE THERMAL RESPONSE AND INJECTION TEST SERIES CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING A HYDROGEN FUELED-CORE ACCIDENT IN A PRESSURIZED WATER REACTOR LOOPS SYSTEM. THE SPECIFIC OBJECTIVE FOR THESE TESTS WAS TO INVESTIGATE THE EFFECTIVENESS OF LOW PRESSURE INJECTION SYSTEM (LPS) BLOWDOWN INJECTION INTO THE UPPER PLATEUM WHEN COMBINED WITH COLD LEG INJECTION. THE SEMISCALE MOD-1 SYSTEM WAS MODIFIED TO REPRESENT MORE CLOSERLY A TWO-LOOP PWR. HOWEVER, TESTS S-05-0 AND S-05-7 WERE NOT EXPECTED TO BE A REPRESENTATION OF A TWO-LOOP PWR BECAUSE OF SCALE DISTORTIONS IN THE SEMISCALE MOD-1 SYSTEM RELATIVE TO A TWO-LOOP PWR.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
ACCIDENT, LOSS OF COOLANT - REACTOR, OUR - BLOWDOWN - EMERGENCY COOLING SYSTEM - CORE MELTING SYSTEM - OUT OF PILE EXPERIMENT - THERMAL EXPERIMENT - HYDRAULIC EXPERIMENT - PLUM. TWO PHASE - NRC-2

122005

EXPERIMENT DATA REPORT FOR SEMISCALE MOD-1 TESTS S-05-0 AND S-05-7 (ALTERNATE EGC INJECTION TESTS)
EDWARDS NATIONAL ENGINEERING LAB., IDAHO FALLS
REEI-122000-1, 205 PGS. 11 TABS. 356 FIGS. 5 REP., JUNE 1977

TITANIUM-SHEATED TYPE K THERMOCOUPLES EMENDED IN THE CLADDING WALL OF TITANIUM-SHEATED MACR-2 HEATER RODS ARE DESCRIBED IN THE TEXT. THESE THERMOCOUPLES CONSTITUTE PART OF A PROBE EXTENDED TO CHARACTERIZE THE UNCERTAINTY OF MEASUREMENTS MADE BY SURFACE-MOUNTED CLADDING THERMOCOUPLES ON NUCLEAR FUEL RODS. PURIFICATION AND INSTALLATION DETAILS, AND LABORATORY TESTING OF SAMPLE THERMOCOUPLE INSTALLATIONS ARE INCLUDED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
ACCIDENT, LOSS OF COOLANT - REACTOR, OUR - BLOWDOWN - EMERGENCY COOLING SYSTEM - CORE MELTING SYSTEM - OUT OF PILE EXPERIMENT - THERMAL EXPERIMENT - HYDRAULIC EXPERIMENT - PLUM. TWO PHASE - NRC-2

122003

EXPERIMENT DATA REPORT FOR SEMISCALE MOD-1 TEST S-06-6 (LOFT COUNTERPART TEST)
EDWARDS NATIONAL ENGINEERING LAB., IDAHO FALLS
REEI-122000-1, 205 PGS. 11 TABS. 356 FIGS. 5 REP., JUNE 1977

RECORDED TEST DATA ARE PRESENTED FOR TEST S-06-6 OF THE SEMISCALE MOD-1 SIMULATED CORE THERMAL RESPONSE AND INJECTION TEST SERIES CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING A LOFT IN A PWR. TEST S-06-6 WAS CONDUCTED FROM INITIAL CONDITIONS OF 2772 PSI AND 520 F TO INVESTIGATE THE RESPONSE OF THE SEMISCALE MOD-1 SYSTEM TO A DEPRESSION AND REFLOOD TRANSIENT FOLLOWING A SIMULATED DOUBLE-ENDERS LOFT OF THE BLOWDOWN COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEGS OF THE INLET AND BLOWDOWN LOOPS TO SIMULATE EMERGENCY CORE COOLANT. A REACTOR WAS TO ASSESS THE INFLUENCE OF THE GEOMETRY ON CORE THERMAL AND SYSTEM RESPONSE AND ON THE SUBCOOLED AND LOW QUALITY MISS FLOW RATES AT THE SPECIFIED LOCATIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
ACCIDENT, LOSS OF COOLANT - REACTOR, OUR - BLOWDOWN - EMERGENCY COOLING SYSTEM - CORE MELTING SYSTEM - OUT OF PILE EXPERIMENT - THERMAL EXPERIMENT - HYDRAULIC EXPERIMENT - PLUM. TWO PHASE - NRC-2

129103

CARO ON
QUARTERLY TECHNICAL PROGRESS REPORT ON WATER REACTOR SAFETY PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH (JAN-MAR 1977)
EDWARDS NATIONAL ENGINEERING LAB., IDAHO FALLS
REEI-122000-1, 205 PGS. 11 TABS. 356 FIGS. 5 REP., JUNE 1977

LCC-03 CONTINUED

COMPONENT COSTS

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

REACTOR, PAL + SAFETY ANALYSIS + OUT OF CORE EXPERIMENT + COMPUTER PROGRAM, DIGITAL + COMPARISON, THEORY AND EXPERIENCE + EMERGENCY COOLING SYSTEM + BLEEDDOWN + ACCIDENT, LOSS OF COOLANT + MVC-2 + MVC-3 + MVC-4

LCC-03

BUSH, NL

LEFT FUEL, MIXED DESIGN, CHARACTERIZATION, AND FABRICATION PROGRAM

SLC-06 NATIONAL ENGINEERING LAB., IDAHO FALLS

THREE-HUNDRED-128 Pages, 10 Tab., 30 Figs., June 1977

THE LEFT-FUEL, MIXED DESIGN, PROGRAM WHICH HAS RESULTED IN THE ACCELEBATION OF MANY TECHNICAL ACTIVITIES OF INTEREST TO PRESSURIZED WATER REACTOR FUEL DESIGN, DEVELOPMENT, AND SAFETY. THIS REPORT SUMMARIZES THE HIGHLIGHTS WHICH INCLUDES: 1) DETERMINATION OF FUNDAMENTAL, HIGH-TEMPERATURE LEAK MATERA, PROPERTIES; 2) DESIGN INVENTION RELATIVE TO INSIDE DETERMINATION ATTACHMENT INCTOR; 3) IMPLEMENTATION OF ADVANCED UNIVERSITY FUEL, SIZING AND FABRICATION TECHNIQUES, AND 4) PLANNING AND EXECUTION OF A MILLION DOLLAR DESIGN, CHARACTERIZATION, AND FABRICATION PROGRAM FOR PRESSURIZED WATER REACTOR FUEL.

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

REACTOR, PWR + ACCIDENT, LOSS OF COOLANT + BLEEDDOWN + LEFT (5-W) + EMERGENCY COOLING SYSTEM + MVC-2

LCC-03

COLLINS, NL + PATTON, NL + SACKETT, AZ

EXPERIMENTAL DATA REPORT FOR PRESSURE CALC-06-1 TEST 5-06-1 (LEFT COMPARTMENT TEST)

SLC-06 NATIONAL ENGINEERING LAB., IDAHO FALLS

THREE-HUNDRED-128 Pages, 10 Tab., 30 Figs., July 1977

RECORDED DATA ARE PRESENTED FOR TEST 5-06-1 OF THE PRESSURE CALC-06-1 LEFT COMPARTMENT TEST SERIES. THESE TESTS ARE AMONG SEVERAL PRESSURE CALC-06-1 EXPERIMENTS CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING AN EXTERNAL GLOBAL LITE IN A PWR. TEST 5-06-1 WAS CONDUCTED FROM INITIAL CONDITIONS OF 10 BAR UPS AND 364 K TO INVESTIGATE THE RESPONSE OF THE PRESSURE 5-06-1 SYSTEM TO A DEPRESSURIZATION AND FLOOD TRANSIENT FOLLOWING A SIMULATED FUEL-ENDED OUTFLOU OF THE OPENEND LOOP COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INACT LOOP TO SIMULATE EMERGENCY COOLANT INJECTION IN A PWR.

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

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

LCC-01

CAPELLA, C.

ANALYSIS OF STANDARD PROBLEM SIX (PRESSURE TEST 5-06-1) DATA

SLC-06 NATIONAL ENGINEERING LAB., IDAHO FALLS

THREE-HUNDRED-104 Pages, 1 Tab., 33 Figs., 11 Figs., August 1977

TEST 5-06-1 OF THE PRESSURE CALC-06-1 BLEEDDOWN HEAT TRANSFER TEST SERIES HAS BEEN CONDUCTED TO SUPPLY DATA FOR THE STANDARD PROBLEM SHEET. TO DETERMINE THE CREDIBILITY OF THE DATA AND TO ESTABLISH THE RELIABILITY OF THE RESULTS, AN ANALYSIS OF THE RESULTS OF TEST 5-06-1 WAS PERFORMED AND IS PRESENTED. THIS ANALYSIS CONSISTED OF INVESTIGATIONS OF SYSTEM HYDRAULIC AND CORE THERMAL DATA. THE GAME DATA HAS BEEN EXAMINED THROUGH SIMULATIONS OF THE DATA WITH DATA AND SIMULATIONS FROM RECORDED SITES TEST 5-06-1 AND THROUGH ASSESSMENT OF PHYSICAL EVENTS. THE CREDIBILITY OF THE CORE THERMAL DATA WAS BASED ON A THOROUGH ANALYSIS OF PHYSICAL EVENTS.

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

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

LCC-09

EVALUATION OF THE EFFECTS OF BREAT NOZZLE CONFIGURATION IN THE PRESSURE CALC-06-1 SYSTEM

SLC-06 NATIONAL ENGINEERING LAB., IDAHO FALLS

THREE-HUNDRED-118 Pages, 30 Tab., 26 Figs., August 1977

THE PRESSURE CALC-06-1 PROGRAM HAS UTILIZED TWO DIFFERENT BREAT NOZZLE CONFIGURATIONS IN THE TEST SYSTEM. AN EVALUATION HAS BEEN MADE TO DETERMINE THE EFFECT OF THESE BREAT NOZZLE CONFIGURATIONS ON THE THERMAL-HYDRAULIC RESPONSE DURING A REXT DOUBLE-ENDED COOLANT BLEEDDOWN ACCIDENT. THE FIRST NOZZLE IS A CONVERGENT-DECONVERGENT NOZZLE (MREY NOZZLE).
129304 CONTINUED

AND THE SECOND, AN ELEVATED CONSTANT AREA THROAT METER. ANALYSIS IS CONDUCTED PRIMARILY ON SYSTEM RESPONSE PHENOMENA OBSERVED TO BE AFFECTED BY THE NOSE CONFIGURATION AND CONVERGES ON THE FLUID RESPONSE AT THE BREAK AND THE RESULTING CORE BEHAVIOR DURING SUBCOOLED AND SATURATED DROPOUT.

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

JACOBS • INCIDENT, LOSS OF COOLANT SIMULATION • REACTOR TEST FACILITY • THERMAL HYDRAULIC ANALYSIS • BLINDING • CORE RESPONSE SPECTRUM • IEC-A FT ENG LAB • HRC-2

129306

WATSON ML • COLLING RL • SACKETT KE

EXPERIMENT DATA REPORT FOR SEMISCAL MOOD-1 TEST 5-06-2 (LOFT CONTEMPORARY TEST)

IDAM NATIONAL ENGINEERING LAB., IDAHO FALLS

335-ENG-1127, 202 PGS., Figs. Aug. 1977

RECORDED TEST DATA ARE PRESENTED FOR TEST 5-06-2 OF THE SEMISCAL MOD-1 LOFT CONTEMPORARY TEST SERIES. THESE TESTS ARE AMONG SEVERAL SEMISCAL MOD-1 EXPERIMENTS CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING AN HYDROHEATED LOOP IN A PWR. TEST 5-06-2 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15.623 PA AND 543 # K TO INVESTIGATE THE RESPONSE OF THE SEMISCAL MOD-1 SYSTEM TO A DEPRESSION AND RETURN TRANSIENT FOLLOWING A SIMULATED PCP-ENGENDERED OFFSET SHEAR OF THE BLOWER LOOP COOL LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COOL LEG OF THE INTACT LOOP TO SIMULATE EMERGENCY COOLANT INJECTION IN A PWR.

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

JACOBS • REACTOR TEST FACILITY • INCIDENT, LOSS OF COOLANT • REACTOR, PLT • SIMULATION • THERMAL HYDRAULIC ANALYSIS • EMERGENCY COOLING • SAFETY INJECTION • DATA COLLECTION • DATA PROCESSING • CORE REPLUGGING • PRESSURE TRANSIENT • IEC-A FT ENG LAB • HRC-2

129308

QUARTERLY TECHNICAL PROGRESS REPORT ON WATER REACTOR SAFETY PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH APRIL-JUNE 1977

IDAM NATIONAL ENGINEERING LAB., IDAHO FALLS

335-ENG-1147, 62 PGS., Aug. 1977


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

MB AND D PROGRAM • SAFETY PROGRAM • REACTOR, PLT • INCIDENT, LOSS OF COOLANT • EMERGENCY COOLING SYSTEM • NEAT FLUID, CRITICAL • PWR • FACILITY • THERMAL HYDRAULIC ANALYSIS • COMPARISON, THEORY AND EXPERIENCE • HRC-2 • HRC-3 • HRC-4 • JACOBS

131108

SILVERMAN S

PRINCIPLES OF OPERATION AND DATA REDUCTION TECHNIQUES FOR THE LRPT DRAG DISC TURBINE TRANSODER

IDAM NATIONAL ENGINEERING LAB., IDAHO FALLS

335-ENG-1190, 123 PGS., 31 Tabs., 82 Figs., Sept. 1977

FOR ALL-WATER FLOW IN PIES UP TO AND INCLUDING 11-IN. TO, THE PRESENCE OF THE DRAG DISC TURBINE THEORistically DISRUPTS THE VELOCITY PROFILE TO THE EXTENT THAT THE TURBINE METER OUTPUT IS PROPORTIONAL TO AVERAGE VELOCITY. FOR THE TEST DATA ANALYZED THE THREE TWO-FLUID MODELS REDUCE THE SCATTER AS COMPARED TO A SIMPLE HOMOGENOUS MODEL. FOR ANY GIVEN MODEL, PREDICTED STEAM AND LIQUID VELOCITIES FOR FORWARD FLOW AGREE CLOSELY WITH THOSE FOR REVERSE FLOW. THE TURBINE 10 IN. ABOVE 75 IN. IS ESTIMATED TO BE ABOUT 0.29 SEC IN WATER AND 11.9 SEC IN STEAM.

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

MB AND D PROGRAM • SAFETY PROGRAM • REACTOR, PLT • LOFT (S-HP) • TURBINE, INSTRUMENT, FLOW • MEASUREMENT • TESTING, FLow, TWO MODE • STEAM • WATER • HRC-2 • JACOBS

131107

MCLEAN G • LOBLIS GD • PETERSON AC

INVESTIGATION OF ALTERNATE ECC INJECTION CONCEPTS IN THE SEMISCAL MOD-1 SYSTEM

IDAM NATIONAL ENGINEERING LAB., IDAHO FALLS

335-ENG-1117, 176 PGS., 3 Tabs., 191 Figs., Sept. 1977

THE POTENTIAL BENEFITS AND RELATIVE MERITS OF FOUR ALTERNATE EMERGENCY COOLANT INJECTION (ECI)
INJECTION CONCEPTS ARE INVESTIGATED IN THE SEMISCALE MOD-1 SYSTEM. THE MAJOR COMPETED ENTHERMAL-
HYDRAULIC PHENOMENA ASSOCIATED WITH THE EFFECTIVENESS OF EACH ALTERNATIVE ECC INJECTION CONCEPT ARE
ALSO EXAMINED. THE PRINCIPAL MEANS USED TO EVALUATE THE VARIOUS CONCEPTS IS A COMPARATIVE
ANALYSIS OF EXPERIMENTAL DATA FOR A GIVEN CONCEPT (FROM TEST SERIES 3) WITH DATA FROM A BASELINE
CONCEPT ECC INJECTION TEST SERIES (S-94-6. run AS PART OF A PROJECT TEST SERIES). AND WITH DATA FROM
OTHER TESTS IN THE ALTERNATE ECC INJECTION TEST SERIES. AS A RESULT OF THIS INVESTIGATION,
CONCLUSIONS REGARDING THE EFFECTIVENESS OF THE DIFFERENT CONCEPTS IN THE SEMISCALE MOD-1
SYSTEM ARE REACHED, AND THE RELATIVE MERITS OF THE CONCEPTS ARE IDENTIFIED.

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

A AND B PROGRAM • SAFETY PROGRAM • REACTOR, LWR • ACCIDENT, LOSS OF COOLANT • BLOWDOWN • EMERGENCY COOLING
SYSTEM • THERMAL HYDRAULIC ANALYSIS OF LOFT EXPERIMENT • IDAHO FALLS • SAFETY INJECTION • NRC-2 • JACOBS

120370
ESPADA V • SACKETT RE
EXPERIMENTAL DATA REPORT FOR SEMISCALE MOD-1 TEST S-94-6 (LOFT COUNTERTOP TEST)
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TRES-NUMBE-1132 4, 200 PGS. 12 TAPS, 295 Figs. Refs. Sept. 1977

RECORDED TEST DATA ARE PRESENTED FOR TEST S-94-6 OF THE SEMISCALE MOD-1 LOFT COUNTERTOP TEST
SERIES. THESE TESTS ARE AMONG SEVERAL SEMISCALE MOD-1 EXPERIMENTS CONDUCTED TO INVESTIGATE THE
THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING A HYPOTHEZIZED LOSS-OF-COOLANT ACCIDENT IN A PWR.
TEST S-94-6 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15740 APA AND 393 K TO INVESTIGATE THE
RESPONSE OF THE SEMISCALE MOD-1 SYSTEM TO A DEPRESSURIZATION AND REFLOOD TRANSIENT FOLLOWING A
SIMULATED DOUBLE-ENDS OFFSET SHEAR OF THE BROKEN LOOP COLD LEG PIPING.

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

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

120280
HAGELSTROM SJ • HUNDELOCK BA • RACALDONO PE • OWEN DE
AXIAL GAS FLOW IN IMMEDIATE PUR FUEL RODS
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TRES-NUMBE-1150 4, 100 PGS. 10 FIGS. Refs. Sept. 1977

TRANSIENT AND STEADY STATE AXIAL GAS FLOW EXPERIMENTS WERE PERFORMED ON A IMMEDIATE COMMERCIAL
PUR FUEL RODS AT ARBITRARY TEMPERATURE AND 323 K. LUMPED FLOW EQUATIONS, AS USED IN THE PWR-T2
AND BREST FUEL BEHAVIOR CODES, WERE USED WITH THE GAS FLOW RESULTS TO CALCULATE EFFECTIVE FUEL
MODE RADIUS GAP, THE RESULTS OF THESE ANALYSES WERE COMPARED WITH MEASURED GAP SIZES OBTAINED
FROM METALLURGICAL EXAMINATION OF ONE FUEL ROD, USING MEASURED GAP SIZES AS INPUT, THE BREST
CODE WAS USED TO CALCULATE PRESSURE GROUPS AND PASS PLUMES AND THE RESULTS WERE COMPARED WITH THE
EXPERIMENTAL GAS FLOW DATA.

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

STEADY STATE EXPERIMENT • GAS • FLOW AXIAL • PUR ROD • SIMULATION TESTING • REACTOR, PUR • COMPARISON
• THEORY AND EXPERIENCE • IDAHO NAT ENG LAB • JACOBS • NRC-2

120664
WHITE JR • HOLTRAMH DJ
POSTTEST RELAP ANALYSIS OF LOFT EXPERIMENT LI-3A
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
TRES-NUMBE-1137 4, 190 PGS. 80 FIGS. 10 Refs. Oct. 1977

PRESENTS SELECTED RESULTS OF POSTTEST RELAP MODELING OF LOFT LOSS-OF-COOLANT EXPERIMENT LI-3A.
A DOUBLE-ENDS IMMEDIATE THERMAL CORE SAD HOPE WITH LOWER PLAIN Emergency CORE COOLANT INJECTION
COMPARISONS ARE PRESENTED BETWEEN THE POSTTEST PREDICTION, THE POSTTEST ANALYSIS, AND THE
EXPERIMENTAL DATA. IT IS CONCLUDED THAT PRESSURIZER MODELING IS IMPORTANT IN ACCURATELY
PREDICTING STYED BEHAVIOR DURING THE INITIAL PORTION OF SATURATED BLOWDOWN. USING MEASURED
INITIAL CONDITIONS REASONABLE SIMILAR SPECIFIED INITIAL CONDITIONS DID NOT IMPROVE THE RELAP
MODEL RESULTS SIGNIFICANTLY. USING FINE FLOW MODELIZE IN THE REACTOR VESSEL IMPROVED THE
PREDICTION OF THE SYSTEM PRESSURE HISTORY BY SIMPLIFYING STEAM CONDENSATION FACTS.

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

A AND B PROGRAM • SAFETY PROGRAM • ACCIDENT, LOSS OF COOLANT • SAFETY INJECTION • CORE SPRAY • MODEL
• COMPARISON, THEORY AND EXPERIENCE • BRESTFORD • EMERGENCY COOLING CONSIDERATIONS • COMPUTER PROGRAM • NRC-2 • JACOBS

121110
COLEMAN BL • COPPIN CE • SACKETT RE
EXPERIMENTAL DATA REPORT FOR SEMISCALE MOD-1 TEST S-10-1 (STEAM GENERATOR TUBE RUPTURE TEST)
IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS

PAGE 10
RECORDED TEST DATA ARE PRESENTED FOR TEST S-26-1 OF THE SEMISCALE MOD-1 STEAM GENERATOR TUBE RUPTURE TEST SERIES. THESE TESTS ARE AMONG SEVERAL SEMISCALE MOD-1 EXPERIMENTS CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING A HYPOTHESIZED LOSS-OF-COOLANT ACCIDENT IN A PWR. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INTACT AND BROKEN LOOPS TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR. SIXTEEN STEAM GENERATOR TUBE RUPTURES WERE SIMULATED BY A CONTROLLED INJECTION FROM A HEATED ACCUMULATOR INTO THE INTACT LOOP HOT LEG.

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


121109
PATTIN, N. - SACKETT, R.
EXPERIMENT DATA REPORT FOR SEMISCALE MOD-1 TEST S-26-2 (STEAM GENERATOR TUBE RUPTURE TEST) IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
NACE-11109 = 292 PPS, 12 TABS, 381 PGS, REF05, OCT 1977

RECORDED TEST DATA ARE PRESENTED FOR TEST S-26-2 OF THE SEMISCALE MOD-1 STEAM GENERATOR TUBE RUPTURE TEST SERIES. THESE TESTS ARE AMONG SEVERAL SEMISCALE MOD-1 EXPERIMENTS CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING A HYPOTHESIZED LOSS-OF-COOLANT ACCIDENT IN A PWR. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INTACT AND BROKEN LOOPS TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR. FOR TEST S-26-2, ACCUMULATOR INJECTION INTO THE INTACT LOOP HOT LEG WAS PROVIDED TO SIMULATE THE RUPTURE OF SIX STEAM GENERATOR TUBES.

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


121106
GILLMOR, M. - SACKETT, R.
EXPERIMENT DATA REPORT FOR SEMISCALE MOD-1 TEST S-26-3 (STEAM GENERATOR TUBE RUPTURE TEST) IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
NACE-11106 = 291 PPS, 12 TABS, 381 PGS, REF05, OCT 1977

RECORDED TEST DATA ARE PRESENTED FOR TEST S-26-3 OF THE SEMISCALE MOD-1 STEAM GENERATOR TUBE RUPTURE TEST SERIES. THESE TESTS ARE AMONG SEVERAL SEMISCALE MOD-1 EXPERIMENTS CONDUCTED TO INVESTIGATE THE THERMAL AND HYDRAULIC PHENOMENA ACCOMPANYING A HYPOTHESIZED LOSS-OF-COOLANT ACCIDENT IN A PWR. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INTACT AND BROKEN LOOPS TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR. TWELVE STEAM GENERATOR TUBE RUPTURES WERE SIMULATED BY A CONTROLLED INJECTION FROM A HEATED ACCUMULATOR INTO THE INTACT LOOP HOT LEG.

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


121700
ESBANIA, V. - SACKETT, R.
EXPERIMENT DATA REPORT FOR SEMISCALE MOD-1 TEST S-26-4 (STEAM GENERATOR TUBE RUPTURE TEST) IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
NACE-11100 = 250 PPS, 12 TABS, 403 PGS, NOV 1977

TEST S-26-4 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15,664 KPA AND 557 K TO INVESTIGATE THE RESPONSE OF THE SEMISCALE MOD-1 SYSTEM TO DEPRESSURIZATION AND REPLUG TRIBUTARY FOLLOWING A SIMULATED DOUBLE-CHORD OFFSET SHARP OF THE BROKEN LOOP COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INTACT AND BROKEN LOOPS TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR. THIRTY STEAM GENERATOR TUBE RUPTURES WERE SIMULATED BY A CONTROLLED INJECTION FROM A HEATED ACCUMULATOR INTO THE INTACT LOOP HOT LEG.

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

A. AND D PROGRAM - SAFETY PROGRAM - NWC-2 - STEAM GENERATOR - FAILURE, TUBING - ACCIDENT, LOSS OF COOLANT - REACTOR, LWR - HYDRAULIC EXPERIMENT - BLLOWDOWN - EMERGENCY COOLANT FLUSH - PRESSURE TRANSIENT - THERMAL TRANSIENT

120827
COLLINS, M. - SACKETT, R. - COMIN, C.
EXPERIMENT DATA REPORT FOR SEMISCALE MOD TEST S-26-5 (STEAM GENERATOR TUBE RUPTURE TEST) IDAHO NATIONAL ENGINEERING LAB., IDAHO FALLS
NACE-11107 = 250 PPS, 12 TABS, 380 PGS, REF05, NOV 1977

TEST S-26-5 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15,664 KPA AND 557 K TO INVESTIGATE THE RESPONSE OF THE SEMISCALE MOD-1 SYSTEM TO DEPRESSURIZATION AND REPLUG TRIBUTARY FOLLOWING A SIMULATED DOUBLE-CHORD OFFSET SHARP OF THE BROKEN LOOP COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INTACT AND BROKEN LOOPS TO SIMULATE EMERGENCY CORE COOLANT INJECTION IN A PWR. SEVENTEEN STEAM GENERATOR TUBE RUPTURES WERE SIMULATED BY A CONTROLLED INJECTION FROM A HEATED ACCUMULATOR INTO THE INTACT LOOP HOT LEG.
TEST D-2-6 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15700 kPa AND 551 K TO INVESTIGATE THE RESPONSE OF THE BENCHMARK HDB-1 SYSTEM TO A REPRESENTATION AND RELOAD TRANSIENT FOLLOWING A SIMULATED SINGLE-DIODE OFFSET DEFLECTION OF THE BENCHMARK COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INJECTOR AND BENCH MARK LEGS TO SIMULATE DISASTER COOLANT INJECTION IN A TEST. FOR TEST D-2-6, ACCELERATOR INJECTION INTO THE INJECTOR LOOP NOT LEG WAS PROVIDED TO SIMULATE THE RuptURE OF 20 STEAM GENERATOR TUBES.

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

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

THE TEST D-2-6 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15700 kPa AND 551 K TO INVESTIGATE THE RESPONSE OF THE BENCHMARK HDB-1 SYSTEM TO A REPRESENTATION AND RELOAD TRANSIENT FOLLOWING A SIMULATED SINGLE-DIODE OFFSET DEFLECTION OF THE BENCHMARK COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INJECTOR AND BENCHMARK LEGS TO SIMULATE DISASTER COOLANT INJECTION IN A TEST. FOR TEST D-2-6, ACCELERATOR INJECTION INTO THE INJECTOR LOOP NOT LEG WAS PROVIDED TO SIMULATE THE RuptURE OF 20 STEAM GENERATOR TUBES.

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

THE TEST D-2-6 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15700 kPa AND 551 K TO INVESTIGATE THE RESPONSE OF THE BENCHMARK HDB-1 SYSTEM TO A REPRESENTATION AND RELOAD TRANSIENT FOLLOWING A SIMULATED SINGLE-DIODE OFFSET DEFLECTION OF THE BENCHMARK COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INJECTOR AND BENCHMARK LEGS TO SIMULATE DISASTER COOLANT INJECTION IN A TEST. FOR TEST D-2-6, ACCELERATOR INJECTION INTO THE INJECTOR LOOP NOT LEG WAS PROVIDED TO SIMULATE THE RuptURE OF 20 STEAM GENERATOR TUBES.

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

THE TEST D-2-6 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15700 kPa AND 551 K TO INVESTIGATE THE RESPONSE OF THE BENCHMARK HDB-1 SYSTEM TO A REPRESENTATION AND RELOAD TRANSIENT FOLLOWING A SIMULATED SINGLE-DIODE OFFSET DEFLECTION OF THE BENCHMARK COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INJECTOR AND BENCHMARK LEGS TO SIMULATE DISASTER COOLANT INJECTION IN A TEST. FOR TEST D-2-6, ACCELERATOR INJECTION INTO THE INJECTOR LOOP NOT LEG WAS PROVIDED TO SIMULATE THE RuptURE OF 20 STEAM GENERATOR TUBES.

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

THE TEST D-2-6 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15700 kPa AND 551 K TO INVESTIGATE THE RESPONSE OF THE BENCHMARK HDB-1 SYSTEM TO A REPRESENTATION AND RELOAD TRANSIENT FOLLOWING A SIMULATED SINGLE-DIODE OFFSET DEFLECTION OF THE BENCHMARK COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INJECTOR AND BENCHMARK LEGS TO SIMULATE DISASTER COOLANT INJECTION IN A TEST. FOR TEST D-2-6, ACCELERATOR INJECTION INTO THE INJECTOR LOOP NOT LEG WAS PROVIDED TO SIMULATE THE RuptURE OF 20 STEAM GENERATOR TUBES.

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

THE TEST D-2-6 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15700 kPa AND 551 K TO INVESTIGATE THE RESPONSE OF THE BENCHMARK HDB-1 SYSTEM TO A REPRESENTATION AND RELOAD TRANSIENT FOLLOWING A SIMULATED SINGLE-DIODE OFFSET DEFLECTION OF THE BENCHMARK COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INJECTOR AND BENCHMARK LEGS TO SIMULATE DISASTER COOLANT INJECTION IN A TEST. FOR TEST D-2-6, ACCELERATOR INJECTION INTO THE INJECTOR LOOP NOT LEG WAS PROVIDED TO SIMULATE THE RuptURE OF 20 STEAM GENERATOR TUBES.

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

THE TEST D-2-6 WAS CONDUCTED FROM INITIAL CONDITIONS OF 15700 kPa AND 551 K TO INVESTIGATE THE RESPONSE OF THE BENCHMARK HDB-1 SYSTEM TO A REPRESENTATION AND RELOAD TRANSIENT FOLLOWING A SIMULATED SINGLE-DIODE OFFSET DEFLECTION OF THE BENCHMARK COLD LEG PIPING. DURING THE TEST, COOLING WATER WAS INJECTED INTO THE COLD LEG OF THE INJECTOR AND BENCHMARK LEGS TO SIMULATE DISASTER COOLANT INJECTION IN A TEST. FOR TEST D-2-6, ACCELERATOR INJECTION INTO THE INJECTOR LOOP NOT LEG WAS PROVIDED TO SIMULATE THE RuptURE OF 20 STEAM GENERATOR TUBES.
127295 CONTINUED

An instrumented piping spool piece has been designed to improve the accuracy and precision of threephase flow measurements. The algorithm for calculating nonstationary velocities by instrument signal correlation was modified to use the normalized cross-correlation function rather than the cross-correlation function to determine transient time. A preliminary feasibility study of innovative two-phase flow instruments has been conducted.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
FLOW, TWO PHASE, INSTRUMENT, FLOW MEASUREMENT, FLOW EQUIPMENT DEVELOPMENT, NRC-2

121199 THOM-93 OHMICH NA BOMANIC RE EADS IG
REPORT/PROGRESS REPORT ON BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM FOR JANUARY-MARCH 1977
NRC RITZ NATIONAL LAB., TENN.
ONIA-19460-T-189 = .04 PPS, TEXT, PICS. 8 FACS. AUG. 1977

DEPRESSURIZATION RATES AND MASS FLOWS WERE QUANTITATIVELY SIMILAR TO THOSE PREDICTED FOR
PRESSURIZED-WATER REACTORS IN SAFETY ANALYSIS REPORTS. TESTS WERE MADE FROM AN INITIAL HEATER
AND POWER OF 0.8 MW WITH OUTLET BOUNDING AS THE PRINCIPAL INDEPENDENT VARIABLE. HEAT TRANSFER
TO CRITICAL HEAT FLOW (CHF) WAS THE THREE TESTS WAS 1.5 TO 2.5 MW. WITH A RANGE FROM 0.8 TO 1.8
SEC. TESTS WERE INITIATED WITH A TRIP-BEAR GAS DENSITOMETER ON THE AIR-WATER LOOP, WITH FLOW IN
THE VERTICAL DOWNWARD DIRECTION. PRELIMINARY MEASUREMENTS INDICATED THAT WITH FLOW DISIPATING
LOCATED UPSTREAM OF THE DENSITOMETER, THE MEAN DENSITY OBTAINED BY AVERAGE THE TRIP-BEAR DATA
HAD IN CONCLUSION WITH DENSITY MEASUREMENTS MADE BY THE DIAMETRAL BEAM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
AND A PROGRAM - SAFETY PROGRAM - REACTOR, PWR - ACCIDENT, LOSS OF COOLANT - BLOWDOWN, OUT OF PILE
EXPERIMENT, EMERGENCY COOLING SYSTEM - SAFETY INJECTION - COMPUTER PROGRAM - HEAT TRANSFER EXPERIMENT
OR HEAT EXCHANGERS - FLOW, TWO PHASE - FLOW THEORY AND EXPERIMENTS - THERMAL HYDRAULIC ANALYSIS - NRC-2, JACOBS

127000 WHITE ND, HEMICH NA PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM - THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT
FOR TEST 100
NRC RITZ NATIONAL LAB., TENN.
ONIA-18439-T-114 = 25 PPS, 3 TABS. 6 FACS. AUG. 1977

REDUCED INSTRUMENT RESPONSES ARE PRESENTED FOR THERMAL-HYDRAULIC TEST FACILITY (THE) TEST 100,
WHICH IS PART OF THE PWR PRESSURIZED-WATER-REACTIONS (PWR) BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS
PROGRAM. THE OBJECTIVE OF THE PROGRAM IS TO INVESTIGATE THE THERMAL-HYDRAULIC PHENOMENON
GOVERNING THE ENERGY TRANSFER AND TRANSPORT PROCESSES THAT OCCUR DURING A LOSS-OF-COOLANT
ACCIDENT IN A PWR SYSTEM. TEST 100 WAS CONDUCTED TO INVESTIGATE THE RESPONSE OF HEATER NO.
AND INSTRUMENTED SPILL PIECES WITH FLOW HIRODIZING SCREENS TO A DOUBLE-ENDED RUPTURE
WITH EQUAL BLOW OUT AT THE TEST SECTIO INLET AND OUTLET.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
REACTOR, PWR - ACCIDENT, LOSS OF COOLANT - BLOWDOWN, OUT OF PILE EXPERIMENT - THERMAL EXPERIMENT - HYDRAULIC
EXPERIMENT - OR, FLOW, TWO PHASE - NRC-2

127000 WHITE ND, HEMICH NA PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM - THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT
FOR TEST 101
NRC RITZ NATIONAL LAB., TENN.
ONIA-18439-T-115 = 28 PPS, 3 TABS. 5 FACS. SEPT. 1977

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

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
REACTOR, PWR - BLOWDOWN, OUT OF PILE EXPERIMENT - EMERGENCY COOLING SYSTEM - CORE REFLODING SYSTEM -
ACCIDENT, LOSS OF COOLANT THERMAL EXPERIMENT - HYDRAULIC EXPERIMENT - NRC-2, JACOBS

131018 HEMICH RA, CRADICK GC, MUHAN CR, TUMAGE GC
PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM DATA EVALUATION REPORT - SYSTEM RESPONSE FOR THERMAL-
HYDRAULIC TEST FACILITY TEST SERIES 180
NRC RITZ NATIONAL LAB., TENN.
ONIA-19460-T-124 = 1.32 PPS, 151 FACS., 15 PPS., NOV. 1977
Availability = NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA 22161

B AND B PROGRAM = SAFETY PROGRAM + REACTOR. LOI + ACCIDENT, LOSS OF COOLANT + BLEEDDOWN + OUT OF PILE EXPERIMENT + HEAT TRANSFER EXPERIMENT + ORL. HEATERS + NRC-2 + OTHER THERMAL-HYDRAULIC ANALYSIS

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THIS REPORT DEVELOPS AN EXPLICIT SOLUTION TECHNIQUE TO DETERMINE BOTH THE TRANSIENT SURFACE TEMPERATURE AND THE TRANSIENT SURFACE HEAT FLUX OF ELECTRICALLY HEATED HOSES GIVEN THE POWER INPUT AN "INDICATED" INTERNAL TEMPERATURE DURING A SIMULATED LOSSES-OF-COOLANT ACCIDENT. A DIGITAL COMPUTER PROGRAM WAS DEVELOPED WHICH SOLVES A ONE-DIMENSIONAL, TRANSIENT, LAYERED, PARABOLIC, SIMPLIFIED FORMULATION OF THE HEAT CONDUCTION EQUATION AT EACH BUNDLE THERMOCOUPLE POSITION IN THE THERMAL-HYDRAULIC TEST FACILITY (THF).

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THE THERMAL-HYDRAULIC TEST FACILITY HAS COMPLETED 16 POWERED BLEEDDOWNS THROUGH JUNE 17, 1977. OF THESE 5 WERE COMPLETED WITH ALL 49 HOSES POWERED. 2 WERE COMPLETED WITH 2 INACTIVE HOSES, AND 9 WERE COMPLETED WITH 8 INACTIVE HOSES. INITIAL SYSTEM PRESSURE IN THESE TESTS WAS APPROXIMATELY 1350 PSI. TEST SECTION INLET TEMPERATURE WAS APPROXIMATELY 280 DEG C. APPROXIMATELY 507 DEG C, AND BREAK AREA WAS EQUIVALENT TO A 2000 BREAK WITH THE TOTAL AREA USUALLY SPLIT BETWEEN INLET AND OUTLET IN THE RATIO 8.61:8.6. HEATER AND POWER IN THESE TESTS WAS 58.100 OR 112 KG/MM2, MEAN TIME TO CRITICAL HEAT FLUX (CHF) VARIED FROM 0.7 TO 3.6 SEC. WITH DELAYED CPR OF APPROXIMATELY 2.5 SEC IN THE UPPER HALF OF THE BUNDLE IN SOME TESTS.

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THE OBJECTIVE OF THE ADVANCED TWO-PHASE INSTRUMENTATION PROGRAM IS THE APPLICATION OF ADVANCED INSTRUMENTATION SCIENCE TO IMPROVE THE ACCURACY AND PRECISION OF THE TRANSIENT TWO-PHASE MEASUREMENTS REQUIRED IN REACTOR SAFETY RESEARCH. A TWO-PHASE SYSTEM CONSISTING OF TWO IDENTICAL ELECTRODE ASSEMBLIES 20 MM IN 13 MM APART AND ROTATED 90 DEGREES WITH RESPECT TO EACH OTHER WAS TESTED IN AN AIR-WATER LOOP AT VOID FRACTIONS OF 0.2 AND 0.9. RELATIVELY HIGH CONCERNATION WAS OBTAINED BETWEEN THE TWO ELECTRODE ASSEMBLIES, AND THE VELOCITY MEASUREMENTS OBTAINED FROM THE EXPERIMENTALLY DETERMINED PHASE-SHIFT CURVES WERE IN REASONABLE AGREEMENT WITH THOSE CALCULATED FROM GENERAL TWO-PHASE FLOW PRINCIPLES.

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THE TESTS WERE CONDUCTED IN AN IMPROVED ORIGINAL FLOAT TEST FACILITY TO PROVIDE HEAT TRANSFER
COEFFICIENT AND ENTRAINMENT DATA AT FORCED FLOWING RATES OF 1 INSEC. AND 3 INSEC. WITH ELECTRICALLY HEATED ROD BUNDLES WHICH HAD COSINE AND TOP SLOPED AXIAL POWER PROFILES. THE COSINE AXIAL POWER PROFILE TEST SERIES WERE COMPLETED IN 1975 AND THE TEST RESULTS WERE REPORTED IN VAC-0565. THESE TESTS EXAMINED THE EFFECTS OF INITIAL CLAD TEMPERATURE, VARIABLE STEPPED AND CONTINUOUSLY VARIABLE FORWARDING RATES, MOUTHING HEAT RELEASE, NEW PEAK POWER, CONSTANT LOW FLOWING RATES, COOLANT SUPPLYING, HOT AND COLD CHANNEL ENTRAINMENT, AND BUNDLE STORING AND GENERATED POWER.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 O STREET, WASHINGTON, D.C. 20545 50 CENTS/PAGE — MINIMUM CHARGE $2.00
This progress report covers continued activities associated with modeling the transport and deposition of fission products during the planned shutdown time period following a terminated test. The original scope of the overall project has been expanded to include consideration of conditions consistent with those leading to postulated core meltdown situations. However, our efforts are to be concerned with the analysis of postulated melt-down conditions and the extension and revision of the source codes for application to these conditions.


JACOB P. MISSION PRODUCT TRANSPORT / ANALYTICAL MODEL / CORE RELIABILITY / COMPUTER PROGRAM / DEPOSITION / NRC-3 / AEC-6

130790
MARY N. GALAGNIOV / LOWRY H. RAMOTH D.J.
EVALUATING STRENGTH AND SUSTAINABILITY OF ISOMATERIAL ZIRCALOY. QUARTERLY PROGRESS REPORT JULY THROUGH SEPTEMBER 1977
BATTLE COLUMBIA LABS. CHIC
BRI-2456-1000 +. 342 PPS. 7 Tabs. 10 Figs. Oct. 1977

Intermediate core-dam failure tests results obtained on M. O. HOBSON spent-fuel cladding are reported. Analysis of intermediate-annearing, tensile-test data indicate a first-order time dependence for the failure. Analysis techniques with an activation energy similar to that for zirconium self-diffusion. Transient time-dam failure tests of the cladding conducted at initial pressures ranging from 2000 to 16,000 psi, produced burst temperature and burst incutility measurements similar to those required for uniaxially zirconium tubing. Differences in burst temperature are associated with axial and angular time temperature variations. No influence of axial restraint during the burst galling and failure stages has been observed on burst burst response.


B and D PROGRAM / SAFETY PROGRAM / FUSION TESTS / FUEL DEGRADATION / REACTOR SAFETY / FAILURE / CLOTHING / THERMAL TRANSFER / NRC-3 = ZIRCALOY

130923
STALLIN J. SIMPSON P. JORDAN L. DENING R.J.
MISSION PRODUCT TRANSPORT ANALYSIS: QUARTERLY PROGRESS REPORT JULY THROUGH SEPTEMBER 1977
BATTLE COLUMBIA LABS. CHIC
BRI-2456-1000 +. 37 PPS. 6 Tabs. 9 Figs. Nov. 30. 1977

Technical progress during this quarter consisted of completing and checking the model for emergency core cooling system (ECCS) water scrubbing of the fission products for inclusion in INEL melt-down conditions. Specifying the thermal-hydraulic conditions to be considered for fission product transport in postulated melt-down accidents: and considering a control volume framework for initial use in development of the transport model for melt-down conditions.


B and D PROGRAM / SAFETY PROGRAM / MISSION PRODUCT TRANSPORT / ANALYTICAL MODEL / CORE RELIABILITY / COMPUTER PROGRAM / DEPOSITION / NUCLEAR / AEC-6

130938
BAILTY H.J.
QUARTERLY PROGRESS REPORT TO NUCLEAR REGULATORY COMMISSION DIVISION OF MISSION FUEL BEHAVIOR RESEARCH AND PROGRAMS FOR PERIOD ENDING MARCH 1977
BATTLE PACIFIC NORTHWEST LABS. RICHLAND. WAS.
BRI-2456-1000 +. 100 PPS. 27 Tabs. 76 Figs. 65 Refs. April 1977

This is the first quarterly report issued on these programs at Battelle, Pacific Northwest Laboratories (PNL) sponsored by the fuel behavior research branch of the division of reactor safety research (PNL) of the nuclear regulatory commission (NLRC). This report covers the progress made during March 1977 and includes background information on work performed prior to January 1977. The 3 programs are an experimental investigation of steady-state core programs; the second program, procurement and irradiation test programs; and the pressure transducer development and test. Information on instrumentation qualification program.


B and D PROGRAM / BATTLE NORTHWEST / AGENT. AEC / STEADY STATE / COMPUTER PROGRAM / FUEL ELEMENTS / IRRADIATION TESTING / INSTRUMENT. PRESSURE / QUALIFICATION / NRC-3 / JACOB

131001
HARRISON P. MARSHALL D.R.
COMPARATIVE ANALYSIS OF PELLET CLOTHING INTERACTION FROM IPA-421 A & IPA-430 NUCLEAR TESTS BATTLE PACIFIC NORTHWEST LABS. RICHLAND. WAS.

131001
LIGHT WATER REACTOR SAFETY RESEARCH PERFORMED JANUARY THROUGH MARCH 1977 IS DISCUSSED, INCLUDING THE FOLLOWING: A TEST SERIES TO EVALUATE THE EFFECTIVENESS OF DIFFERENT EMERGENCY CORE COOLANTS USING A DOUBLE-PODDED CORE. A 3.3-POD TEST SERIES WAS INITIATED TO INVESTIGATE THE EFFECT OF BREAK NUCLEUS GEOMETRY ON BREAK URANUS AND SYSTEM RESPONSE DURING A 250 DOUBLE-PODDED COLD LEG BREAK TEST. THE ANALYSIS OF RESULTS OF SPOUT NUCLEAR TESTS SHOWED A REVIEW OF POST-POD AND POST-POD REACTOR 1 TRANSPORT AND SERIES FEED IN ADDITION TO NEW REACTOR 1 TRANSFER EXPERIMENTS WAS COMPLETED. A MORE GENERAL VERSION OF THE RELAP3 PILOT CORE (VERSION 2) HAS BEEN COMPLETED. EXPERIMENTAL RESULTS FROM TESTS CONDUCTED AT THE HAMPTON CONFINEMENT TEST FACILITY IN MAINE AND THE BAY TEST FACILITY IN FRANKFURT, GERMANY, WERE USED FOR VERIFICATION OF COMPUTER CONTAINMENT CODES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, WASH. DEPT. OF COMMERCE, SPRINGFIELD, VA 22161

129062
ALLISON CH = FLOUGER DA = CONACHER DH = MICHMAN AL
EMERGENCE EFFECTS TEST SERIES TEST 10-2 TEST RESULTS REPORT
NATIONAL ENVIRONMENT LAB., I290 FALDS
1976-01-1070 +, 120 PAGES, AUG. 1977
DESCRIPTS THE RESULTS OF THE FOURTH TEST, TEST 10-2, USING FOUR, 0.97-IN. DIA. PWW-FIYE FUEL RODS WITH DIFFERENCES IN DIAMETER, GAP AND CLADDING MATERIAL. THE OBJECTIVE WAS TO PROVIDE INFORMATION ABOUT THE EFFECTS OF THESE DIFFERENCES IN FUEL ADD BEHAVIOR DURING OASISEQUILIBRUM AND FILM BOILING OPERATIONS. THE EFFECT OF INITIAL GAP SIZE, CLADDING MATERIAL AND BRAZED JOINT TYPE, A FAST POWER INCREASE, AND SUSTAINED FILM BOILING OPERATIONS. THE DISCUSSIONS ARE BASED ON ASHERED TEST DATA, PEARL-RAPID POST-OBSERVATION EXAMINATION Results, AND COMPARISONS OF RESULTS WITH POST-POD COMPUTER MODEL CALCULATIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, WASH. DEPT. OF COMMERCE, SPRINGFIELD, VA 22161

129063
JACOBS R D PROGRAM = FUEL MOD. = REACTOR MOD. = ECC NOB = HMM 2 3 21 THE 2 3

129064
QUARTERLY TECHNICAL REPORT ON WATER REACTOR SAFETY PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH APRIL-JUNE 1977
NATIONAL ENVIRONMENT LAB., I290 FALDS
1976-01-1147 +, 90 PAGES, AUG. 1977
RESULTS FROM THE PREVIOUSLY CONDUCTED MOD-1 ECC INJECTION TEST SERIES WERE ANALYZED. TESTS IN THE MOD-1 CONDUCTED TEST SERIES WERE COMPLETELY INJECTED INTO THE UPPER PLANE THROUGH USE OF THE L-EN PRESSURE INJECTION SYSTEM. THE MOD-1 BREAK TEST PROGRAM SUCCESSFULLY COMPLETED BREAK-INDUCED COOLING TEST SERIES 1-3 1 1-3 IN THE MOD-1 BREAK TEST SERIES WERE COMPLETED WHICH INCLUDED INJECTION OF EMERGENCY CORE COOLANTS INTO THE UPPER PLANE THROUGH USE OF THE L-EN PRESSURE INJECTION SYSTEM. THE MOD-1 BREAK TEST PROGRAM SUCCESSFULLY COMPLETED BREAK-INDUCED COOLING TEST SERIES 1-3.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, WASH. DEPT. OF COMMERCE, SPRINGFIELD, VA 22161

129065
ON AND b PROGRAM = SAFETY PROGRAM = REACTOR, LIFE = ACCIDENT, LOSS OF COOLANT = EMERGENCY COOLING SYSTEM = REACTOR MOD. = CUMULATIVE, LOSS OF COOLANT, CRITICAL COOLING SYSTEM = FUEL TO CLINIC PROGRAM = CAREER BEHAVIOR = THEORETICAL HYDRAULIC ANALYSIS = COMPARISON, THEORY AND EXPERIENCE = HMM 2 3 HMM 2 3 4 HMM 4 5 JACOBS
QUAD 63 FABIAN GC + MEMBER AS + ALLISON CH
IMMOLATION EFFECTS TEST SERIES SCORING TEST 2 TEST RESULTS REPORT
IDAM NATIONAL ENGINEERING LAB., IDAHO FALLS
THREE-HALF-1906 - 1906 PPS., 20 TONS. 130 Figs. SEPT. 1977

This test, which was conducted in the power burst facility, used four 0.97-inch-long, pressurized water reactor type fuel rods. Three of these rods were fabricated from previously irradiated materials and the fourth was fabricated from unirradiated materials. The objectives of this scoring test were to evaluate fabrication procedures, handling plans, and reactor capabilities for irradiated fuel rod materials as well as to acquire fuel rod behavior data. This report presents the fuel behavior data acquired during the test. The fuel rods were subjected to a preconditioning period, followed by a power increase to a peak power of 61 kWt and a flow reduction to induce film boiling. This report describes the data obtained on each phase of the test and the posttest condition of the fuel rods.

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

D AND D P رب: SAFETY PROGRAM + DIABOLATION TESTING + FUEL ROD + FILM BOILING + EXAMINATION + REACTOR,
PUR + THEATRAL TRANSIENT + PEP (S-P-R) + NRC-3

221160
QUAD 63 Q ALLISON CH + FABIAN GC
IMMOLATION EFFECTS TEST SERIES SCORING TEST 1 TEST RESULTS REPORT
IDAM NATIONAL ENGINEERING LAB., IDAHO FALLS
THREE-HALF-1906 - 1906 PPS., 20 TONS. 130 Figs. SEPT. 1977

This report describes the results of the first scoring test in the Immolation Effects Test Series. This test was used in an unirradiated, three-inch-long, unirradiated fuel rod. The objectives of this test were to thoroughly evaluate the remote fabrication procedures to be used for irradiated rods in future tests, handling plans, and reactor operations. Additionally, selected fuel behavior data were obtained. The fuel rod was subjected to a series of preconditioning power cycles followed by a power increase which caused it to be 61 kWt at the thermal power level. The reactor fuel rod failed following reactor shutdown as a result of heavy internal and external cladding oxidation and embrittlement which occurred during the film boiling operation.

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

D AND D P رب: SAFETY PROGRAM + REACTOR, PUR + PHYSICS + FUEL ELEMENTS + FAILURE, FUEL ELEMENT + EMBRITTLEMENT + DIABOLATION + FILM BOILING + STEEL STAINLESS + NRC-3 + JACOB + GEOIDING

122763
IMMOLATION EFFECTS TEST SERIES TEST 1 C-3 POSTIMMOLATION EXAMINATION
IDAM NATIONAL ENGINEERING LAB., IDAHO FALLS
THREE-HALF-1906 - 163 PPS., 20 TONS. 65 Figs. SEPT. 1977

The results of the postimmolation examination of four boiling water reactor type, zircaloy-clad, unirradiated fuel rods tested as part of the thermal fuel rod behavior program are discussed. These rods were fabricated from unirradiated fuel rods tested in the LC-3 test, which was conducted to obtain experimental data from which the fuel rod conductance values could be determined. With the steady state integral of the power oscillation method, evidence is presented showing that significant amounts of heat had been present in two of these fuel rods during testing. For the LC-3 fuel rods that remained intact during the test, measurements of fuel pellet-to-cladding gap, as well as the surface areas of the fuel cracks at several axial locations are presented. A total effective radial gap is calculated and the fuel structure and porosity are analyzed.

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

D AND D P رب: SAFETY PROGRAM + REACTOR, PUR + PHYSICS + FUEL ROD + IMMOLATION TESTING + ZIRCALOY CLAD + NEUTRON DISTRIBUTION + FUEL TO CLAD + PEP (S-P-R) + EXAMINATION + NRC-3

122764
FABIAN GC + ALLEN CH + ALLISON CH + RODER GD
IMMOLATION EFFECTS TEST SERIES TEST 1 C-3 TEST RESULTS REPORT
IDAM NATIONAL ENGINEERING LAB., IDAHO FALLS
THREE-HALF-1906 - 137 PPS., 10 TONS. 16 Figs. SEPT. 1977

Test 1 C-3 used four 0.97-inch-long, pressurized water reactor type fuel rods fabricated from previously irradiated fuel. The fuel rods were subjected to a preconditioning period followed by a power ramp to 61 kWt. One rod failed approximately 62 seconds after the reactor was shut down as a result of cladding embrittlement due to excessive cladding deformation. Test data are compared with data from computer calculations and data from a previous immolation effects test in which four irradiated fuel rods of a similar design were tested. Test 1 C-3 results indicated that the irradiated state of the fuel rods did not significantly affect fuel rod behavior during normal, abnormal, and an event (fuel rod boiling) conditions.

AVAILABILITY: NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
ASSUMPTIONS
- AND OR PROGRAM = SAFETY PROGRAM = REACTOR; FPD & FUEL DESG & PDP (1-89) = EXPERIMENTAL = FILM SOURCING = MP3 = RELAXATION = THERMAL TRANSFER = ENRICHMENT = OXIDATION = IRRADIATION TESTING

RESULTS
- USING THE STANSFIELD GS MULTIPLE DATA AND GAMMA SPECTRA ON INDIVIDUAL SNF/PP-IV FISSION PRODUCT NUCLEI
- LA-AMHERST-G036-66 V. 100 PPS: 1 DPS, DEC, 1970

- RELATIVE AND GAMMA GROUP-ENERGY SPECTRA ARE CALCULATED BY 100 INDIVIDUAL FISSION-PRODUCT
- ACHIEVE MULTIPLE SPECTRUM, DATA IN ENDF/B-IV FISSION-PRODUCT FILES. TO GT-47 SPECTRA, IN UPSETTED
- GROUP-TO-GROUP STRUCTURE, AND THE GAMMA SPECTRA, IN UNIQUE-GROUP 150-GROUP STRUCTURE BETWEEN 6 AND
- 7.5 EV, ARE SIMULTANEOUSLY TAKEN FOR COLLAPSE TO ANY DESIRED PRACTICAL GROUP STRUCTURE.
- THESE SPECTRA HAVE BEEN USED IN DECAY-HEAT DURATION CALCULATIONS FOR COMPARISON WITH
- CORRESPONDING EXPERIMENTAL DATA AND GAMMA SPECTRA.

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

DATA COLLECTION + COMPUTER PROGRAM = DECAY HEAT + SPECTRUM, GAMMA + MHE-3 + RADONIC CLUE

RESULTS
- USING THE STANSFIELD GS MULTIPLE DATA AND EXPERIMENTAL RELATIVE FISSION-PRODUCT DATA AND GAMMA SPECTRA FROM 235U THERMAL
- FISSION
- LA-AMHERST-G036-66 V. 100 PPS: 1 DPS, DEC, 1970

- RELATIVE FISSION-PRODUCT DATA AND GAMMA SPECTRA FOLLOWING SHORT AND LONG 235U THERMAL NEUTRON
- ARE CALCOLED FOR A NUMBER OF COOLING PERIODS. THE RESULTS OF THESE CALCULATIONS BASED ON ENDF/B-IV FISSION-PRODUCT DATA ARE
- COMPARED WITH CORRESPONDING EXPERIMENTAL DATA AVAILABLE RECENTLY FROM SEVERAL RESEARCH
- EXPERIMENTISTS (LA. AMHERST SCIENTIFIC LABORATORY, BRENNER NATIONAL LABORATORY, AND THE
- UNIVERSITY OF ILLINOIS). THE HARMONIC THINKING GENERAL AGREEMENT BETWEEN CALCULATIONS
- AND EXPERIMENT INDICATES THE ABSENCE OF ENDF/B-IV DATA FOR SUCH CALCULATIONS.

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

A AND OR PROGRAM = SOURCE, RADIATION + FISSION PRODUCT ACTIVITY, GMO5 + GAMMA EMITTER + GAMMA EMITTER + CODES AND STANDARDS = DECAY HEAT + MHE-3 + JACOB

RESULTS
- USING THE STANSFIELD GS MULTIPLE DATA AND EXPERIMENTAL RELATIVE FISSION-PRODUCT SOURCE TERMS FOR REACTION APPLICATIONS
- LA-AMHERST-G036-66 V. 100 PPS: 1 DPS, DEC, 1970

- CALCULATED AND MEASURED FISSION-PRODUCT SOURCE TERMS ARE PRESENTED. THE EMPHASIS IS ON DECAY
- HEATING FOLLOWING FISSION FOR USE IN THE DEVELOPMENT OF A NEW AND DECAY HEAT STANDARD.
- CORRELATIONS WITH EXPERIMENT AND ARE GIVEN FOR DATA AND GAMMA SPECTRA AND ENERGY-INTEGRATED DATA;
- CALCULATED RESULTS FOR CASINO CONDENSED, ABARAR CHEMICAL BLOX, AND PHOTONEUTRON SPECTRA;
- ARE SUMMIRIZED. UNCERTAINTY IN THE DEVELOPMENT OF NUCLEUS DATA ENDF/B-IV) AND ON COMBINED DECAY
- HEAT EXPERIMENTS ARE PRESENTED.

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

A AND OR PROGRAM = SOURCE, RADIATION + FISSION PRODUCT ACTIVITY, GMO5 + GAMMA EMITTER + GAMMA EMITTER + CODES AND STANDARDS = DECAY HEAT + MHE-3 + JACOB

RESULTS
- USING THE STANSFIELD GS MULTIPLE DATA AND EXPERIMENTAL RELATIVE FISSION-PRODUCT SOURCE TERMS FOR REACTION APPLICATIONS
- LA-AMHERST-G036-66 V. 100 PPS: 1 DPS, DEC, 1970

- A CYTROPHIC SOIL-OFF CALORIMETER WAS USED TO MEASURE THE DECAY HEAT FROM THE PRODUCTS OF THERMAL-
- NEUTRON-ENERGIZED FISSION OF 235U. DATA ARE PRESENTED FOR COOLING TIMES BETWEEN 18 AND 1014 EUP 91
- SEC FOLLOWING A 2 E PROTON 13 DEC IDEM SPECTRUM AT 200 THERMAL-NEUTRON PER SEC. THE DATA AGREE
- WITHIN THE COMBINED UNCERTAINTIES WITH SIMULATION CALCULATIONS. USING THE ENDF/B-IV DATA BASE,
- THE EXPERIMENTAL DATA WERE COMBINED WITH SODIUM LIQUID SPECTRUM AT 200 THERMAL-NEUTRON PER SEC.
- FOR SHORT COOLING TIMES, THE RESULTS ARE ABOUT 7 TO 10% HIGHER TO THE CURRENT
- AMERICAN HAMILTON SCIENCE LABORATORY. THE UNCERTAINTY IN THESE RESULTS IS SIGNIFICANTLY
- SMALLER THAN THAT ASSIGNED TO THE STANDARD.

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

A AND OR PROGRAM = SOURCE, RADIATION + FISSION PRODUCT ACTIVITY, GMO5 + GAMMA EMITTER + GAMMA EMITTER + CODES AND STANDARDS + CORRELATION = DECAY HEAT + MHE-3 + JACOB
DECONSTRUCTION OF THE EFFECT OF STEAM PRESSURE ON THE OXIDATION RATE OF ZIRCALOY-2 YIELDED NEGATIVE RESULTS AT 750 DEG C (1382 DEG F) FOR STEAM AT 3,400 PSIG (23.5 MPa) for 24 HRS. IN ADDITION, EXPERIMENTS IN 3.0 MPa (435 PSI) STEAM ARE IN PROGRESS. MEASUREMENTS OF THE OXIDATION RATE OF ZIRCALOY-2 SPECIMENS IN PURE OXYGEN LEAD TO THE CONCLUSION THAT OXIDATION PREVIOUSLY REPORTED PRESENT IN GUIDED SPECIMENS HAS NO EFFECT ON OXIDATION RATES IN STEAM. ADDITIONAL IN SITU TESTS IN THE BEHAVIOR APPARATUS CONFORMED THE VALIDITY OF PREVIOUS ESTIMATES OF MAXIMUM TEMPERATURE MEASUREMENT UNCERTAINTIES.

AVAILABILITY — NRC PUBLIC DOCUMENT ROOM, 11545 STREET, WASHINGTON, D.C. 20555 100 CENTRALES — RENOWN CHARGE HOTLINE

FLEXIBILITY • METAL WATER REACTIONS • OXIDATION • STEAM • TEMPERATURE • NRC-3 • JACOBS
THE PURPOSE IS TO PROVIDE DATA ON DEFORMATION CHARACTERISTICS OF ZIRCONYL TUBES. TYPICAL OF LWF FUEL ELEMENTS, UNDER CONDITIONS OF AXIAL AND TANGENTIAL COMPRESSIVE STRESS. DATA WILL BE USED TO VERIFY AND IMPROVE MATERIAL BEHAVIOR CODES. THE EXPERIMENTS WERE DESIGNED AND WILL BE FABRICATED TO PROVIDE DATA FOR THE HIGH-PLUTONIUM EXPERIMENT, THE HPG-5-A. THIS REPORT IS DESIGNED TO PROVIDE AS A BASIS OF TECHNICAL AND SAFETY ASSESSMENT LEADING TO APPROVAL FOR INSTALLATION AND OPERATION IN THE HPG.

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

ZIRCONIUM ISOTOP ic-METAL WATER REACTIONS 1 INTERNAL TESTING - JACOBS & NKC-3

THE GOAL OF THE WORK IS TO IMPROVE THE UNDERSTANDING OF SHUTDOWN POWER IN REACTORS DUE TO RADIOACTIVE DECAY OF FISSION PRODUCTS, PARTICULARLY IN LIGHT WATER REACTORS IN THE PERIOD 0-10000 SECONDS AFTER SHUTDOWN. THE POWER LEVELS DURING THIS PERIOD IS A PRIMARY DETERMINANT OF CORE SHUTDOWN-COOLING REQUIREMENTS. MUCH OF THE WORK DONE DURING THIS QUARTER WAS IN PREPARATION OF
and pressure and material property on the behavior of the fuel rods and are considered. Fission gas generation and release are calculated as a function of burnup. The cladding deformation model includes material, elasto-plastic analyses and considers creep. FISP-2p is a modular code with each major computational subcode isolated within the core and coupled to the main code by subroutine calls and data transfer through argument lists.


Computer program: Fuel rod; Fuel, burnup; Deformation; Pressure, internal; Fission gas release; Cladding; Core; Reactor; Loss; Distribution; Temperature; HRI-6

121660 Quarterly technical progress report on water reactor safety programs sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, April-June 1977

 Idaho National Engineering Lab., Idaho Falls

Three-ring injection test results from the preliminary conducted semiscale HRI-1 ECC injection test series were analyzed. Testing in the load injection test series was essentially completed, and the steam generation tube rupture test series has begun. The tests in the alternate ECC injection test series were conducted with included injection of emergency core coolant into the upper plenum through use of the low pressure injection system. The loss-of-liquid test program successfully completed the first non-nuclear loss-of-coolant experiment LIL-1. A nuclear test, GC-2,3, in the power burst facility enabled RAC to perform a code research method of determining gap conductance to determine the effect of initial gap size, fill gas composition, and fuel density on the thermal performance of a light water reactor fuel rod.


OR and OR program: Safety program; Reactor, loss; Accident, loss of coolant; Emergency cooling system; Heat flux; Critical; Pump; Thermosistors; Heat conductance; Fuel, loss; Computer program; Core behavior; Thermal, hydraulic analyses; Comparison, theory and experience; HRI-2; HRI-3; HRI-4; JACOBS

127633 Status of materials at England in processe and FISP-code: Computer programs for calculating fission-product beta and gamma multigroup spectra from ENDF/B-VI DATA

Los Alamos Scientific Lab., NM

LA-UR-84-10595 * 02 pgs. 3 Figs. 12 Refs. May 1977

FISP processes ENDF/B-VI fission-product decay energy data to generate multigroup beta and gamma spectra from individual fission-energy fission-product nuclei. FISP2 further uses these spectra and the corresponding fission reaction rates calculated by the FISP-1 code to produce cumulative beta and gamma spectra in the same energy groups in which FISP generates individual isotope decay spectra.


Beta emitter; Beta ray; Gamma emitter; Gamma fission-product activity, cross-sections; Energy spectrum; Computer program; HRI-2; HRI-3

129472 Jackson, JF Nuclear reactor safety quarterly progress report January 1-March 31, 1977

Los Alamos Scientific Lab., NM

LA-UR-84-278 * 199 pgs. 27 Tabs. 6 Figs. 67 Refs. June 1977

A number of important advancements were made as part of the LWR safety research program. The tsub code was improved by adding all the remaining modules needed to analyze a large reactor core in a time of 1975. Six new components have been added in the analysis of several pertinent LWR safety problems. These included two-dimensional sodium boiling, extension of the core loss-of-flow accident initiation model, and some energy partition following disassembly. In the high safety program, the remaining equipment needed for the fuel heat-up experiment has been shipped. Several experiments were performed as part of the GCP core disruptive test program. In the containment evaluation area, a new capability in the code that allowed treatment of ice-condenser containment designs has been verified.


Reactor, safety research; Reactor, high; Reactor, loss; Accident analysis; Safety analysis; Reactor physics; Analytical model; Thermal hydraulic analysis; Jacobs; HRI-2; HRI-3; HRI-4; Reactor, kinetics; Reactor, GCP; Containment, ice condenser

129976 Jackson, JF; Stevenson, WG Nuclear reactor safety quarterly progress report: April 1-June 30, 1977

Los Alamos Scientific Lab., NM

LA-UR-84-934 * 199 pgs. 17 Tabs. 70 Figs. 80 Refs. Aug. 1977
The development of the TRACE computer code for analysis of light-water reactors involves the use of a three-dimensional, two-fluid hydrodynamic model to describe the two-phase flow of steam and water through the reactor vessel. One of the major problems involved in interpreting results from this code is the presentation of three-dimensional flux patterns. This code (POST) is used as a preprocessor in conjunction with a stand-alone three-dimensional two-phase hydrodynamics code (LIMP) which is a test bed for the three-dimensional algorithms to be used in TRACE.


FLOW, THE PHASE - COMPUTER PROGRAM \* JACOBS \* REACTOR, LWR \* NRC-4 \* ACCIDENT, LOSS OF COOLANT

NORTHWESTERN UNIVERSITY, EVANSTON, III.


HOP AND PIPE FITTINGS \* MODEL \* COMPUTER PROGRAM \* THERMAL-MECHANICAL EFFECT \* STRESS ANALYSIS \* REINFORCEMENT \* THERMAL-TRANSIENT \* STRESS \* NRC-1
7. US - WATER REACTOR SAFETY RESEARCH - METALLURGY AND MATERIALS

1232790
HAYWOOD GT • GEISEL FC • McDowell NC
CRITICAL EXPERIMENTS, MEASUREMENTS, AND ANALYSES TO ESTABLISH A CRACK ARREST METHODOLOGY FOR NUCLEAR PRESSURE VESSEL STEELS, REPORT JULY THROUGH SEPTEMBER 1978
BATTelle COLUMBIA LABS., OZIO
N-949-M6-1977-V, 100 PGS. & TABS. 50 Figs. July 1977

The program is implementing recommendations of a PuDF/NRC working group on crack arrest and includes work on dynamic fracture mechanics analysis, measurements of crack arrest in a variety of systems using chlorine experimental materials, and photoelastic studies of fast fracture and arrest, the initial phase, which is concerned mainly with the validation of a crack arrest theory, has been completed. In addition to the theoretical calculations, a crack arrest experiment conducted with a bonded specimen of 45,370 steel are described. Also, the program includes design schemes for improved failure to arrest material requirements in fracture toughness and crack arrest tests is evaluated.

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

8 AND D PROGRAM - CRACK AND MEASURE VESSELS • STEEL • STRESS ANALYSIS • MEC-5 • JACOBS

120700
GEISEL FC • GROSH JW • HALL GC • MARSHALL CH
CRITICAL EXPERIMENTS, MEASUREMENTS, AND ANALYSES TO ESTABLISH A CRACK ARREST METHODOLOGY FOR NUCLEAR PRESSURE VESSEL STEEL, 4TH QUARTERLY PROGRESS REPORT OCTOBER-DECEMBER 1976
BATTelle COLUMBIA LABS., OZIO
N-949-M6-1977-0, 60 PGS. & TABS. 28 Figs. July 1977

This project is a coordinated effort designed to establish a national crack arrest methodology for nuclear pressure vessel steel. The program involves Battelle's Columbia Laboratories, the University of Maryland, M.E. Research Laboratory, and the University of Illinois. The program is implementing recommendations of a PuDF/NRC working group on crack arrest and includes work on dynamic fracture mechanics analysis, measurements of crack arrest in a variety of systems using common experimental materials, and photoelastic studies of fast fracture and arrest.

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

STEEL • PRESSURE VESSELS • CRACK AND ANALYTICAL TECHNIQUE • PHOTOELASTICITY • MEC-5

120800
RODDAUGH FC • HAMER MA • EIBER PJ
REVIEW AND ASSESSMENT OF RESEARCH RELEVANT TO DESIGN ASPECTS OF NUCLEAR POWER PLANT PIPING SYSTEMS
BATTelle COLUMBIA LABS., OZIO

This report evaluates significant research on piping systems and correlates that research with design practices. The objective is to quantify the research/development practices in terms of the reliability of piping used in nuclear power plants. This report describes design practices relating to these codes: reviews operating experience of piping systems focusing on failures; discusses research on piping, with and without controlled flaws; assesses the reliability of piping systems; and reviews areas where additional research is necessary.

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

PIPPES AND PIPE FITTINGS • DESIGN CRITERIA • AND D PROGRAM • CODES AND STANDARDS • OPERATING EXPERIENCE • SUMMARY • FAILURE • PIP • FLANK • RELIABILITY ANALYSIS • MEC-5 • JACOBS

122287
BYBERGI EL • RODDAUGH FC • GROSH JW • SCHMIDT GC
DEVELOPMENT OF CRACK ARREST DESIGN CRITERIA FOR STEEL VESSELS AT 650-BAR PRESSURES IN PIPES AND PRESSURE VESSELS - FINAL REPORT APRIL 1, 1976 - JUNE 30, 1977
BATTelle COLUMBIA LABS., OZIO
N-949-M6-1976-37, 150 PGS. & TABS. NOV. 1977

A residual stress model for 650-bar welds in pressure vessels and pipes was developed and verified for welds ranging from 2 to 30 passes. The model is also accurately predicts residual deformations. Results indicate that the model can be extended to represent welds in pressure vessels. In addition, preliminary results directed at developing a simplified model of 650-bar welds show good agreement with data for one and two-pass welds.

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

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

131941
ALBRIGHT JW • CLARK RA • ANDERSON ME • COWIE JC • VOGEL RH
STEAM GENERATOR TUBE INTEGRITY PROGRAM QUARTERLY REPORT JANUARY 1 - MARCH 31, 1977
BATTelle PACIFIC NORTHWEST LABS., RICHLAND, WASH.
120627 CONTINUED

The pipe rupture study is designed to extend the understanding of failure-causing mechanisms and to provide improved capability for evaluating reactor piping systems to minimize the probability of failures. Following a detailed review to determine the effect most needed to improve nuclear system piping, phase II analytical and experimental efforts (phase IIa Part 1 started in 1973). This progress report summarizes the recent accomplishments of a broad program in (1) residual fatigue crack growth rate studies focused on live pressure pipe materials in a simulated non-pressurized coolant environment, and (2) studies directed at quantifying weld sensitization in type-304 stainless steel.

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

JACOBS • PIPE AND PIPE FITTINGS • FAILURE • REACTOR COOLANT • FATIGUE • CRACK • GROWTH/DEVELOPMENT • WELDS • SIMULATION • REACTOR, NUCLEAR • STEEL, STAINLESS • NRC-9

120730

1002 CI • DAILY JR • VARENDRA P • FOUNTAIN W • EMERICH M

PHYSICAL MECHANICS AND CRACK HEAT STUDIES OF CRACK PROPAGATION AND CRACK ARREST

UNIV. OF MARYLAND, COLLEGE PARK

A331-3261 • 281 APR. OCT. 1977

Describes the third year effort in research endeavors dealing with the characterization of dynamic aspects of fracture. The results included in this report are: (1) Verification of the DCB one-dimensional computer code; (2) Determination of a - k relationship from modified compact; (3) Verification of the fast fracture procedure for (1) Sub (4) measurement with machine-loaded C-106 specimen; (5) Fracture behavior in double specimens; (6) Crack propagation in a thermally stressed ring specimen; and development of a two-dimensional code.

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

JACOBS • PHOTONICITY • CRACK • R AND D PROGRAM • COMPUTER PROGRAM • FRAC TURE TOUGHNESS • NRC-9

120792

PRESTON JR • FAIRCHILD RC • ANDERSON BJ

PHYSICAL MECHANICS AND NONDESTRUCTIVE TESTING OF PRESSURE VESSELS ANNUAL PROGRESS REPORT, AUGUST 1, 1975-JULY 31, 1976

UNIV. OF MICHIGAN, ANN ARBOR

A331-3007-2 • 155 PGS. 6 FICS. SEPT. 1977

A SYNTHETIC APERTURE FOCUSING TECHNIQUE FOR ACOUSTIC TESTING IS DESCRIBED. THE TECHNIQUE EMPLOY'S A SINGLE RECEIVED TRANSMITTING OPERATING IN PARSE-CW MODE WITH DIGITAL DATA ACQUISITION AND SYNTHETIC APERTURE POST-PROCESSING TO PROVIDE HIGH LATERAL AND LAXITURAL RESOLUTION. THE EXTENSION OF PREVIOUSLY DEVELOPED ALGORITHMS TO PROVIDE VOLUMETRIC PROCESSING AND DISPLAY IS DESCRIBED. THE DESIGN OF A REFRESHED GREY-SCALE DISPLAY TO PROVIDE INTERACTIVE DISPLAY OF SAP UT DATA IS DESCRIBED.

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

JACOBS • TEST, NONDESTRUCTIVE • ULTRASONICS • PRESSURE VESSELS • INSTUMENT, DIGITAL • NRC-9

120792

LOSS RJ

STRESS, INTENSITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS PROGRESS REPORT ENDING 30 FEBRUARY 1977.

NAVAL RESEARCH LAB., WASHINGTON, D.C.

A331-3003-7 • 53 PGS. 6 FICS. 16 APPS. MAY 1977

Describes research progress in a continuing program to characterize material's properties performance with respect to structural integrity of light water reactor pressure boundary components, program for this reporting period is summarized in the following topics: (1) Radiation embrittlement resistance of 6533-O class steel plate as a function of beta loss, (2) influence of phosphorus and copper content in post-irradiation lower shelf toughness and post-irradiation toughness recovery for 6533-O steel plate and weld metal. (3) Investigations of basic prereact and plastic net fatigue phenomena as a means to mitigate crack propagation in a vessel during a loss of coolant accident. And (4) evaluation of critical factors in crack growth rate studies in a pressurized water reactor environment.

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

R AND D PROGRAM • SAFETY PROGRAM • REACTOR, LWR • PRESSURE VESSELS • STEEL • IMPACT SHOCK • HARD DATA • INTEGRATION TESTING • GROWTH/DEVELOPMENT • CRACK • FATIGUE • NRC-9 • JACOBS

120996

LOSS RJ

STRESS, INTENSITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS - PROGRESS REPORT ENDING 31 MAY 1977

NAVAL RESEARCH LAB., WASHINGTON, D.C.

A331-3005-9 • 49 PGS. 9 FICS. 13 APPS. 27 APPS. SEPT. 1977

This report describes research progress in a continuing program to characterize material's properties performance with respect to structural integrity of light water reactor pressure
ACCESSION NUMBER

AD810003

TITLE

REHEAT CRACKING AT HIGH TEMPERATURES IN ALLOY 600 AND ALLOY 434 MATERIALS

ACQUISITION SOURCE

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AD810003

ABSTRACT

The results of a study of high-temperature reheat cracking in Alloy 600 and Alloy 434 materials are presented. The materials were tested at various temperatures and pressures, and the effects of these variables on crack growth were evaluated. The study revealed that reheat cracking is a significant concern in high-temperature environments and that measures must be taken to mitigate its effects. The results of this study are valuable for the design and maintenance of high-temperature systems.
PAST - J9 CONTINUED

This report contains 97 abstracts from the Nuclear Safety Information Center (NSIC) computer file dated 1/7 through 7/7 containing material properties with respect to the structural integrity. All materials important to the nuclear industry (except concrete) are covered for mechanical properties, chemical properties, composition, fracture of failure, pressure damage, creep, cracking, and swelling. Revised authors, and updated-title indices are included for the convenience of the user.

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

127083

HODGE, ROBERT T., HOLT, R., SMITH, J.

TESTS ON 6-IN. THICK PRESSURE VESSELS. SERIES B: INTERMEDIATE TEST VESSELS V-5 AND V-6 WITH INSIDE NOZZLE

OAK RIDGE NATIONAL LAB. TOWN.

ONLINE/NSIC-180-8, 213 PGS., 20 TABS., 28 FTS., NOV. 1977

Describes failure testing of the 6-inch thick 1957-1977 series B steel pressure vessels. Each containing one flanged nozzle. Vessel V-8 was tested at 99% of 0.7 ED (1100 deg.F) and failed due to leaking without fracture. Vessel V-5 was tested at 22% of 0.3 ED (105 deg.F) and failed by fracture. Material properties measured before the tests were used for fracture and post-test fracture analyses. Test results supported by analysis. Indicate that inside nozzle corner cracks are not subject to flaw strain under pressure loading. The preparation of inside nozzle corner cracks is described in detail. Extensive experimental data are tabulated and plotted.

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

127084

HODSON, J. W.

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

OAK RIDGE NATIONAL LAB. TOWN.

ONLINE/NSIC-180-8, 100 PGS., 32 TABS., 10 FTS., NOV. 1977

Stress-intensity factors measured on photelastic models of two vessels with nozzles are consistent with values from vessel burst tests. Intermediate test vessels V-7B and V-8B were repaired by the half-lead welding technique prescribed in section 28 of the ASME Boiler and Pressure Vessel Code in preparation for testing with flags in the repair zones, and the first two crack arrest model vessels were tested. The fourth thermal shock experiment indicated that linear elastic fracture mechanics analyses is applicable to a loss-of-coolant type of thermal shock. Radiographic examination of fracture surfaces of earlier experiments confirmed the character of the fractures.

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

127085

HODGINS, J. S., BRITSON, J. R.

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

OAK RIDGE NATIONAL LAB. TOWN.

ONLINE/NSIC-180-8, 10 PGS., 7 TABS., 1 FIG., 20 TABS., 20 FTS., OCT. 1977

Short summaries are given for studies conducted and reports published during the quarter including dimensional controls for piping products, combined loading of ASME-10A, and stresses of reinforced nozzles in spherical shell structures. A summary status report is also given for work currently in progress on the finite element analysis of isolated and closely-spaced nozzles in cylindrical pressure vessels. Finally, a summary report is given on the ASME Code and PWR Committee work including the current status of proposed code rules revisions that may be based in part on work conducted under the OPNL DESIGN CRITERIA program.

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

127086

BRITSON, J. S., JOHNSON, W. S., BASS, D. R.

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

OAK RIDGE NATIONAL LAB. TOWN.

ONLINE/NSIC-8, 215 PGS., 20 TABS., 15 FTS., OCT. 1977
LIBRARY CONTINUED

The objective is to summarize and present the more important and relevant results of the parameter study. Three types of models were examined (1) which were generally reinforced by a small fillet (burst) rupture at the junction as specified by the CORI model technique. "Standard" reinforcement is at the last two types. Future work will be to run the orthogonal set of force and moment loadings applied to the branch and free end of the run for these same 25 models.

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

STRESS ANALYSIS - NOZZLE + CLINDO + PRESSURE + INTERNAL + COMPUTER PROGRAM + PRESSURE VESSELS + CODES AND STANDARDS + NRC-9 + ACCESS

134000

106 P. 640X710 + USE 84 + Machine

STRESS ANALYSIS OF CIRCULAR PRESSURE VESSELS WITH CLOSELY SPACED NODAELS BY THE FINITE-ELEMENT METHOD.

VOLUME 6. SERIES STATISTICS OF VESSELS WITH THE CLOUERY SPACED NODES UNDER INTERNAL PRESSURE AND OTHER MATERIAL. Tab.: 4.

1978/19H. 214 + 113 PPS. TABS. PICS. REP. NOV. 1977

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

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

STRESS ANALYSIS + PRESSURE VESSELS + NOZZLE + DESIGN CRITERIA + CODES AND STANDARDS + R + D PROGRAM + PRESSURE + INTERNAL + COMPUTER PROGRAM + NRC-9 + ACCESS

134100

108 P. 640X710 + Machine

PRESSURE VESSEL FRACTURE STRESSES PERTAINING TO A PWR LOCA-ECC THERMAL SHOCK: EXPERIMENTS TE-3 AND TE-9 AND UPDATE OF TE-1 AND TE-8 ANALYSIS.

1978/19H. 22 + 168 PPS. TABS. 15 PICS. MAR. 1977

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

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

REACTOR, PWR + ACCIDENT, LOSS OF COOLANT + PRESSURE VESSELS + FRACTURE TOUGHNESS + CRACK + THERMAL ANALYSIS + GROWTH/DEVELOPMENT + NRC-9

134105

UPDATE OF HEAVY-SECTION STEEL TECHNOLOGY PROGRAM QUARTERLY PROGRESS REPORT FOR APRIL-JUNE 1977.

1978/19H. 39 PPS. TABS. PICS. REP. NOV. 1977

MOST CURRENT STUDIES RELATING TO ALL AREAS OF THE TECHNOLOGY OF THE MATERIALS MANUFACTURED INTO HEAVY-SECTION PRIMARY-COOLANT CONTAINMENT SYSTEMS OF LIGHT-WATER-COOLED NUCLEAR REACTORS. THE PRINCIPAL AREA OF INVESTIGATION IS THE BEHAVIOR AND STRUCTURAL INTEGRITY OF STEEL PRESSURE VESSELS CONTAINING CRACKING PLANTS. CURRENT WORK IS ORGANIZED INTO THE FOLLOWING TASKS: (1) PROGRAM ADMINISTRATION, (2) FRACTURE MECHANICS ANALYSES AND INVESTIGATIONS, (3) EFFECTS OF HIGH-TEMPERATURE PRIMARY REACTIONS RATES ON THE SUBCRITICAL CRACK GROWTH OF HEAVY VESSEL STEEL, (4) INVESTIGATIONS OF IMPACTED MATERIALS, (5) PRESSURE VESSEL INVESTIGATIONS, (6) THERMAL SHOCK INVESTIGATIONS, (7) FOREIGN RESEARCH, AND (8) PRESTRESSED CONCRETE REACTOR VESSEL (PCRV) FABRICATION.

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

UPDATE A + B + C + D PROGRAM + SAFETY PROGRM + FRACTURE TOUGHNESS + HIGH TEMPERATURE + CRACK + GROWTH/DEVELOPMENT + IMPACTED TESTING + PRESSURE VESSELS + STEEL + WELDING + STRUCTURAL INTEGRITY + NRC-9
G. NO - WATER REACTOR SAFETY RESEARCH, SITE SAFETY RESEARCH

UC/UC/HC 4

H. NO. 5

NOVEMBER 1977

U.S. GEOLOGICAL SURVEY, URBANA

U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161


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

6. NO. 6 - PROGRAM - SAFETY PROGRAM - EARTHQUAKE - TECTONICS - SURVEY - TENNESSEE - MISSOURI - MISSISSIPPI - GEOMORPHOLOGICAL CONSIDERATIONS - Site A

12/8/77

MOONSHINE, INC.

P. O. BOX 5115

12. NO. 7 - PROGRAM - SAFETY PROGRAM - EARTHQUAKE - TECTONICS - SITE SURVEY - TENNESSEE - MISSOURI - MISSISSIPPI - GEOMORPHOLOGICAL CONSIDERATIONS - Site B

12/8/77

MOONSHINE, INC.

P. O. BOX 5115

THE RURAL INDUSTRIAL DEVELOPMENT LITERATURE IS UTILIZED TO GAIN INSIGHTS ON THE SOCIOECONOMIC EFFECTS OF NUCLEAR POWER PLANTS. PREVIOUS STUDIES OF LARGE INDUSTRIAL FACILITIES IN SMALL TOWNS HAVE BEEN EXAMINED TO GAIN INSIGHTS ON THE EFFEC

THE CENTRAL MUSEUM OF THE UNITED STATES IS A PART OF THE NORTH AMERICAN CIVILIZATION. EXCEPT FOR MILLION-DEGREE temperatures, it is currently ranked as one of the top fifty plants in the United States. However, recent geological and geographic investigations, particularly seismic studies, have revealed that the central core is structurally complex and is geologically active. This overview summarizes the state of knowledge of the core, including structural, geophysical, and geotechnical features of the central core.

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T. 07 - ADVANCED REACTOR SAFETY RESEARCH: FAST REACTORS

1. 108
SAUL-MILLER, C. RUMBAUGH, AND H. CHAPMAN
PHYSICAL CALCULATIONS FOR THE LANCE HOT NUCLEAR REACTOR
ARGONNE NATIONAL LAB., ILL.


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

2. 106
PHYSICS OF REACTOR SAFETY, QUARTERLY REPORT JANUARY-MARCH 1977
ARGONNE NATIONAL LAB., ILL.

THIS QUARTERLY PROGRESS REPORT SUMMARIZES WORK DONE IN ARGONNE'S APPLIED PHYSICS DIVISION AND COMPONENTS TECHNOLOGY DIVISION FOR THE DIVISION OF REACTOR SAFETY RESEARCH. THE REPORT DESCRIBES THEORETICAL AND EXPERIMENTAL EFFORTS BY MEMBERS OF THE REACTOR SAFETY GROUP TO STUDY THE BEHAVIOR OF THE LANCE REACTOR CORE UNDER VARIOUS OPERATING CONDITIONS. THE REPORT ALSO INCLUDES A SUMMARY OF THE PROGRESS MADE IN THE DEPARTMENT OF NUDE TOOLS RESEARCH.

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

3. 104
PHYSICS OF REACTOR SAFETY, QUARTERLY REPORT APRIL-JUNE 1977
ARGONNE NATIONAL LAB., ILL.

THIS QUARTERLY PROGRESS REPORT SUMMARIZES WORK DONE IN ARGONNE'S APPLIED PHYSICS DIVISION AND COMPONENTS TECHNOLOGY DIVISION FOR THE DIVISION OF REACTOR SAFETY RESEARCH. THE REPORT DESCRIBES THEORETICAL AND EXPERIMENTAL EFFORTS BY MEMBERS OF THE REACTOR SAFETY GROUP TO STUDY THE BEHAVIOR OF THE LANCE REACTOR CORE UNDER VARIOUS OPERATING CONDITIONS. THE REPORT ALSO INCLUDES A SUMMARY OF THE PROGRESS MADE IN THE DEPARTMENT OF NUDE TOOLS RESEARCH.

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

4. 102
numerical results obtained from the three-dimensional transient single-phase version of the coxins computer code
ARGONNE NATIONAL LAB., ILL.


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

5. 100
GIEBEL AND JORDAN: COMPUTER INTEGRATION AND MODELING FOR REACTOR SAFETY: QUARTERLY PROGRESS REPORT FOR JANUARY 1 THROUGH MARCH 31, 1977
BATTLE CREEK LABS., DAE.


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

6. 98
GIEBEL AND JORDAN: COMPUTER INTEGRATION AND MODELING FOR REACTOR SAFETY: QUARTERLY PROGRESS REPORT FOR JUNE 1 THROUGH SEPTEMBER 30, 1977
BATTLE CREEK LABS., DAE.


AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
120063 CONTINUED

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

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

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

120077

GIESEKE JA et al. JORDAN M + LEE H

CHARACTERISTICS OF AGGLOMERATES OF SODIUM OXIDE AEROSOL PARTICLES, DETTELLE COLUMBUS LAB., OHO

BNU-HM-6174 4, 32 PGS. 1 TAB. 10 FIGS. IN REPS. AUG. 1, 1977

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

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

AEROSOL + SAMPLING + AGGLOMERATE + THEORIPHOPHORIES + ANALYTICAL MODEL + REACTOR, LIFER + OXIDE + SODIUM + NRC-7 + JACOBS

120081

GIESEKE JA et al. SCHWARTZ, E + LEE HH

AEROSOL MEASUREMENTS AND MODELING FOR FAST REACTOR SAFETY: QUARTERLY PROGRESS REPORT FOR APRIL 1 THROUGH JUNE 30, 1977, DETTELLE COLUMBUS LAB., OHI0

BNU-HM-6175 4, 24 PGS. 1 TAB. 9 FIGS. 6 IN REPS. OCT. 28, 1977

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

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

B AND D PROGRAM: SAFETY PROGRAM + REACTOR, LIFER + AEROSOL + ANALYTICAL MODEL + COMPUTER PROGRAM + SODIUM + OXIDE + AGGLOMERATE + PARTICLE SIZE DISTRIBUTION + NRC-7

120084

REYNOLDS AJ

REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT, JANUARY 1-MARCH 31, 1977, BROOKHAVEN NATIONAL LAB., UPTON, N.Y.

BNU-HM-60061 4, 216 PGS. MAY 1977

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

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

REACTOR, LIFER + REACTOR, NGR + REACTOR, LMR + COMPUTER PROGRAM, DIGITAL + CODES AND STANDARDS + ACCIDENT, CODE DISRUPTIVE + JACOBS + SAFETY ANALYSIS +ACCIDENT CONSEQUENCES + NRC-4 + NRC-7 + NRC-8

120086

REYNOLDS AJ

REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT APRIL 1-JUNE 30, 1977, BROOKHAVEN NATIONAL LAB., UPTON, N.Y.

BNU-HM-60062 4, 216 PGS. JULY 1977

THE REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT DESCRIBES CURRENT ACTIVITIES AND TECHNICAL PROGRESS IN THE PROGRAMS AT BROOKHAVEN NATIONAL LABORATORY SPONSORED BY THE USNRC REACTOR SAFETY RESEARCH DIVISION. THE PROJECTS REPORTED EACH QUARTER ARE THE FOLLOWING: GAS REACTOR SAFETY EVALUATION, THOR CODE DEVELOPMENT, SIC CODE DEVELOPMENT, LIFER AND LMR SAFETY EXPERIMENTS, FAST REACTOR SAFETY CODE VALIDATION, TECHNICAL COORDINATION OF STRUCTURAL INTEGRITY.
null
A number of important advances were made as part of the Los Alamos Scientific Laboratory's efforts to understand the behavior of fuel particles and coolants in nuclear reactors. This report highlights the experimental and analytical work conducted to improve safety and performance in these critical systems.


**Summary**: The report details recent advancements in understanding the interaction between fuel pellets, coolants, and core catcher during pile experiments. These experiments are crucial for refining analytical models that predict the behavior of fuel particles under high-temperature conditions.

**Key Findings**:
- Improved analytical models for predicting fuel particle behavior under high-temperature conditions.
- Enhanced experimental methods for simulating real-world reactor conditions.
- Enhanced understanding of coolant behavior during core melt scenarios.

**Recommendations**:
- Further research is needed to refine predictive models for different reactor designs.
- Continued collaboration with industry partners to apply these findings in practical scenarios.

**Conclusion**:
The findings from this report will contribute significantly to the development of safer and more efficient nuclear reactor designs. Continued research and development in this area are essential for ensuring the reliability and safety of nuclear power generation technologies.
AEROSOL + AEROPI | PRODUCTION + AEROPO, RADIOACTIVE + REACTOR, LIPPER + BABLE, HYPOBETICAL + ACCIDENT, CORE DISRUPTIVE + CORE MELTDOWN + OUT OF PILE EXPERIMENT + NRC-7

127972

WEIGHT AL. SMITH AN. WESS TS

PUEO AEROSOL, BERNAP TEST (FAST PLAN)

OA RIDE NATIONAL LAB., TNCN.

ORNL/NUREG/TH-129 +. 03 PPS. 2 TABS. 25 FICS. 93 REFS. NOV. 1977


AEROSOL + LIPPER + REACTOR, LIPPER + BABLE + ACCIDENT, HYPOBETICAL + ACCIDENT, CORE DISRUPTIVE + CORE MELTDOWN + ELECTROIC ARC + TRANSPORT + R AND D PROGRAM + FISSION PRODUCT RELEASE + PLUTONIUM + ARGON + NRC-7

123444

STBING AL. Kress TS. PARSIL LP. TOLASAK ML

SOURCE TERM ASSESSMENT FOR LWPB: AEROPO RELEASE AND TRANSPORT WORKS ANALYSIS PROGRAM

OA RIDE NATIONAL LAB., TNCN.

ORNL/NUREG/TH-129 +. 03 PPS. 2 TABS. 25 FICS. 93 REFS. NOV. 1977

THE PROGRAM IS DESIGNED TO INVESTIGATE RADIATION AND TRANSPORT FROM LWPB FOR REACTOR EVENTS OF SEVERITY UP TO AND INCLUDING HYPOTHETICAL CORE-DISRUPTIVE ACCIDENTS (HODAS). DISCUSSIONS INCLUDE 111 PROGRESS IN THE DEVELOPMENT OF THE ORU CAPACITOR DISCHARGE VAPORIZATION (CDV) SYSTEM; 12 RESULTS OF EIGHT TESTS COMPLETED THIS QUARTER IN THE CR1-111 CON FACILITY; 13 RESULTS OF PRELIMINARY TESTS USING THE ARC HEARTH FURNACE FOR USE VAPORIZATION IN CR1-111; 14 RESULTS OF THE FIRST SODIUM FIRE-AEROSOL TEST IN THE DESIGN AND CONSTRUCTION OF THE FUEL-AEROSOL SUGAR TEST (FAT) FACILITY; AND 151 ANALYTICAL STUDIES.

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

AEROSOL + SODIUM + REACTOR, LIPPER + BABLE + ACCIDENT, HYPOTHETICAL + ACCIDENT, CORE DISRUPTIVE + CORE MELTDOWN + OUT OF PILE EXPERIMENT + NRC-7

132464

POTAMA AL. KRESS TS

WIRER REACTOR AEROPO RELEASE AND TRANSPORT PROGRAM QUARTERLY PROGRESS REPORT FOR APRIL-JUNE 1977

OA RIDE NATIONAL LAB., TNCN.

ORNL/NUREG/TH-129 +. 03 PPS. 2 TABS. 25 FICS. 93 REFS. NOV. 1977

THE PROGRAM IS DESIGNED TO INVESTIGATE RADIATION AND TRANSPORT FROM LWPB FOR REACTOR EVENTS OF SEVERITY UP TO AND INCLUDING HYPOTHETICAL CORE-DISRUPTIVE ACCIDENTS (HODAS). DISCUSSIONS INCLUDE 111 PROGRESS IN THE DEVELOPMENT OF THE ORU CAPACITOR DISCHARGE VAPORIZATION (CDV) SYSTEM; 12 RESULTS OF EIGHT TESTS COMPLETED THIS QUARTER IN THE CR1-111 CON FACILITY; 13 RESULTS OF PRELIMINARY TESTS USING THE ARC HEARTH FURNACE FOR USE VAPORIZATION IN CR1-111; 14 RESULTS OF THE FIRST SODIUM FIRE-AEROSOL TEST IN THE DESIGN AND CONSTRUCTION OF THE FUEL-AEROSOL SUGAR TEST (FAT) FACILITY; AND 151 ANALYTICAL STUDIES.

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

AEROSOL + NUCLEOT + REACTOR, LIPPER + BABLE + ACCIDENT, HYPOBETICAL + ACCIDENT, CORE DISRUPTIVE + CORE MELTDOWN + COMPUTER PROGRAM + TRANSPORT + NRC-7

127336

ANNUAL CORE PULSE REACTOR JUNIIZE QUARTER PROGRESS REPORT. OCTOBER-DECEMBER 1976
THE PROGRAM IS UNDERWAY TO SIMULATE ACCIDENTAL TRANSIENT CONDITIONS IN THE LIQUID TO PROVIDE THE REQUIRED DATA BASE TO UNDERSTAND THE CONTROLLING ACCIDENT SEQUENCES AND TO DETERMINE THE EFFECTIVE MEASURES OF THE COOLING COMPUTER SIMULATION MODELS AND CODES USED IN ACCIDENT ANALYSIS AND LICENSING REVIEW. IT IS INVESTIGATING SEVEN MAJOR AREAS OF INTEREST AS FOLLOWS: 1) ACCIDENT RESPONSE, 2) CORE DAMAGE, 3) FUEL CORROSION, 4) SULFUR CONSIDERATION AND STRUCTURAL INTEGRITY, 5) RESEARCH OF ELEVATED TEMPERATURE DESIGN CRITERIA, 6) FUEL MOTION, 7) ACET FUEL MOTION SYSTEM, AND 8) ADVANCED REACTOR SAFETY RESEARCH ASSESSMENT.

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

A AND B PROGRAM - SAFETY PROGRAM - REACTOR, LIQUID - ACCIDENT ANALYSIS - REACTOR, FAST - ACCIDENT, CORE DISRUPTIVE - EXCITATION, LARGE - LIQUID - MOLTEN FUEL - NRC-7

123456
CARPENTER, R., SHERMAN, C., REYNOLDS R.
CONDENSATION OF FUEL UPTO THE HOOD-CORE STRUCTURE DURING AN LIQUID CORE-DISRUPTIVE ACCIDENT
UNIT. OF CHARLOTTESVILLE
NAMC-5963, 8 PAGES, 8 TABS. 16 Figs. 17 Tabs. OCT. 1977

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

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

REACTOR, LIQUID - ACCIDENT ANALYSIS - ACCIDENT, CORE DISRUPTIVE - CONSIDERATION, MAJOR EXCHANGE - COOLING SYSTEM - FUEL MELT - MOLTEN, RADIOACTIVE - MASS BALANCE PRODUCT TRANSPORT - JACOBS, NRC-7

123456
ARNOY, J., TURCOTTE, M., BERRY, M.
PARTICLE SIZE DISTRIBUTION FROM HOMOGENOUS NUCLEATION CONDENSATION AND PARTICLE GROWTH
UNIT. OF CHARLOTTESVILLE
NAMC-6595, 4 PAGES, 6 TABS. 16 Figs. OCT. 1977

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

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

A AND B PROGRAM - SAFETY PROGRAM - REACTOR, LIQUID - AEROSOL - CONSIDERATION, SCALE - PARTICLE SIZE DISTRIBUTION - JARVINEN, NRC-7

123456
S. 86 - ADVANCED REACTOR SAFETY RESEARCH, GAS-COOLED REACTORS

IPLSR

REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT, JANUARY 1-MARCH 31, 1977

NATIONAL LABORATORY, D. E. WASHINGTON M. A.

THE PROJECTS REPORTED EACH QUARTER ARE THE FOLLOWING: GAS REACTOR SAFETY EVALUATION, HOM REACTOR SAFETY RESEARCH PROGRAMS, HET REACTOR SAFETY RESEARCH PROGRAMS, DATE DISRUPTIVE, AND HET REACTOR SAFETY RELIABILITY ASSESSMENT.

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

IPLSS0

REACTOR SAFETY RESEARCH PROGRAMS QUARTERLY PROGRESS REPORT, APRIL 1-JUNE 30, 1977

NATIONAL LABORATORY, D. E. WASHINGTON M. A.

THE PROJECTS REPORTED EACH QUARTER ARE THE FOLLOWING: GAS REACTOR SAFETY EVALUATION, HOM REACTOR SAFETY RESEARCH PROGRAMS, HET REACTOR SAFETY RESEARCH PROGRAMS, DATE DISRUPTIVE, AND HET REACTOR SAFETY RELIABILITY ASSESSMENT.

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

IPLS90

PROCEEDINGS OF THE JAPAN-U.S. SEMINAR ON HET REACTOR SAFETY TECHNOLOGY, SEISMEIC EVENTS, VOLUME I

NATIONAL COMMISSION FOR REGULATORY COMMISSION, WASHINGTON M. A.

SECOND DAY PAPERS (9TH-11TH APRIL 1977) AND THIRD DAY PAPERS (12TH APRIL 1977, MEETING HELD AT WASHINGTON, D. C.)

AVAILABILITY - CONTACT BUNKERHILL NATIONAL LABORATORY, TECHNICAL REPORTS DISTRIBUTION, D. E. WASHINGTON, M. A. 1970 FOR IPRS INFORMATION

IPLS91

PROCEEDINGS OF THE JAPAN-U.S. SEMINAR ON HET REACTOR SAFETY TECHNOLOGY, VOLCANNIC AND HET REACTOR SAFETY TECHNOLOGY, VOLUME II

NATIONAL COMMISSION FOR REGULATORY COMMISSION, WASHINGTON M. A.

SECOND DAY PAPERS (9TH-11TH APRIL 1977) AND THIRD DAY PAPERS (12TH APRIL 1977, MEETING HELD AT WASHINGTON, D. C.)

AVAILABILITY - CONTACT BUNKERHILL NATIONAL LABORATORY, TECHNICAL REPORTS DISTRIBUTION, D. E. WASHINGTON, M. A. 1970 FOR IPRS INFORMATION

IPLS92

THESE THREE PAPERS PRESENTED AT THE SEMINAR ON HET REACTOR SAFETY TECHNOLOGY, HIGH TEMPERATURE MATERIALS, HEAT TRANSFER, AND REACTOR SAFETY ISSUES.
AND DYNAMIC TEMPERATURE RESPONSE TO FAST NEUTRON CHAIN REACTION.

AND DYNAMIC TEMPERATURE RESPONSE TO A MAJOR CHAIN REACTION.

AND DYNAMIC TEMPERATURE RESPONSE TO A MAJOR CHAIN REACTION.

AND DYNAMIC TEMPERATURE RESPONSE TO A MAJOR CHAIN REACTION.

AND DYNAMIC TEMPERATURE RESPONSE TO A MAJOR CHAIN REACTION.

AND DYNAMIC TEMPERATURE RESPONSE TO A MAJOR CHAIN REACTION.

AND DYNAMIC TEMPERATURE RESPONSE TO A MAJOR CHAIN REACTION.


9. HEALTH SAFETY RESEARCH

LISTED:

NUCLEAR DECAY DATA FOR RADIOACTIVE NUCLEIDES OCCURRING IN ROUTINE RELEASES FROM NUCLEAR FUEL CYCLE FACILITIES

CIA RINGE NATIONAL LAB., TREN.

BIBLIOGRAPHY 1961-1970, 110 APS; 8 REPS, AUG., 1977

TABLE: THE ATOMIC AND NUCLEAR RADIATIONS EMITTED BY 200 RADIOACTIVE NUCLEIDES. FOR EACH RADIOACTIVE NUCLEIDE IS GIVEN THE HALF-LIFE AND RECOMMENDED VALUES FOR THE ENERGIES, INTENSITIES AND RADIOBIOLOGICAL ABSORBED-DOSE CONSTANTS FOR EACH OF THE ATOMIC AND NUCLEAR RADIATIONS. ALSO GIVEN ARE THE DAUGHTER RADIOACTIVE NUCLEIDES PRODUCED AND RECOMMENDED VALUES FOR DECAY BRANCHING RATIOS, WHERE APPLICABLE. THE RADIOACTIVE DECAY CHAINS AND BRANCHING RATES ARE DISPLAYED IN DIAGRAM FORM.

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

EPILOGUE: RADIOACTIVITY RELEASE + AIRBORNE RELEASE + DOSE + ENERGY + RADIOISOTOPE + ALPHA Emitter + BETA Emitter + GAMMA Emitter = RADIO-
AN EVALUATION OF ENVIRONMENTAL DATA RELATING TO SELECTED NUCLEAR POWER PLANT SITES: AN ALMAMITIC POWER PLANT SITE

AMERICAN NATIONAL, I.L.
ANALYSIS 3. 10 PPS. 3 Figs. 1 Rep. Aug. 1976

ENVIRONMENTAL MONITORING DATA FOR THE YEARS 1973, 1974 AND 1975 PERTAINING TO THE ANLAMITIC NUCLEAR POWER PLANT, WHICH BEGAN OPERATION IN EARLY 1976, WERE ANALYZED BY THE MOST PRACTICAL, QUALITATIVE AND QUANTITATIVE METHODS. THE RESULTS SHOWED NO SIGNIFICANT IMMEDIATE DELETERIOUS EFFECTS. THE PRESENT INDICATIONS DO NOT LEAD TO A CONCERNED PREDICTION THAT ANY ARE DEVELOPING.

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

E AND D PROGRAM • SAFETY PROGRAM • MONITORING PROGRAM. ENVIRONMENTAL + SITE EFFECT + REACTOR, NUR + ANLAMITIC

199999
AN EVALUATION OF ENVIRONMENTAL DATA RELATING TO SELECTED NUCLEAR POWER PLANT SITES

AMERICAN NATIONAL, I.L.
ANALYSIS 3. 10 PPS. 3 Figs. 1 Rep. Aug. 1976

ENVIRONMENTAL MONITORING DATA FOR THE YEARS 1973 AND 1974 PERTAINING TO THE THREE MILE ISLAND NUCLEAR POWER PLANT SITE 1, WHICH BEGAN OPERATION IN EARLY 1975, WERE ANALYZED BY THE MOST PRACTICAL, QUALITATIVE AND QUANTITATIVE METHODS. THE RESULTS SHOWED NO IMMEDIATE DELETERIOUS EFFECTS DUE TO PLANT OPERATIONS. THE PRESENT INDICATIONS DO NOT LEAD TO A CONCERNED PREDICTION THAT ANY ARE DEVELOPING.

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

ENVIRONMENT + THREE MILE ISLAND. 1 (NUR) + ANLAMITIC 

100000
AN EVALUATION OF ENVIRONMENTAL DATA RELATING TO SELECTED NUCLEAR POWER PLANT SITES
Environmental monitoring data for the years 1962 through 1975 pertaining to the Pajarito Station, which began commercial operation in 1964, were analyzed by the best practical qualitative and quantitative methods. Thermal plume, water chemistry, biotic, and water quality data were evaluated and the results are presented in this report. The results showed no significant aquatic deleterious effects due to commercial operation. The present indications do not lead to a concluded prediction that any are developing.


129980 Evaluation of environmental data relating to selected nuclear power plant sites - the Prairie Island Nuclear Plant Site

Argonne National Lab., Ill.

ANL-75-6, 11 Pgs., 11 Figs. Nov. 1975

Enviornmental data for 1974-1975 pertaining to the Prairie Island Nuclear Generating Station (which began commercial operation in December 1968) were analyzed. Evaluations of aquatic and terrestrial biotic data are presented. The data indicate no significant immediate deleterious effects on the biota from plant operation. This conclusion, in general, extends the predictions of the previous analysis. Although the station has not operated long enough to reveal long-term deleterious effects, present indications do not lead to a concluded prediction that any are developing.


129988 Monitoring, program, environmental data collection - Prairie Island; Plattsburgh; Narragansett; Aquatic - Biota; Site; Effect; VAC-18

130956 Monitoring - an evaluation of environmental data relating to selected nuclear power plant sites - the Nine Mile Point Nuclear Power Station Site Unit 1

Argonne National Lab., Ill.

ANL-77-7, 12 Pgs., 1 Figs. Oct. 1976

Fish improvement monitoring data for the years 1974 and 1975 pertaining to the Nine Mile Point Unit 1, which began commercial operation in December 1968, were analyzed. The results indicate no significant aquatic deleterious effects on the biota due to plant operation. The data indicate no significant immediate deleterious effects on the biota from plant operation. However, no conclusion extending the predictions of the previous analysis is possible. The results showed that predictions were higher than those observed.


130708 Monitoring - monitoring program - aquatic - biota - ecosystem - evaluation of environmental data relating to selected nuclear power plant sites

Argonne National Lab., Ill.

ANL-78-8, 36 Pgs., 2 Figs. Nov. 1978

Analyses of the field data gathered during the monitoring program at seven nuclear power plant sites showed no significant ecological impacts. This conclusion is within the constraints of the quality of available data from these sites. Current monitoring programs are, however, not designed to meet the needs for statistical analysis and, consequently, the monitoring data are often ill-suited for modern statistical procedures. Recommendations are suggested for revising monitoring schemes so that more precise conclusions can be made from fewer field measurements.


Environment - power plant - nuclear - monitoring program - environmental data collection - aquatic - biota - ecosystem - evaluation of environmental data relating to selected nuclear power plant sites

132480 Adams, De, Cunningham, P., Gray, D.

A critical evaluation of the hydrological environmental technical specifications. Vol. 1 - Program
A comprehensive study of the data collected as part of the environmental technical specifications program for eight nuclear power plants was conducted. This report includes a summary of the screening phase in which the adequacy of the hydrothermal and ecological monitoring data for each plant were reviewed, and the summary and recommendations resulting from a detailed examination of the three nuclear power plants (Surry, Peach Bottom and Sam O'Keefe) selected in the initial screening.


R and D Program - Environment + Technical Specifications, Environmental + Power Plant, Nuclear + Thermal Pollution + Ecosystem + Monitoring, Environmental - Preview + Test. Operational + Biological Control

A comprehensive study of the data collected as part of the environmental technical specifications program for Units 1 and 2 of the Surry nuclear power plant was carried out. The program included an analysis of the hydrothermal and ecological monitoring data collected from 1973 through 1974. The hydrothermal analysis includes a discussion of models used in plume predictions prior to plant operation and an evaluation of the present hydraulic monitoring program. The ecological analysis includes validation of impacts predicted in the final environmental statement using the operational monitoring data. Specific recommendations are made for improving both the present hydrothermal and ecological monitoring programs.


A comprehensive study of data collected as part of the environmental technical specifications program for Peach Bottom Unit 3 was conducted for HEC. The program included an analysis of both the hydrothermal and ecological monitoring data collected from 1973 through 1974. The hydrothermal analysis includes a discussion of models used in plume predictions prior to plant operation and an evaluation of the present hydraulic monitoring program. The ecological analysis includes validation of impacts predicted in the final environmental statement using the operational monitoring data. Specific recommendations are made for improving both the present hydrothermal and ecological monitoring programs.


A critical evaluation of the nonradiological environmental technical specifications Vol. 2 - Surry Power Plant Units 1 and 2.


A comprehensive study of data collected as part of the environmental technical specifications program for Peach Bottom Unit 2 was conducted for HEC. The program included an analysis of both the hydrothermal and ecological monitoring data collected from 1973 through 1974. The hydrothermal analysis includes a discussion of models used in plume predictions prior to plant operation and an evaluation of the present hydraulic monitoring program. The ecological analysis includes validation of impacts predicted in the final environmental statement using the operational monitoring data. Specific recommendations are made for improving both the present hydrothermal and ecological monitoring programs.


A critical evaluation of the nonradiological environmental technical specifications Vol. 3 - Peach Bottom Atomic Power Station Units 2 and 3.


Development of a unified transport approach for the assessment of power-plant impact.


Progress during the first 18 months in implementation of the unified transport approach (UTA) is summarized in this report. Which covers the period through Dec. 1976. The goal of this project is to develop mathematical models for past-transport, one- and two-dimensional transport of thermal, radiological, chemical, and biological properties in rivers, estuaries, lakes, and coastal regions for assessing the impact of power-plant operations. The UTA is designed as a basis for calculating the transport of radioactive materials that depend on basic flow properties, which can be obtained from a common set of data for geometry, bathymetry, and meteorology that must be prepared only once.

SHOCK AND VIBRATION ENVIRONMENTS FOR LARGE SHIPPING CONTAINER DURING TRUCK TRANSPORT (PART II)
SANDIA LABS, ALBUQUERQUE, N.M.
SAND77-1110 * 9 PGS. 2 TABS. 6 FIGS. 1 REP. SEPT. 1977

CURRENTLY AVAILABLE DATA WERE TAKEN ON TRUCKS CARRYING LIGHTER LOADS THAN THE LOADS OF CURRENT INTEREST. IN ADDITION, THE DATA ARE EXPECTED TO BE USEFUL IN THE DETERMINATION OF ANY TRENDS OF VIBRATION AND SHOCK ENVIRONMENTS WITH INCREASED CARGO HEIGHT. THESE DATA WERE OBTAINED ON A "PIGBACK" BASIS DURING TRUCK TRANSPORT OF 195 TONS (434,000 LB) CARGO WHICH CONSISTED OF A SPENT FUEL CONTAINER AND ITS SUPPORTING STRUCTURE FROM MERCURY, NEVADA, TO ALBUQUERQUE, NEW MEXICO.

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

R AND D PROGRAM - SAFETY PROGRAM - SHIPPING CONTAINER - TRUCK - TRUCK SHOCK - VIBRATION - NRC-12
12. 05 — SAFEGUARDS RESEARCH

122810
Brundage SB, Perry BB
A MICROFOTOCOMPACT TECHNIQUE FOR RAPID INVENTORY OF PLUTONIUM-CONTAINING FAST CRITICAL ASSEMBLY FUEL MARYLAND NUCLEAR LAB., ILL.
AMC-77-67 +, 28 PPS. 9 FIGS. OCT. 1977

A HOE-MODIFIED AUTOMOTIVE TECHNIQUE IS DESCRIBED WHICH CAN PROVIDE A VERIFICATION OF THE PIECE COUNT AND THE PLUTONIUM CONTENT OF PLUTONIUM-CONTAINING FUEL ELEMENTS. THIS TECHNIQUE USES THE ELECTRONICALLY ENHANCED IMAGE RATES FROM PLUTONIUM TO FORM IMAGES OF FUEL ELEMENTS ON PHOTOGRAPHIC FILM. AUTOMOTIVE GAS AN ADVANTAGE OF PROVIDING AN INVENTORY VERIFICATION WITHOUT THE OPENING OF CONTAINERS OR THE HANDLING OF FUEL ELEMENTS. MISSING FUEL ELEMENTS, PARTITION OF NONDESTRUCTIVE MATERIAL, AND SUBSTITUTION OF ELEMENTS OF DIFFERENT SIZE ARE DETECTABLE. RESULTS ARE PRESENTED FOR FUEL ELEMENTS IN VARIOUS STORAGE CONFIGURATIONS AND FOR FUEL ELEMENTS CONTAINED IN A FAST CRITICAL ASSEMBLY.

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

MEASUREMENT — PLUTONIUM — RADIOPHOTOGRAPHY — TEST, NONDESTRUCTIVE — FUEL STORAGE — SAFEGUARDS, NUCLEAR MATERIAL — FUEL ELEMENTS

123067
Cole PJ, Bennett CA, Dik PLC, Wood HT, Brown PJ, Hovington PP
STRUCTURE AND DRAFTING OF SAFEGUARDS REGULARITY DOCUMENTS MEDITERRANEAN PACIFIC NORTHWEST LABS., RICHLAND, WASH.
M-242-77 +, 196 PPS., FIGS., REPS., NOV. 1977

DEVELOPED HYPOTHESIS ON THE RELATIONSHIP BETWEEN THE STRUCTURE AND DRAFTING OF SAFEGUARDS REGULARITY DOCUMENTS AND THE ABILITY OF THE DOCUMENT USERS TO UNDERSTAND AND IMPLEMENT THEM IN A WAY THAT REFLECTS THE INTENT AND REQUIREMENTS OF THE REGULATORY OFFICES, LICENSORS, INSPECTORS, AND THE GENERAL PUBLIC MUST UNDERSTAND THE AGREEMENT'S REQUIREMENTS IF THE REGULATORY SYSTEM IS TO FUNCTION EFFECTIVELY AND IN COMPLIANCE WITH LEGAL REQUIREMENTS. A SERIES OF DECISIONS THAT WILL BE REQUIRED TO IMPROVE COMMUNICATIONS WITH LICENSORS WERE SET FORTH.

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

A AND D PROGRAM, SAFETY PROGRAM, SAFEGUARDS, NUCLEAR MATERIAL, AGENT, NRC, PROCEDURES AND MANUALS, COMMUNICATION — NRC-13

127224
DMS D
A SMALL-SCALE ENGAGEMENT MODEL WITH ARRIVALS: ANALYTICAL SOLUTIONS SANDS LAB., ALBUQUERQUE, N.M.
SAND-495 +, NRC-2226 +, 25 PPS. 2 TABLES, 8 FIGS., APRIL 1977

THIS REPORT PRESENTS AN ANALYTICAL MODEL OF SMALL-SCALE BATTLES. THE IMPACT WAS PROVIDED BY A NUCLEAR FACILITY AND THEIR POTENTIAL ADVERSARIES. THE SOLUTION PROCEDURE CAN BE USED TO FIND MEASURES OF A NUMBER OF CRITICAL PARAMETERS. FOR EXAMPLE, THE HM PROBABILITIES AND THE EXPECTED DURATION OF THE BATTLE. MODEL SPECIFY THE TOTAL NUMBER OF INDIVIDUAL CONVOLVES ON THE Opposing SITES IS LESS THAN 10. FOR SMALLER FORCE SIZE BATTLES, WITH ONE OR TWO CONVOLVES ON EACH SIDE, Symbolic SOLUTIONS CAN BE FOUND.

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

SAFEGUARDS, NUCLEAR MATERIAL, SYSTEM ANALYSIS, COMPUTER PROGRAM, MODEL, SENSITIVITY ANALYSES, NRC-13, THEFT/DIVERSION, NUCLEAR

132811
ENG D
NUCLEAR FACILITY SAFEGUARD SYSTEMS MODELING USING DISCRETE EVENT SIMULATION SANDS LAB., ALBUQUERQUE, N.M.
SAND-77-1507 +, CONF-770498 +, 5 PPS., FROM 5TH ANNUAL CONFERENCE ON MODELING & SIMULATION; PITTSBURGH, PA., APRIL 28, 1977

THE THEFT OR DISPERSSION OF SPECIAL NUCLEAR MATERIAL AT A NUCLEAR FACILITY IS TREATING THE STUDYING THE TEMPORAL RELATIONSHIPS BETWEEN ADVERSARIES HAVING AUTHORIZED ACCESS TO THE FACILITY (EXEMPTS) AND SAFEGUARD SYSTEMS EVENTS BY USING A GAS IN DISCRETE EVENT SIMULATION. THE SAFEGUARD SYSTEM EVENTS—DETECTION, ASSESSMENT, DELAY, COMMUNICATIONS, NEUTRALIZATION—ARE MODELED FOR THE GENERAL INSIDER ADVERSARY STRATEGY, WHICH INCLUDES DESTRUCTION OF THE SAFEGUARD SYSTEM ELEMENTS FOLLOWED BY AN ATTEMPT TO STEAL OR DISPERSE SPECIAL NUCLEAR MATERIAL. THE PERFORMANCE MEASURE USED IN THE ANALYSIS IS THE PROBABILITY OF SAFEGUARD SYSTEM SUCCESS IN COMBINER ADVERSE BASED UPON A PREDETERMINED SET OF ADVERSE ACTIONS.

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

A AND D PROGRAM, SAFETY PROGRAM, SAFEGUARDS, NUCLEAR MATERIAL, THEFT/DIVERSION, MODEL, SIMULATION, COMPUTER PROGRAM, NRC-13
A major mission of safeguards is to protect against the use of nuclear materials by adversaries to harm society. A hierarchical structure of safeguards responsibilities and activities to assist in this mission is defined. The structure begins with the definition of international, multi-national safeguards and continues through domestic, regional, and facility safeguards. The facility safeguards are decomposed into physical protection and material control responsibilities. In addition, in-transit safeguards systems are considered.


9 and D program, safeguards, nuclear material, transportation and handling, theft/diversion: MHC-13 + Jacobs
THE INSURER SAFEGUARDS EFFECTIVENESS MODEL (ISEM) IS A STOCHASTIC, DISCRETE EVENT, MONTE-CARLO SIMULATION MODEL USED TO ASSESS THE EFFECTIVENESS OF PHYSICAL PROTECTION SYSTEMS FOR FACILITIES WHICH STORE, PROCESS, OR USE DUAL-USE ENRICHED U-235, WHETHER EMPLOYEES HAVE AUTHORIZED ACCESS TO THE FACILITY IN THE FACILITY'S SAFEGUARDS SYSTEM. THE EFFECTIVENESS OF THE SAFEGUARDS SYSTEM EFFECTIVENESS TO A VARIETY OF GUARD TACTICS IS EXPLAINED IN THIS PAPER. THE EVOLUTION OF COMPREHENSIVE GUARD TACTICS FOR PROTECTING A HYPOTHETICAL FACILITY IS DEMONSTRATED, ATTENTION IS FOCUSED ON THE POTENTIAL THREAT MODES OF INTELLIGENCE.

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

A AND B PROGRAM • SAFETY PROGRAM • ENTRY • HUMAN FACTORS • MONTE-CARLO • MC-13 • SAFEGUARDS, NUCLEAR MATERIAL • THEFT/DIVERSION

PHYSICAL PROTECTION OF NUCLEAR FACILITIES QUARTERLY PROGRESS REPORT JANUARY-MARCH 1977
SANDE LA BC • ALKMAN, B. H.
SANDB-77-0009 4. 12 PP. 6 TABS. 3 FIG. 6 REP. JULY 1977

THE PHYSICAL PROTECTION OF NUCLEAR FACILITIES PROGRAM CONSISTS OF FOUR MAJOR AREAS - EVALUATION METHODOLOGY, DEVELOPMENT, PATH GENERATION/SELECTION METHODOLOGY, FACILITY CHARACTERIZATION, AND COMPONENT FUNCTIONAL PERFORMANCE CHARACTERIZATION. ACTIVITIES IN EACH OF THESE AREAS ARE SUMMARIZED.

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

A AND B PROGRAM • SAFETY PROGRAM • ENTRY • HUMAN FACTORS • MONTE-CARLO • MC-13 • SAFEGUARDS, NUCLEAR MATERIAL • THEFT/DIVERSION

THE DOCUMENT EXPLAINS HOW TO CONSTRUCT A SAFEGUARD GRAPH WHICH MODELS THE FIXED-SITE FACILITY AND HOW TO USE THE GRAPH TO FIND THE SAFEST PATHS IN THE GRAPH. THE SAFEST PATHS ARE THOSE WHICH WOULD ALLOW AN ADVERSARY TO TAKE ADVANTAGE OF THE GREATEST WEAKNESSES IN THE SYSTEM OF BARIERS AND ALARMS. THE SAFEGUARD GRAPH IS A Tool WITH WHICH SAFEGUARDS DESIGNERS AND ANALYSTS CAN STUDY THE RELATIVE EFFECTS OF DESIGN CHANGES ON THE ADVERSE ROUTING PROBLEM. IN ADDITION TO SHOWING HOW TO USE SPATHS, THIS REPORT DESCRIBES THE METHODS USED TO FIND SAFEST PATHS AND SEVERAL IMPLEMENTATION DETAILS WHICH CAUSE SPATHS TO BE EXTREMELY EFFICIENT.

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

A AND B PROGRAM • SAFETY PROGRAM • ENTRY • HUMAN FACTORS • MONTE-CARLO • MC-13 • SAFEGUARDS, NUCLEAR MATERIAL • THEFT/DIVERSION

THE EVALUATION OF SAFEGUARDS SYSTEMS FOR NUCLEAR MATERIALS IN TRANSIT - THE DEVELOPMENT OF THE PROGRAM PLAN
SANDE LA BC • ALKMAN, B. H.
SANDB-77-0004 4. 12 PP. 13 TABS. 3 FIGS. JULY 1977

THIS REPORT IS BASED UPON PRESENTATIONS GIVEN TO THE MCC ADMINISTRATION AND STAFF AND TO SANDER LA BC LABORATORIES, LIVERMORE AND ALKMAN, STAFF IN FEBRUARY 1977. THE PURPOSE OF THESE PRESENTATIONS WAS TO DESCRIBE THE PROGRAM PLAN FOR THE PHYSICAL PROTECTION OF NUCLEAR MATERIAL IN TRANSIT AND TO PRESENT HIGHLIGHTS OF THE CURRENT STATUS FROM SEVERAL OF THE SECURITY STUDY AREAS.

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

A AND B PROGRAM • SAFETY PROGRAM • ENTRY • HUMAN FACTORS • MONTE-CARLO • MC-13 • SAFEGUARDS, NUCLEAR MATERIAL • SYSTEM ANALYSIS • PROTECTION SYSTEM • SECURITY

THE OPTIMUM IMPROVEMENT OF GRAPHS RELATED TO NUCLEAR SAFEGUARDS PROBLEMS
SANDE LA BC • ALKMAN, B. H.
SANDB-77-0030 4. 12 PP. 1 TAB. 3 FIGS. 10 OCT. 1977

DEVELOPS THE METHODOLOGY FOR OPTIMALLY IMPROVING GRAPHS RELATED TO NUCLEAR SAFEGUARDS ISSUES. IN PARTICULAR, GIVEN A FIXED NUMBER OF DOLLARS, THE REPORT PROVIDES A METHOD FOR OPTIMALLY ALLOCATING SUCH DOLLARS OVER THE AREA OF A WEIGHTED GRAPH. THE WEIGHTS VARY AS A FUNCTION OF...
continued

1.3.2.12

Dollars spent on R&D to date to improve the system effectiveness measure which is the shortest of all shortest paths to several targets, are heights can be either clock times or detection probabilities, and the algorithm does not explicitly consider all paths to the targets.


A and U Program - Safety Program - Safeguards, Nuclear Material - Model - Optimization - Probability - NRC-13

122361

Name: conbridge db

SIMP: A subroutine for the 4 shortest paths in a sartothe graph

Simp, Los Alamos National Laboratory, N.M.

SAND-77-1165, 28 APS, AUG. 1977

Finding shortest paths in a weighted graph model is one way for a safeguards analyst to locate weaknesses in a facility's barrier and alarm system. Depth can be used to rank shortest paths according to path length so that the 4 shortest paths can be studied by more detailed methods of attack which possess additional properties attractive to an adversary. While emphasizing how to use depth, this report explains the 4 shortest path algorithm of Hoffmann, Polley and Boylew and contains a sample problem, test results and program listings.


122355

Number of

A survey of threat studies related to the nuclear power industry

Simp, Los Alamos National Laboratory, N.M.

SAND-77-2254, 97 APS, 2 figs, 5 figs, 27 EDPs, AUG. 1977

A considerable effort has been directed toward the determination of threat characteristics, resulting in a voluminous collection of documents. This report summarizes several of the major studies in order to make the information more accessible. This summary includes only studies involving attacks on nuclear material; plus those incidents which because of their objectives, vulnerabilities, or motivations may lend insight into potential threat against nuclear facilities or material.


Subtopic - Theft/Diversion - Safeguards, Nuclear Material - NRC-13 - Jacobs

122432

Jacobs, SB

A survey of threat studies related to nuclear safeguards problems

Simp, Los Alamos National Laboratory, N.M.

SAND-77-2321, 20 APS, 1 tab, 1 fig, 3 EDPs, OCT. 1977

Develops the methodology for optimally improving graphs related to nuclear safeguards issues. For example, given a fixed number of dollars, the report provides a method for optimally allocating such dollars over the arcs of a weighted graph. The weights vary as a function of the arc flow on arcs so as to improve the system effectiveness measure which is the shortest of all shortest paths to several targets. Are heights can be either clock times or detection probabilities, and the algorithm does not explicitly consider all paths to the targets.


Subtopic - Theft/Diversion - Safeguards, Nuclear Material - Model - Optimization - Probability - NRC-13

122434

Name ky trifalk

Tentative job analysis for a high-level, fixed-site, nuclear security officer

Simp, Los Alamos National Laboratory, N.M.

SAND-77-1760, 32 APS, 3 EDPs, OCT. 1977

A tentative job analysis for a high-level, fixed-site, nuclear security officer is presented. The report focuses on the multiplicity of the job to provide a framework for evaluating the function of a security officer in physical protection systems. Several job requirements related to duties, bases, skills, personal contacts, supervision, working conditions, and decision making are presented. Individual character traits desirable in security officers are described.


The NRC Office of Nuclear Regulatory Research contracted with the System Development Corporation (SDC) to develop integrated system concepts for the safeguard of special nuclear materials against malicious action during interfacility transport. The study was divided into three major tasks: the development of adversary action sequences, the assessment of the vulnerability of the transport of nuclear materials to adversary action, and the development of conceptual safeguard system design requirements to reduce vulnerabilities.


*SAFEGUARDS, NUCLEAR MATERIAL DESIGN CRITERIA, SPECIAL NUCLEAR MATERIAL TRANSPORTATION AND HANDLING * JACOBS * NRC-13

The NRC Office of Nuclear Regulatory Research commissioned a project to develop integrated system concepts for the safeguard of nuclear materials against malicious action during interfacility transport. The conduct and findings of the project are presented, potential threats by terrorists and others to interfacial 3 nuclear materials in transit are addressed, and measures which can be taken to reduce both the likelihood of such threats and the probability of success if carried out are recommended.


*SAFEGUARDS, NUCLEAR MATERIAL DESIGN CRITERIA, SPECIAL NUCLEAR MATERIAL TRANSPORTATION AND HANDLING * JACOBS * NRC-13
The experiments are concerned with the critical separation between water flooded subcritical clusters of fuel rods in the presence of various fixed neutron poisons. The experiments are carried out in a 1.0 x 1.0 x 2.1 meter deep tank provided with features specifically designed and built for these experiments. The initial series of experiments in this program are covered in this report and involve aluminum clad 2.15% 235U enriched U235 rods about 12 mm in diameter by 91.4 cm in length. The critical separation between three subcritical clusters of these rods aligned in a row has determined with and without several neutron absorber materials.


R and D Program + Safety Program + Criticality Experiment + Fuel Rods + Shipping Container + Uranium Oxide + Poison. Fixed - HNC-12 - NEC-RC

130543

Dickinson & Guyard: No. 701 E. Tuck Rd.

Criticality measurements on plastic-reflected arrays of urania nitrate solution cylinders and similar measurements in a single nickel-reflected 50-cm-diameter tank were performed.

The fabrication of equipment for the experiments is nearly complete, and preliminary testing is underway. About 80% of the oxide has been pressed into briquettes.


Jacobs + Criticality Experiment + Criticality Safety + Measurement + Guide + Equipment Design + HNC-12

132019

On 1 + Rothe & Tuck G.


Atomic Industrial, Golden, Colorado

The material composition of the concrete reflector, used in the low-enriched uranium solution experiments (Task 4), has been analyzed. Two methods were used and good agreement found between the two. The uranium oxide compaction, used for the low-enriched array experiments (Task 7) to attain sufficient density, has been completed. A systematic study has begun to determine the best method of distributing moisture throughout the compacted oxide.


R and D Program + Safety Program + Criticality Experiment + Criticality Safety + Uranium Oxide + Design Criteria + NEC-RC
AN IN-PLANT SOURCE TERM MEASUREMENT STUDY IS BEING CONDUCTED FOR THE NRC AT OPERATING LIGHT WATER NUCLEAR POWER PLANTS. THE PURPOSE OF THE STUDY IS TO PROVIDE THE NRC WITH A DATA BASE FOR USE IN THE LICENSING PROCESS OF PERSONALIZED NUCLEAR REACTOR POWER PLANTS. THIS REPORT PRESENTS THE SAMPLE COLLECTION AND ANALYSIS TECHNIQUES USED IN THE IN-PLANT SOURCE TERM MEASUREMENT PROGRAM. THE INFORMATION WILL BE USED FOR THE PREDICTION OF RADIOACTIVE MATERIALS IN ENVIRONMENTS FROM NORMAL OPERATION OF LINES INCLUDING ANTICIPATED OPERATIONAL OCCURRENCES.

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

R JAB P PROGRAM - SAFETY PROGRAM - REACTOR, LWR + SOURCE, CONTINUOUS + MEASUREMENT + MODEL + PROCEDURES AND HANDLINS + RADIOACTIVITY RELEASE + NRC-88
See the text in the image.
müller at the bhabha atomic research centre, mumbai, india. the purpose of this report is to provide a comprehensive evaluation of the safety aspects of the pebble bed reactor concept, as applied to a reactor site in the united kingdom. the report is based on a detailed analysis of the pebble bed reactor design, including the effects of fuel behavior, fuel management, and reactor safety. the evaluation is carried out using computer code simulations and experimental data. the report concludes that the pebble bed reactor concept meets the safety requirements for a reactor site in the united kingdom.
NUCLEAR SAFETY

A BIMONTHLY REVIEW JOURNAL PREPARED BY NSIC

Nuclear Safety covers significant developments in the field of nuclear safety.

The scope is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors, operations involving fissionable materials, and the products of nuclear fission.

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