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# Regulatory and Technical Reports

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## Preface

This compilation lists formal regulatory and technical reports issued from January through March 1980 by the U.S. Nuclear Regulatory Commission (NRC) staff and by NRC contractors. The compilation is divided into three major sections. The first major section consists of a sequential listing of all NRC reports in report-number order. The first portion of this sequential section lists staff reports, the second portion lists NRC-sponsored conference proceedings, and the third lists contractor reports. Each report citation in the sequential section contains full bibliographic information, including:

- NRC report number
- Report title
- Month and year of issuance
- Contractor (if appropriate)
- Contractor report number (if appropriate)
- NRC originating or sponsoring office
- Availability (where report can be obtained) and price
- Abstract

The second major section of this compilation consists of a key-word index to <u>report titles</u>. Each key word is cross-referenced to the report or reports in the sequential listing that contains that word.

The third major section contains an alphabetically arranged listing of contractor report numbers cross-referenced to their corresponding NRC report numbers.

<u>How to Read Citations</u>

The principal bibliographic elements in a typical report citation are as follows:

<u>Staff Report</u>

① ②

NUREG-0571 Summary of Tube Integrity Operating Experience with Once-Through Steam Generators.

③ → March 1980.

④ → ONRR GPO: $3.25 NTIS ← ⑥

⑤

⑦ Operating problems have occurred in the steam generators designed by each of the three manufacturers of pressurized water reactor nuclear steam supply systems: Babcock & Wilcox, Combustion Engineering, and Westinghouse Electric Corporation. This report focuses on the problems associated with steam generators of the once-through type that are designed and manufactured by Babcock & Wilcox. It identifies the operational experiences observed to date and the position the NRC staff has taken to ensure steam generator tube integrity. It should be noted that a number of research efforts related to these problems are currently under way, and that the information included in this report represents the staff's current understanding of each issue. A similar report, NUREG-0523, discusses tube integrity experience with the U-tube type of steam generator designed by Westinghouse and Combustion Engineering.

1. NRC report number
2. Title
3. Month and year issued
4. Originating office
5. Purchase price at NRC
6. National Technical Information Service availability
7. Abstract

The key to abbreviations for NRC Offices appears at the end of this preface.

Contractor Report

$\underbrace{\text{NUREG/CR-0619}}_{1}$ $\underbrace{\text{Properties of Radioactive Wastes and Waste Containers.}}_{2}$

3 → February 1980.

4 → Brookhaven National Lab.

5 → BNL/NUREG-50957

6 → ONRR GPO: $7.00 NTIS ← 8

7 → GPO

9 This topical report summarizes the results of the NRC-sponsored program, Properties & Radioactive Wastes and Waste Containers, from its inception through September 1978. The properties of waste forms and packages resulting from the solidification of liquid concentrate and solid wastes generated as byproducts of the liquid radioactive waste treatment systems in commercial BWRs and PWRs have been determined. The solidification agents currently employed at power reactors in the United States and (to a lesser extent) agents actively marketed for solidification of these wastes were considered. The waste form and package properties that have been studied in the experimental program include leachability, thermal stability, combustibility, thermal conductivity, mechanical strength, dispersibility, radiation stability, corrosion, and biodegradability. In addition, work has been conducted to determine the effects of various processing parameters on basic waste form criteria.

1. NRC report number
2. Title
3. Month and year issued

4 Contractor

5 Contractor report number

6 NRC sponsoring office

7 Purchase price at NRC

8 National Technical Information Service availability

9 Abstract

## Availability of NRC Publications

Copies of NRC staff and contractor reports may be purchased either from the NRC-GPO Sales Office or from the National Technical Information Service, Springfield, Virginia 22161. To purchase documents from the NRC-GPO Sales Office, send a check or money order, payable to the Superintendent of Documents, to the following address:

U.S. Nuclear Regulatory Commission
ATTN: Sales Manager
Washington, D.C. 20555

You may charge any purchase to your GPO Deposit Account, Master charge card, or VISA charge card by calling the NRC-GPO Sales Office on (301) 492-9530. Non-U.S. customers must make payment in advance either by International Postal Money Order, payable to the Superintendent of Documents, or by draft on a United States or Canadian bank, payable to the Superintendent of Documents.

## NRC Report Codes

The NUREG designation, NUREG-XXXX, indicates that the document is a formal NRC staff-generated report. Contractor-prepared formal NRC reports carry the report code NUREG/CR-XXXX. This type of identification replaces the former uncontrolled use of contractor-established codes such as ORNL/NUREG/TM-XXX and TREE-NUREG-XXXX, as well as various other numbers that could not be correlated with NRC sponsorship of the work being reported.

In addition to the NUREG and NUREG/CR codes, NUREG/CP is used for NRC-sponsored conference proceedings.

All these report codes are controlled and assigned by the NRC Division of Technical Information and Document Control.

## Abbreviations for NRC Offices

OADM - Office of Administration

ACRS - Advisory Committee for Reactor Safeguards

OELD - Office of the Executive Legal Director

OI&E - Office of Inspection and Enforcement

OIP - Office of International Programs

OMPA - Office of Management and Program Analysis

ONMSS - Office of Nuclear Material Safety and Safeguards

ONRR - Office of Nuclear Reactor Regulation

OC - Office of the Controller

OPE - Office of Policy Evaluation

ORES - Office of Nuclear Regulatory Research

OSD - Office of Standards Development

OSP - Office of State Programs

A break in the sequence of NUREG listings indicates that although a NUREG number was issued for a report, either the report was cancelled or it will be issued at some future date.

| Report No. | Bibliographic Data |
|---|---|
| NUREG-0011, Supp. 1 | Supplement 1 to Safety Evaluation Report for Sequoyah Nuclear Plant, Units 1 and 2. Docket Nos. 50-327 and 50-328.<br>March 1980.<br>ONRR GPO: $6.00. NTIS.<br><br>Supplement No. 1 to the Safety Evaluation Report of Tennessee Valley Authority's application for licenses to operate its Sequoyah Nuclear Plant, Units 1 and 2, located in Hamilton County, Tennessee, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement reports the status of certain items that had not been completely resolved at the time of publication of the Safety Evaluation Report, and defines all TMI-2 requirements that must be met for loading fuel and conducting the low-power test program. |
| NUREG-0054 | Supplement No. 3 to Safety Evaluation Report for Offshore Power Systems (Floating Nuclear Plants 1-8). (Supplement 3 to NUREG75/100).<br>February 1980.<br>ONRR GPO: $6.00. NTIS.<br><br>Supplement No. 3 to the Safety Evaluation Report for the Offshore Power System's application for a license to manufacture eight standardized floating nuclear plants in a shipyard-like facility in Jacksonville, Florida, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. This Supplement provides an evaluation of a refractory sacrificial bed, called a core ladle, proposed by Offshore Power Systems to be included in the design of the Floating Nuclear Plant to delay the melt-through penetration of molten core debris in the unlikely event of a core meltdown accident. This design feature is in response to a specific requirement in the Floating Nuclear Plant Final Environmental Statement, Part III, NUREG-0502, that the concrete pad beneath the reactor vessel be replaced with a pad constructed of magnesium oxide or other equivalent refractory material, that will provide increased resistance to melt-through by a molten reactor core, that will not react with core debris to form a large volume of gases, and that will not have any deleterious effects on safety. |
| NUREG-0090, Vol.1, No.3 | Report to Congress on Abnormal Occurrences. July-September 1979.<br>February 1980.<br>OMPA GPO: NTIS<br><br>Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report, the eighteenth in the series, covers the period July 1 to September 30, 1979. During the period, there were five abnormal occurrences. One occurred at the 70 nuclear power plants with operating licenses and involved a major degradation of primary containment boundary. Two occurred at fuel cycle facilities; one involved a mill tailings impoundment dam failure and the second involved an unresolved nuclear material inventory difference. The other two occurred at Agreement State licensees; both involved overexposure of radiography personnel. This report also contains information updating previously reported abnormal occurrences. |
| NUREG-0325, Rev. II | U.S. Nuclear Regulatory Commission Functional Organization Charts.<br>January 1980.<br>OMPA GPO: $2.25. NTIS.<br><br>This document presents detailed functional organizational charts for the U.S. Nuclear Regulatory Commission. This second revision of the chart reflects changes and reorganizations within the NRC through December 10, 1979. |
| NUREG-0386, Supp. 1 | United States Nuclear Regulatory Commission Staff Practice & Procedure Digest. Supplement 1 to Digest No. 2. April 1, 1978 through September 30, 1978.<br>February 1980.<br>OELD GPO: $3.25. NTIS.<br><br>This first supplement to the second edition of the NRC Staff Practice and Procedure Digest contains a digest of a number of Commission, Atomic Safety and Licensing Appeal Board and Atomic Safety and Licensing Board decisions issued during the period from April 1, 1978 to September 30, 1978, interpreting the NRC's Rules of Practice in 10 CFR Part 2. The supplement also includes additional material from adjudicatory decisions rendered prior to April 1, 1978, and, to a very limited degree, material |

from adjudicatory decisions and regulation changes after September 30, 1978. The supplement, which is intended to be used as a "pocket-part" supplement to the Digest itself, includes a number of new subsections and topics not covered in the Digest. The new subsections are noted in the index for the supplement. The Practice and Procedure Digest and this supplement thereto were prepared by attorneys in the NRC's Office of the Executive Legal Director as an internal research tool. Because of the Digest's usefulness to these attorneys, it was decided that it might also prove useful to members of the public. Accordingly, the decision was made to publish the Digest and subsequent editions thereof and supplements thereto, periodically.

NUREG-0398

Federal-State Cooperation in Nuclear Power Plant Licensing.
March 1980.
ONRR GPO: $4.25. NTIS.

A growing number of States are involved in environmental review in nuclear power plant licensing. The diversity of State siting laws and licensing procedures are sources of inefficiency in Federal-State cooperation. Improvements are needed to avoid wasteful duplication and licensing delays. This paper, presented at the Third Annual Meeting of the National Association of Environmental Professionals in February 1978, is updated and presented in three parts. Part I, A Review of Roles and Environmental Issues, discusses: (i) environmental issues and the related roles of the NRC, the utility applicant, and other State and Federal agencies; (ii) the basis of interest for State involvement in licensing activities; and (iii) the basis of interest for a continued Federal role. Part II, Diversity of State Practices in Nuclear Power Plant Licensing, provides a review of contrasting State practices in the licensing process, with special focus on issues such as need for facility and siting. Part III, Federal Initiatives to Improve Federal-State Cooperation in Nuclear Power Plant Licensing, describes initiatives by the NRC to improve licensing cooperation, formal agreements with States regarding licensing procedures, research studies on safety and environmental review methodologies, improved coordination with other Federal agencies, and generic rulemaking.

NUREG-0460, Vol.4

Anticipated Transients Without Scram for Light Water Reactors.
March 1980.
NRC GPO: $6.00. NTIS.

This is the fourth volume of the NRC staff's review of anticipated transient without scram (ATWS). It contains the proposed resolution of this unresolved safety issue (TAP A-9) in the form of requirements recommended to be imposed on licensees and applicants. A phased approach is used: near-term improvements in safety, both hardware and procedural, are required over the next 1 to 2 years to provide an expeditious safety increment. Later, more extensive requirements will be imposed on some plants, but the implementation of these major hardware changes may be stretched out to accommodate equipment acquisition and plant refueling schedules. This delay is intended to allow the changes to be accomplished with minimum disruption and downtime, thus with minimum expense consistent with the level of safety to be achieved. Our present recommendation is more severe than the previous recommendations because the intended generic verification of the safety adequacy of the proposal was not achieved. This report describes the proposed requirements and their phase implementation, gives the staff's technical basis, and considers the values and impacts. The appendices describe the recently submitted industry information and the staff's evaluation of that information, technical details, and some related risk estimates associated with the revised requirements.

NUREG-0543

Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR Part 190).
January 1980.
ONRR GPO: $1.75. NTIS.

The USEPA Uranium Fuel Cycle Standard (40 CFR 190) went into effect for nuclear power facilities on December 1, 1979. This document presents the model technical specifications that implement 40 CFR Part 190 and Appendix I to 10 CFR Part 50, explains the rationale for using Appendix I as a basis for demonstrating compliance with 40 CFR Part 190 for sites with four or less reactors, and describes methods for establishing compliance with 40 CFR Part 190 for sites with radioactive effluents exceeding the Appendix I design objectives or sites with significant direct radiation dose rates (> 5 mrem/yr).

| Report No. | Bibliographic Data |
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NUREG-0565

Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177OFA Operating Plant.
January 1980.
ONRR GPO: $4.75. NTIS.

Slow system depressurization resulting from small-break loss-of-coolant accidents (LOCAs) in the reactor coolant system have not, until recently, received detailed analytical study comparable to that devoted to large breaks. Following the TMI-2 accident, the staff had a series of meetings with Babcock & Wilcox (B&W) and the B&W licensees. The staff requested that B&W and the licensees: (1) systematically evaluate plant response for small-break loss-of-coolant accidents; (2) address each of the concerns documented in the Michelson Report; (3) validate the computer codes used against the TMI-2 accident; (4) extend the break spectrum analysis to very small breaks, giving special consideration to failure of pressurizer valves to close; (5) analyze degraded conditions where AFW is not available; (6) prepare design changes aimed at reducing the probability of loss-of-coolant accidents produced by the failure of a PORV to close; and (7) develop revised emergency procedures for small breaks. This report describes our review of the generic analyses performed by B&W based on the requests stated above.

NUREG-0571

Summary of Tube Integrity Operating Experience with Once-Through Steam Generators
March 1980.
ONRR GPO: $3.25. NTIS.

Operating problems have occurred in the steam generators designed by each of the three manufacturers of pressurized water reactor nuclear steam supply systems: Babcock & Wilcox, Combustion Engineering, and Westinghouse Electric Corporation. This report focuses on the problems associated with steam generators of the once-through type that are designed and manufactured by Babcock & Wilcox. It identifies the operational experiences observed to date and the position the NRC staff has taken to ensure steam generator tube integrity. It should be noted that a number of research efforts related to these problems are currently under way, and that the information included in this report represents the staff's current understanding of each issue. A similar report, NUREG-0523, discusses tube integrity experience with the U-tube type of steam generator designed by Westinghouse and Combustion Engineering.

NUREG-0626

Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications.
January 1980.
ONRR GPO: $6.00. NTIS.

The results of the Bulletins & Orders Task Force's generic evaluation of feedwater transients, small-break loss-of-coolant accidents, and other Three Mile Island Unit 2-related events for General Electric Company-designed operating plants and near-term operating license applications to confirm or establish the bases for the continued safety operation of the operating plants are summarized. As a result of its review, the Bulletins & Orders Task Force has concluded that (1) the continued operation of the General Electric Company-designed operating plants is acceptable provided that certain actions related to the plants' designs and operation and training of operators are implemented consistent with the recommended implementation schedule, and (2) the actions taken by the licensees with General Electric Company-designed operating plants in response to Office of Inspection & Enforcement Bulletin 79-08 provide added assurance for the protection of the health and safety of the public. In addition, the Bulletins & Orders Task Force has independently confirmed the safety significance of those related actions recommended by other Office of Nuclear Reactor Regulation task forces.

NUREG-0627, ES

A Safeguards Case Study of the Nuclear Materials and Equipment Corporation Uranium Processing Plant, Apollo, Pennsylvania. (Performed at the Request of the Honorable Morris K. Udall, Chairman, Committee on Interior and Insular Affairs, U.S. House of Representatives.) Executive Summary.
January 1980.
ONMSS GPO: $1.75. NTIS.

The report characterizes the Atomic Energy Commission safeguards requirements and the safeguards systems and procedures in place at the Nuclear Materials and Equipment (NUMEC) uranium processing plant in Apollo, Pennsylvania, during the Spring of 1964. Based upon this characterization, a list of safeguards weaknesses which would be considered deficiencies under 1979 requirements is developed. Appendices A and B to the report provide a detailed characterization of AEC safeguards requirements as well

| Report No. | Bibliographic Data |
|---|---|
| | as a side-by-side comparis n of NUMEC's safeguards p ogram in 1964 with the safeguards program currently required of a comparable licensed facility. The main report discusses AEC regulatory requirements and philosophy during the mid-1960s, lists the specific areas in which the NUMEC safeguards program would be considered deficient under 1979 NRC requirements, and discusses the conclusions to be drawn from the comparison of the 1964 NUMEC safeguards program with AEC requirements during the mid-1960s and with 1979 NRC requirements. Based upon the deficiencies identified, the report concludes that it is possible that during the mid-1960s, significant quantities of the high enriched uranium could have been removed from the NUMEC Apollo facility, by a knowledgeable insider or by an outside group with the assistance of an insider, without detection. |
| NUREG-0627, MR | A Safeguards Case Study of the Nuclear Materials and Equipment Corporation Uranium Processing Plant, Apollo, Pennsylvania. (Performed at the Request of the Honorable Morris K. Udall, Chairman, Committee on Interior and Insular Affairs, U.S. House of Representatives.) Main Report.<br>March 1980.<br>ONMSS GPO: $5.00. NTIS.<br><br>The report characterizes the Atomic Energy Commission safeguards requirements and the safeguards systems and procedures in place at the Nuclear Materials and Equipment (NUMEC) uranium processing plant in Apollo, Pennsylvania, during the spring of 1964. Based upon this characterization, a list of safeguards w nesses which would be considered deficiencies under 1979 requirements is develop . Appendices A and B to the report provide a detailed characterization of AEC safeguards requirements as well as a side-by-side comparison of NUMEC's safeguards program in 1964 with the safeguards program currently required of a comparable licensed facility. The main report discusses AEC regulatory requirements and philosophy during the mid-1960s, lists the specific areas in which the NUMEC safeguards program would be considered deficient under 1979 NRC requirements, and discusses the conclusions to be drawn from the comparison of the 1964 NUMEC safeguards program with AEC requirements during the mid-1960s and with 1979 NRC requirements. Based upon the deficiencies identified, the report concludes that it is possible that during the mid-1960s, significant quantities of high enriched uranium could have been removed from the NUMEC Apollo facility, by a knowledgeable insider or by an outside group with the assistance of an insider, without detection. |
| NUREG-0628 | NRC Staff Analysis of Public Comments on Advance Notice of Proposed Rulemaking on Emergency Planning.<br>January 1980.<br>OSD GPO: $8.50. NTIS.<br><br>On July 17, 1979, the Nuclear Regulatory Commission (NRC) published an advance notice of proposed rulemaking on emergency planning around nuclear facilities (44 FR 41483). The advance notice announced that NRC was considering adoption of additional regulations which would require increased emergency readiness for public protection on the part of both power reactor licensees and local and State authorities. The notice requested comments on emergency planning matters, including 14 specific issues. Over 100 public comment letters were received. The NRC staff prepared a preliminary analysis of the public comments received and submitted the analysis to the Commission on November 13, 1979. This report consists of the preliminary staff analysis and copies of the public comment letters. |
| NUREG-0629 | U.S. Nuclear Regulatory Commission Budget Estimates Fiscal Year 1981.<br>January 1980.<br>ODB GPO: $3.75. NTIS.<br><br>The budget estimates for Salaries and Expenses for FY 1981 are $468,490. This report provides for the summary of obligations by program; financing; analysis of outlays; function; and the proposed appropriation langauge and analysis and the narrative summary of NRC programs. |

| Report No. | Bibliographic Data |
|---|---|
| NUREG-0635 | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants.<br>February 1980.<br>ONRR GPO: $7.50. NTIS.<br><br>This report summarizes the results of our generic evaluation of feedwater transients, small-break LOCAs, and other TMI-2-related events for the Combustion Engineering-designed operating plants, and confirms the bases for their continued operation. The results of this evaluation are presented in this report in the form of a set of findings and recommendations in each of the principal review areas. |
| NUREG-0637 | Report to the Nuclear Regulatory Commission from the Staff Panel on the Commission's Determination of an Extraordinary Nuclear Occurrence (ENO).<br>January 1980.<br>NRC GPO: $6.00. NTIS.<br><br>In July 1979, the Nuclear Regulatory Commission formally initiated the making of a determination as to whether or not the accident at the Three Mile Island Unit 2 (TMI-2) reactor on March 28, 1979 constituted an "extraordinary nuclear occurrence" (ENO). On August 17, 1979, the Commission directed that a panel comprised of members of the principal staff be formed to evaluate public comments, assemble information relevant to an ENO determination, and report to the Commission its findings and recommendations. This staff report finds and recommends that the TMI-2 accident does not constitute an ENO. |
| NUREG-0639 | Final Environmental Statement Related to the Operation of Split Rock Uranium Mill - Western Nuclear, Inc., Docket No. 40-1162.<br>February 1980.<br>ONMSS GPO: $6.50. NTIS.<br><br>A Final Environmental Statement for Western Nuclear, Inc., related to the renewal of Source Material License SUA-56 for the Split Rock Uranium Mill located in Fremont County, Wyoming (Docket No. 40-1162), has been prepared by the Office of Nuclear Material Safety and Safeguards. This statement provides (1) a summary of environmental impacts and adverse effects of the proposed action and (2) a consideration of principal alternatives. Also included are comments of governmental agencies and other organizations on the Draft Environmental Statement for this project and staff responses to these comments. The NRC has concluded that, after weighing the environmental, economic, technical, and other benefits of the Split Rock Uranium Mill against environmental and other costs and considering available alternatives, the action called for is renewal of the source material license, subject to stipulated conditions. |
| NUREG-0642 | A Review of NRC Regulatory Processes and Functions.<br>January 1980.<br>ACRS GPO: $3.50. NTIS.<br><br>A reexamination by the ACRS of the Regulatory Process has been made. Objectives were to provide in a single source, ACRS' understanding of the Regulatory Process and to point out perceived weaknesses and to make appropriate recommendations for change. |
| NUREG-0643 | Radioactive Waste Processing and Disposal.<br>January 1980.<br>TID-3311-S<br>ONMSS GPO: $12.00. NTIS.<br><br>The Technical Information Center, beginning in 1958, periodically issues bibliographies on radioactive wastes. This compilation contains 4144 citations of foreign and domestic research reports, journal articles, patents, conference proceedings, and books. These citations represent those entered in the DOE Energy Information Data Base since June 1976 through October 1978. These references, as well as references from the period 1967 through June 1976 are available for online searching and retrieval using the DOE RECON system. |

| Report No. | Bibliographic Data |
|---|---|
| NUREG-0644 | Radioactive Waste Processing and Disposal.<br>January 1980.<br>TID-3311-S9<br>GNMSS GPO: $12.00. NTIS.<br><br>The Technical Information Center, beginning in 1958, periodically issues bibilographies on radioactive wastes. This compilation contains 3597 citations of foreign and domestic research reports, journal articles, patents, conference proceedings, and books. These citations represent those entered in the DOE Energy Information Data Base since July 1978 through December 1979. These references, as well as references from the period January 1967 through July 1978, are available for online searching and retrieval using the DOE RECON system. |
| NUREG-0645, Vol.1 | Report of the Bulletins & Orders Task Force of the Office of Nuclear Reactor Regulation.<br>January 1980.<br>ONRR GPO: $4.75. NTIS.<br><br>The results of the Bulletins & Orders review of the Office of Inspection and Enforcement bulletins, Commission Orders, and the Office of Nuclear Reactor Regulation generic evaluation of feedwater transients, small-break loss-of-coolant accidents, and other Three Mile Island Unit 2-related events in operating plants to confirm or establish the bases for their continued safe operation are summarized. As a result of its review, the Bulletins & Orders Task Force has concluded that (1) the continued operation of the operating plants is acceptable provided that certain actions related to the plants' designs and operation and training of operators are implemented consistent with the recommended implementation schedules, and (2) the actions taken by the licensees with operating plants in response to the IE bulletins (including the actions specified in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors") provide added assurance for the protection of the health and safety of the public. In addition, the Bulletins & Orders Task Force has independently confirmed the safety significance of those related short-term and long-term actions recommended by other Office of Nuclear Reactor Regulation task forces. |
| NUREG-0645, Vol.2 | Report of the Bulletins & Orders Task Force of the Office of Nuclear Reactor Regulation. Appendices.<br>January, 1980.<br>ONRR GPO: $12.00. NTIS.<br><br>The results of the Bulletins & Orders review of the Office of Inspection and Enforcement bulletins, Commission Orders, and the Office of Nuclear Reactor Regulation generic evaluation of feedwater transients, small-break loss-of-coolant accidents, and other Three Mile Island Unit 2-related events in operating plants to confirm or establish the bases for their continued safe operation are summarized. As a result of its review, the Bulletins & Orders Task Force has concluded that (1) the continued operation of the operating plants is acceptable provided that certain actions related to the plants' designs and operation and training of operators are implemented consistent with the recommended implementation schedules, and (2) the actions taken by the licensees with operating plants in response to the IE bulletins (including the actions specified in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors") provide added assurance for the protection of the health and safety of the public. In addition the Bulletins & Orders Task Force has independently confirmed the safety significance of those related short-term and long-term actions recommended by other Office of Nuclear Reactor Regulation task forces. |
| NUREG-0646 | Report of the Advisory Committee on Construction During Adjudication.<br>January 1980.<br>NRC GPO: $8.50 NTIS.<br><br>In the Nuclear Regulatory Commission's Seabrook Opinion of January 6, 1978, the Commission directed that an internal study be made of:<br>1. the effect which would be achieved by relaxation of NRC's stay standards so that site-related issues in potentially troublesome nuclear power plant licensing proceedings could be taken up before utilities invest large sums of money and sites are irrevocably altered; and<br>2. ways in which the NRC's appellate administrative procedures could assure earlier resolution of all the issues arising out of nuclear power plant licensing and cut relitigation and piecemeal review to a minimum. |

Report No. Bibliographic Data

In December 1978, the Commission chose Gary Milhollin, a faculty member at the University of Wisconsin Law School, to chair a study group composed of nine other members. These members were chosen by the heads of various offices within the Commission. This report describes the Committee's work, summarizes the data gathered by the Committee, and recommends action by the Commission.

NUREG-0648 Study of the Nuclear Regulatory Commission's Appellate System.
January 1980.
OGC GPO: $3.25. NTIS.

At the Chairman's request, the Office of General Counsel studied the Commission's appellate system. The study included examination of the development of the study, analysis of the current workload, investigation of the practices of other agencies, and consideration of alternatives to the present system. The study recommends retention of the present system with some limited modifications to enable the Commission to more effectively use its existing appellate powers.

NUREG-0649 Task Action Plan for Unresolved Safety Issues Related to Nuclear Power Plants.
February 1980.
ONRR GPO: $5.00. NTIS.

This document contains Task Action Plans for generic tasks addressing Unresolved Safety Issues related to nuclear power plants. Progress on "Unresolved Safety Issues" is reported to Congress each year in the NRC Annual Report pursuant to the requirements of Section 210 of the Energy Reorganization Act of 1974, as amended. Those issues that have been designated "Unresolved Safety Issues" are listed. A description of the process used to identify issues is included in NUREG-0510 issued January 1979. The Task Action Plans in this document included a description of the issue, a description of the NRC staff's approach to resolving the issue, a general discussion of the basis for continued operation and licensing pending resolution of the issue, a discussion of the technical organizations involved in the task, and the requirements for manpower and program support funding. The report only includes Task Action Plans for generic tasks that have not been completed as of the issuance date. Those tasks that have already been completed are also identified and the NUREG reports providing the NRC staff's resolution of each is referenced.

NUREG-0651 Evaluation of Steam Generator Tube Rupture Events.
March 1980.
ONRR GPO: $4.25. NTIS.

The NRC staff's review of three domestic pressurized water reactor steam generator tube rupture events has shown that no significant offsite doses or systems performance inadequacies have occurred. The plant operators and systems successfully avoided direct releases to the environment (via the steam generator power operated relief valves or safety valves) and brought the reactor to a safe shutdown condition. However, a number of relatively minor procedural and equipment deficiencies were noted.

NUREG-0652 Facilities License Application Record.
March 1980.
OMPA GPO: $2.00. NTIS.

The FLAR is divided into three parts as follows: Part I - The first part is a listing of all operating licenses, authorizations, and construction permits in effect plus applications pending. Part II - The second part is a listing of applications that have been withdrawn or denied, and licenses and construction permits terminated or revoked. Part III - The third part is a listing of research, test, and power reactors for export.

NUREG-0654 Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants.
February 1980.
FEMA-REP-1
NRC/FEMA GPO: $4.00. NTIS.

The purpose of this interim guidance and upgraded acceptance criteria document is to provide a basis for NRC licensees, State, and local governments to develop radiological emergency plans and improve emergency preparedness. The guidance is the product of the joint FEMA/NRC Steering Committee established to coordinate the agencies' work in emergency preparedness associated with nuclear power plants. This document superseded previous guidance and criteria published by FEMA and NRC. It will be used by reviewers in determining the adequacy of State, local, and nuclear power plant operator emergency plans and preparedness.

NUREG-0656

Study of Alternative Methods for the Management of Liquid Scintillation Counting Wastes.
January 1980.
ONMSS GPO: $2.50. NTIS.

The Nuclear Engineering Waste Disposal Site in Richland, Washington, is the only radioactive waste disposal facility that will accept liquid scintillation counting wastes (LSCW) for disposal. That site is scheduled to discontinue receiving LSCW by December of 1982 The objective of this document is to study alternative methods for the management of LSCW which constitutes 40% in volume of the institutional waste generated in the U.S. The LSCW total radioactivity content is estimated at 8 Ci/year. Although this amount of radioactivity is relatively small, LSCW must be treated as radioactive waste under existing regulations. Disposal problems are compounded and some options limited because of the chemical nature of the organic solvents, e.g., their flammability and chemical toxicity. The techniques that have been evaluated as alternative methods for the management of LSCW are: (1) evaporation, (2) distillation, (3) solidification, (4) conversion to a less hazardous chemical form, and (5) combustio (which includes incineration and addition to fuel). Presently, incineration appears to be the most viable alternative, although some of the other methods seem promising but require further development. Also, the waste disposal is likely to be simplified by reduction of the volume generated at the source and segregation of the waste by radioactivity concentration.

NUREG-0657

Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program for Fiscal Year 1981.
February 1980.
ACRS GPO: $3.75. NTIS.

Public Law 95-209 includes a requirement that the Advisory Committee on Reactor Safeguards submit an annual report to Congress on the safety research program of the Nuclear Regulatory Commission. This report presents the results of the ACRS review and evaluation of the NRC safety research program for Fiscal Year 1981. The report contains a number of comments and recommendations.

NUREG-0658

Sequoyah - Unit 1 Technical Specifications.
March 1980.
ONRR GPO: $9.00. NTIS.

The Sequoyah Unit 1 Technical Specifications were prepared by the U.S. Nuclear Regulatory Commission. The Sequoyah Unit 1 Technical Specifications set forth the limits, operating conditions, and other requirements applicable to a nuclear reactor facility as set forth in Section 50.36 of 10 CFR Part 50 for the protection of the health and safety of the public.

| Report No. | Bibliographic Data |
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| NUREG/CP-0010 | CSNI Specialists' Meeting on Plastic Tearing Instability.<br>January 1980.<br>Washington University<br>ORES GPO: $9.00. NTIS. |

The Specialist Meeting on Plastic Tearing Instability was held at the Center for Fracture Mechanics, Washington University, St. Louis, USA, and was sponsored by the Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency. The meeting was hosted by the Center for Fracture Mechanics and by the United States Nuclear Regulatory Commission. The Meeting constituted a forum for the 45 participants to make presentations and to thoroughly discuss details of the material presented. Copies of the material presented appear in this report.

| Report No. | Bibliographic Data |
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| NUREG/CR-0497, Rev.1 | MATPRO II Rev. 1: A Handbook of Material Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior.<br>February 1980.<br>EG&G Idaho<br>TREE-1280, Rev.1<br>ONRR GPO: $6.50. NTIS.<br><br>This handbook describes the materials properties correlations and computer subcodes, MATPRO-Version 11 (Revision 1) developed for use with various LWR fuel rod behavior analytical programs at the Idaho National Engineering Laboratory. Formulations of fuel rod material properties, which are generally semiempirical in nature, are presented for uranium dioxide and mixed uranium-plutonium dioxide fuel, zircaloy cladding, and fill gas mixtures. |
| NUREG/CR-0575 | PINSIM-MOD1: A Nuclear Fuel Pin/Electric Fuel Pin Simulator Transient Analysis Code.<br>January 1980.<br>Oak Ridge National Lab<br>ORNL/NUREG/TM-291<br>ONRR GPO: $6.00. NTIS.<br><br>The PINSIM-MOD1 digital computer code has been developed on an IBM 360/195 computer for simulating the thermal response of both nuclear fuel pins and electrically heated fuel pin simulators to reactor accident environments, determining transient simulator power input, and analyzing the results of simulator experiments. The code is capable of simulating the thermal response of up to four pins and their associated coolant channels. |
| NUREG/CR-0619 | Properties of Radioactive Wastes and Waste Containers.<br>February 1980.<br>Brookhaven National Lab<br>BNL/NUREG-50957<br>ONRR GPO: $7.00. NTIS.<br><br>This topical report summarizes the results of the NRC sponsored program, Properties & Radioactive Wastes and Waste Containers, from its inception through September 1978. The properties of waste forms and packages resulting from the solidification of liquid concentrate and solid wastes generated as by-products of the liquid radioactive waste treatment systems in commercial BWRs and PWRs have been determined. The solidification agents currently employed at power reactors in the United States and (to a lesser extent) agents actively marketed for solidification of these wastes were considered. The waste form and package properties that have been studied in the experimental program include leachability, thermal stability, combustibility, thermal conductivity, mechanical strength, dispersibility, radiation stability, corrosion, and biodegradability. In addition, work has been conducted to determine the effects of various processing parameters on basic waste form criteria. |
| NUREG/CR-0675 | Test of 6-inch-thick Pressure Vessels, Series 3: Intermediate Test Vessel V-8.<br>March 1980.<br>Oak Ridge National Lab<br>ORNL/NUREG-58<br>ONRR GPO: $6.00. NTIS.<br><br>Intermediate test vessel V-8, a 152-mm-thick vessel fabricated of SA 553, grade B, Class 1 steel, was pressurized to failure at a test temperature of -23°C. The vessel contained a large fatigue-sharpened notch adjacent to a half-bead weld repair that had not been stress relieved. Residual stresses and fracture toughness were determined before the pressure test by measurements on a prototypic weld, and fracture predictions were made by linear elastic fracture mechanics. Predictions agreed well with results of the test, which demonstrated the important influence of high residual stresses on fracture behavior. |

| Report No. | Bibliographic Data |
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| NUREG/CR-0709 | KENO-IV/CG, The Combinatorial Geometry Version of the Keno Monte Carlo Criticality Safety Program.<br>January 1980.<br>Oak Ridge National Lab<br>ORNL/NUREG/CSD-7<br>ONMSS GPO: $4.25. NTIS.<br><br>The purpose of this report is to demonstrate the unique capabilities contained in the geometric modeling facilities of KENO-IV/CG, a new combinatorial geometry version of the KENO Monte Carlo criticality program. KENO-IV/CG was developed to merge the simple geometry input description utilized by combinatorial geometry with the repeating lattice feature of the original KENO geometry package. The result is a criticality code with the ability to model a complex system of repeating rectangular lattices inside a complicated three-dimensional geometry system. Furthermore, combinatorial geometry was modified to differentiate between combinatorial zones describing a particular KENO BOX to be repeated in a KENO array and those combinatorial zones describing geometry external to an array. This allows the user to maintain a simple coordinate system without any geometric conflict due to spatial overlap. Several difficult criticality design problems have been solved with the new geometry package KENO-IV/CG, thus illustrating the power of the code to model difficult geometries with a minimum of effort. |
| NUREG/CR-0722 | Fission Product Release from Highly Irradiated LWR Fuel.<br>March 1980.<br>Oak Ridge National Lab<br>ORNL/NUREG/TM-287/R2<br>ONRR GPO: $5.00. NTIS.<br><br>A series of experiments was conducted with highly irradiated light-water reactor fuel rod segments to investigate fission products released in steam in the temperature range 500 to 1200°C. (Two additional release tests were conducted in dry air.) The primary objectives were to quantify and characterize fission produce release under conditions postulated for a spent-fuel transporation accident and for a successfully terminated loss-of-coolant accident (LOCA). Various parameters that affect fission product release are discussed, and experimental observations and analysis of the chemical behavior of releaseable fission products in inert, steam, and dry-air atmospheres are examined. |
| NUREG/CR-0749 | The Thermal and Mechanical Behavior of Xenon-Filled Fuel Rod as a Function of Burnup.<br>February 1980.<br>Battelle Northwest Lab<br>PNL-2948<br>ONRR GPO: $4.25. NTIS.<br><br>This report examines the life history behavior of Rod 4, IFA-431 based on in-reactor fuel centerline temperature and cladding elongation data, and postirradiation, nondestructive examination. Instrumented fuel assembly IFA-431 was irradiated in the Halden Boiling Water Reactor from June 1975-February 1976. Rod 4 of this assembly was built with concentric and eccentric fuel regions and then backfilled with 100% xenon gas in order to study the effects of fill gas composition and fuel-cladding geometry. It has been concluded that fuel relocation was responsible for the significant decrease in thermal resistance that was observed for the concentric fuel region. The thermal resistance for the eccentric fuel region was lower than for concentric region (for equal linear heat rate) and remained nearly constant during the irradiation. |
| NUREG/CR-0797 | Post-Irradiation Data Analysis for NRC/PNL Halden Reactors Assembly IFA-431.<br>February 1980.<br>Battelle Northwest Lab<br>PNL-2975<br>ONRR GPO: $6.00. NTIS.<br><br>This report presents results of the post irradiation examination performed on IFA-431, which was a 6-rod test fuel assembly irradiated in Halden Reactor, Norway, under sponsorship of the Nuclear Regulatory Commission. The irradiation conditions included: peak powers of 33 kW/m; coolant pressure and temperature of 3.3 MPa and 240°C, respectively; and peak burnup of 4300 MWd/MTM. IFA-431 included instrumented rods of basic boiling water reactor design, with variations in fill gas composition, gap size, and $UO_2$ fuel type. The irradiation was designed to measure the effect of these variations upon fuel rod thermal and mechanical performance. |

| Report No. | Bibliographic Data |
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NUREG/CR-0803

Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. August 7, 1978-July 31, 1979.
February 1980
Univ. of Washington, Seattle
ONRR GPO: $3.75. NTIS.

Sediments potentially serve as important sinks and sources for radionuclides released into aquatic systems. In this regard, we have collected data over the past year to evaluate mechanisms affecting particle-water distributions of selected radionuclides ($^{57}$Co, $^{60}$Co, $^{85}$Sr, $^{106}$Ru, $^{137}$Cs, $^{237}$Pu, $^{241}$Am and $^{244}$Cm). Samples used in these studies were obtained from several natural freshwater, estuarine and marine locations around the continental United States. Particle-water distribution coefficients ($K_d$) were found to vary from $10^2$ to $10^6$ depending on the specific radionuclide, the type of particle(s), and the overall physical and chemical properties of a given test system. In general, $K_d$ values increased as follows: $^{85}$Sr < $^{137}$Cs < Ru or $^{237}$Pu < $^{241}$Am. However, both the absolute values of these Kd's and their magnitudes relative to other radionuclides were found to be dramatically and nonuniformly affected by certain variables. The variables tested and discussed in this report include: the quantity and chemical composition of the particles. the degree of reversibility of radionuclide-sediment interactions, the pH of the water, the abundance and composition of organic ligans and the chemical speciation (i.e., molecular weight size associations) of the radionuclides. This data will contribute to the formulation of predictive models to describe distributions and eventual fates of radionuclides in aquatic environments.

NUREG/CR-0857

Properties of Radioactive Wastes and Waste Containers Progress Report No. 10. July-September 1978.
February 1980.
Brookhaven National Lab
BNL/NUREG-51026
ONRR GPO: $2.25. NTIS.

Modified IAEA and equilibrium leaching studies were performed using portland cement, urea-formaldehyde and vinyl ester-styrene waste forms containing BWR chemical regenerative waste. This set of experiments should cover the range of expected leach rates for most conditions. The equilibrium leach method generally resulted in lower release rates than the modified IAEA method. In the case of Strontium-85 release from portland cement, this decrease was dramatic. The use of finite and semi-infinite diffusional mass transport models for leaching data analysis to predict long-term releases from full scale waste forms is discussed. While the finite model can give a more accurate description of activity release since it considers specimen depletion effects, the double series function solution converges slowly for small effective diffusivities and small total numbers of summation terms. This can result in computer truncation error dominating in the solutions for small releases. In this report period, the first topical report for the program "Properties of Radioactive Wastes and Waste Containers" was also written. This document appears under a separate cover.

NUREG/CR-0858

Stress Corrosion Cracking of Inconel 600 Tubing in Deaerated High Temperature Water.
February 1980.
Brookhaven National Lab
BNL/NUREG-51027
ONRR GPO: $3.25. NTIS.

Because of the importance of avoiding primary-to-secondary leack through the Inconel 600 pressure boundary, a research program was started at Brookhaven National Lab in an attempt to improve the qualitative and quantitative understanding of factors influencing SCC in deaerated high temperature aqueous media. These data are also intended for predicting service performance under given sets of conditions. This report includes preliminary results which indicate that intergranular SCC is produced quite readily in some heats of tubing when exposed at (a) constant deflection, (b) when slightly cold worked and subjected to straining at low rates, (c) when subjected to slow cyclic stress, or (d) when the electrochemical potential of stressed pieces is controlled.

| Report No. | Bibliographic Data |
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| NUREG/CR-0929 | Sabres II: An Individual Resolution Small Arms Combat Simulation Model.<br>March 1980.<br>Sandia Lab, Livermore<br>SAND79-8249<br>ONRR GPO: $3.50. NTIS.<br><br>SABRES II is an individual resolution computer simulation of combat between two small groups using small arms weapons. The model is designed to simulate a battle between the protective force of a special nuclear material (SNM) transportation convoy and an adversary force attempting to steal the material. The usefulness of the model lies in its ability to compare, in a consistent manner, the relative value of the variations in the characteristics of an SNM transportation physical protection system. Two versions of the SABRES II model are described. An interactive version, allowing the user to control the movement of all combatants, aids in the generation of plans and tactics to be followed by each force during the conflict. A Monte Carlo version of the model, using the interactive plans and tactics derived from multiple simulations of the scenario, generates statistical analysis of the simulated conflict. |
| NUREG/CR-0933 | An Improvement in the Calculation of Turbulent Friction in Smooth Concentric Annuli.<br>January 1980.<br>Brookhaven National Lab<br>BNL/NUREG-51025<br>ONRR GPO: $2.00. NTIS.<br><br>Data and calculation techniques for frictional pressure loss in smooth concentric annuli are reviewed. It is shown that the accepted method of Meter and Bird, and of Rothful, Monrad, and Senecal deviate by 4% and 8%, respectively, from the correct limit for small gaps. Further, neither correctly predicts the data trends with decreasing radius ratio. In addition, the data upon which the Meter and Bird correlation was based, in part, was probably deficient due to strong secondary flow caused by the unrestricted exit tee. Finally, the method of Jones, previously applied to rectangular ducts, has been extended to smooth, concentric annuli. It is shown that both the trends and the limiting behavior are quantitatively correct. |
| NUREG/CR-0957 | FLX: A Shell Code for Coupled Fluid-Structure Analysis of Core Barrel Dynamics.<br>March 1980.<br>Los Alamos Scientific Lab<br>LA-7957<br>ONRR GPO: $3.25. NTIS.<br><br>The coupled fluid-structure dynamics of a light water reactor core support barrel can be calculated by the K-FIX(3D, FLX) and SOLA-FLX codes. The fluid dynamics is described by the three-dimensional, two-fluid code K-FIX(3D) or the two-dimensional, drift-flux code SOLA-DF. The structural dynamics is described by the three-dimensional, elastic shell code FLX. FLX, which uses an explicit finite-difference solution algorithm to solve the shell equations, is explicitly coupled to the fluid-dynamics codes. Motion may be induced by blowdown, prescribed displacement, or seismic action. A sample calculation is provided for verification. |
| NUREG/CR-0962 | Reactor Safety Research Programs, Quarterly Report April-June 1979.<br>January 1980.<br>Battelle Northwest Lab<br>PNL-3040-2<br>ONRR GPO: $4.75. NTIS.<br><br>This document summarizes the work performed by Pacific Northwest Laboratory from April through June 1979 for the Division of Reactor Safety Research of the Nuclear Regulatory Commission. Each program is considered separately and discussed according to major tasks or topics, depending on the nature of the project. |

| Report No. | Bibliographic Data |
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| NUREG/CR-0966 | Sim, 'ified Damage Assessment of Nuclear Power Plants Objected to Turbine Fragments.<br>March 1980.<br>Los Alamos Scientific Lab<br>LA-7944-MS, Rev.<br>ONRR GPO: $2.50. NTIS.<br><br>An analytical method and associated computer program are described by which the probability of damaging critical components of a nuclear power station can be determined. Input information consists of parameters that define the turbine, barriers, and targets. The program produces a matrix that relates the probability of damaging each target to each turbine wheel individually, as well as the aggregate of all turbine wheels in the plant. In addition, the probability of either striking or penetrating selected barriers can be obtained. Results of an example calculation are presented. |
| NUREG/CR-0970 | Qualification Testing Evaluation Program Light-Water-Reactor Safety Research Quarterly Report January-March 1979.<br>January 1980.<br>Sandia Lab, Albuquerque<br>SAND79-1314<br>ONRR GPO: $2.25. NTIS.<br><br>The objectives of the Qualification Testing Evaluation (QTE) progrm are to confirm the suitability of current standards and regulatory guides for Class IE safety-related equipment and to provide an improved technical basis for modifications of these standards and guides. Major objectives of the research are: (1) to provide assessments of post-LOCA qualification testing methodologies, including a qualitative assessment of the synergistic effects resulting from the combined environmental testing of representative Class IE equipment; (2) to determine the radiation environment from the nuclear source term for a design basis LOCA and evaluate the adequacy of radiation simulators; and (3) to provide methods that can be used to simulate the natural aging process of representative Class IE materials by accelerated aging methods. |
| NUREG/CR-0977 | New Madrid Seismotectonic Study, Activities During Fiscal Year 1979.<br>March 1980.<br>St. Louis University<br>ONRR GPO: $5.00. NTIS.<br><br>A bibliography of more than 1000 studies on continental rifts was prepared. Aeromagnetic and gravity data from much of the area were integrated and gridded on a 2-km grid. A new hypothesis, based on the integrated geophysical data, suggests that the linear tectonic features associated with the New Madrid seismic zone may extend northeastward across the Rough Creek Fault Zone toward central Indiana. Several earthquakes recorded by the Wabash Valley seismograph array had focal depths greater than 16 km, the deepest ever recorded in the central Mississippi Valley region. Most faulting in the Fluorspar District, southeastern Illinois was shown to have taken place prior to deposition of Late Cretaceous sediments. An isopach map of the basal clastic sediments in the area showed the Rough Creek Graben and Reelfoot Rift to be features that subsided significantly prior to Late Cambrian time. |
| NUREG/CR-0984 | Advanced Reactor Safety Research - Quarterly Report April - June 1979.<br>January 1980.<br>Sandia Lab, Albuquerque<br>SAND79-1597 Vol. 10<br>ONRR GPO: $6.00. NTIS.<br><br>Sandia Laboratories, Albuquerque, New Mexico, is conducting the Advanced Reactor Safety Research Program on behalf of the U.S. Nuclear Regulatory Commission (NRC). The overall objective of the program is to provide NRC a comprehensive data base essential to (1) defining key safety issues, (2) understanding the controlling accident sequences, (3) verifying the complex computer models used in accident analysis and licensing reviews, and (4) assuring the public that advanced power reactor systems will not be licensed and placed in commercial service in the United States without appropriate consideration being given to their effects on health and safety. |

| Report No. | Bibliographic Data |
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**NUREG/CR-0994**

A Radiative Heat Transfer Model for the TRAC Code.
January 1980.
Los Alamos Scientific Lab
LA-7965-MS
ONRR GPO: $3.50. NTIS.

This report describes a radiative heat transfer model proposed for the boiling water reactor (BWR) version of the Transient Reactor Analysis Code (TRAC). The model includes radiative heat transfer to the nonequilibrium two-phase fluid, as well as surface-to-surface radiative heat transfer. Radiative properties and geometric view factors are calculated. It is assumed that each surface is a gray surface and that the liquid and vapor are gray fluids. A stand-alone code RADHT is described, and this code was used to examine the effect of the void fraction, liquid droplet diameter, vapor temperature, wall emissivity, and the number of rod groups. The results show that five rod groups (including the channel wall) are sufficient to describe the radiative heat transfer in a BWR bundle. The results also show that radiative heat transfer to the two-phase mixture should be considered in addition to the surface-to-surface effects.

**NUREG/CR-1003**

Generalized Sensitivity Theory for Systems of Coupled Nonlinear Equations.
February 1980.
Oak Ridge National Lab.
ORNL/NUREG/TM-349
ONRR GPO: $4.25. NTIS.

A general sensitivity theory is presented for treating problems characterized by systems of nonlinear equations with nonlinear responses. Frechet derivatives are used in both differential and variational approaches to derive appropriate adjoint equations and expressions for sensitivity functions. The two approaches are unified to form a complete operator viewpoint of sensitivity theory. Also presented is an alternative sensitivity formalism for systems of nonlinear matrix equations such as those arising from the application of numerical methods to many practical problems. This approach significantly enlarges the scope and versatility of sensitivity theory as it allows direct treatment of parameters which are purely of numerical methods origin. Practical applications are discussed.

**NUREG/CR-1006**

Preliminary Design of a Large Scale Graphite Oxidation Loop.
February 1980.
Brookhaven National Lab
BNL/NUREG-51054
ONRR GPO: $2.25. NTIS.

A preliminary design study of a large scale graphite oxidation loop was performed in order to assess feasibility and to estimate capital costs. The nominal design operates at 50 atmospheres helium and 1800 F with a graphite specimen 30 inches long and 10 inches in diameter. It was determined that a simple, single-walled design was not practical at this time because of a lack of commercially available thick-walled, high temperature alloys. Two alternative concepts, at reduced operating pressure, were investigated. Both were found to be readily fabricable to operate at 1800 F and capital cost estimates for these are included. A design concept was briefly considered. The full design pressure, temperature and dimensions of the nominal design could possibly be accommodated in a design in which the pressure boundaries of the loop are maintained at lower temperatures through use of thermal barriers and/or double-walled sections. This would greatly complicate the mechanical design of the test section, the heat exchangers, and the piping, however.

**NUREG/CR-1007**

Evaluation of Selected Signal Processing Methods for the Characterization of Steam Generator Eddy Current Signals.
March 1980.
Brookhaven National Lab
BNL/NUREG-51044
OSD GPO: $3.75. NTIS.

The primary objective of this program was to implement a digital computer subtraction technique and evaluate the method using eddy current data derived from models of observed inservice defects. The Zetec ML-2 signal analyzer was also briefly examined. A secondary objective was to investigate the multifrequency aspects of eddy current inspection with regard to the detection and characterization of signal types.

| Report No. | Bibliographic Data |
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NUREG/CR-1010

Evaluation of Docket Files for Terminated Source Material Licenses.
February 1980.
Oak Ridge National Lab
ORNL/NUREG/TM-342
ONRR GPO: $3.50. NTIS.

Terminated source material licenses from docket files of the NRC have been evaluated with respect to the potential for residual radiological health problems. The purpose of the present study is to effect a preliminary screening of those source material docket files which have been terminated by the NRC or its predecessor, the AEC. Because source material licensing is covered in Part 40 of Title 10 Code of Federal Regulations, the source material dockets are referred to as Part 40 dockets. Evaluation of these docket files was done to determine which, if any, of the sites involved might present a potential public radiological health hazard.

NUREG/CR-1011

Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2.
February 1980.
Oak Ridge National Lab
ORNL/NUREG/TM-350
ONRR GPO: $6.00. NTIS.

The experimental and COBRA-IV computational data presented in this report confirm that increased pressure losses, induced by the steady-state axial flow of water exterior to deformed Multirod Brust Test (MRBT) bundles B-1 and B-2, may be closely predicted using a bundle-averaged approach for describing flow channel restrictions. One anomaly that was encountered using this technique occurred while modeling the B-2 flow test data near a severe channel restriction: the COBRA-IV results tended to underestimate experimental pressure losses.

NUREG/CR-1012

Procedure for the Qualitative Interpretation of Fuel Centerline Thermocouple Response to Step-Power Decreases.
January 1980.
Battelle Pacific Northwest Lab
PNL-3096
ONRR GPO: $4.25. NTIS.

This report reviews the present calculational techniques that may be used to interpret the transient response of fuel centerline thermocouples to step decreases in rod power. A new technique developed herein involves plotting the natural logarithm of the normalized thermocouple data versus time, plotting various calculations in the same way, and observing the curvature of the resulting lines. Also described is the small computer code, MWRAM, which facilitates testing various models against transient data. Transient data from IFA-513 is presented. This test assembly in the Halden Reactor, Norway, is jointly sponsored by the Nuclear Regulatory Commission and the Halden Project. A comparison of MWRAM calculations with this data has shown that fuel cracking appears to greatly influence the heat transfer modes in the fuel rod. A method of estimating the effective fuel-cladding gap size from this transient data is also discussed.

NUREG/CR-1027

Disassembly Phase Energetics: An Examination of the Impact of Simmer Models and Assumptions.
January 1980.
Los Alamos Scientific Lab
LA-7998-MS
ONRR GPO: $3.50. NTIS.

This report presents an assessment of several aspects of material motion as predicted by the SIMMER code during the disassembly and early transition phase of an unprotected loss of flow transient in a liquid metal cooled advanced reactor. The study was divided into the fuel expansion phase of an energetic disassembly and into the early fuel motion stage of subassembly scale pin disruption under the forces of fission gas dispersal and sodium vapor flow.

| Report No. | Bibliographic Data |
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NUREG/CR-1029

Program Plan for the Investigation of Vent-Filtered Containment Conceptual Design for Light Water Reactors.
March 1980.
Sandia Lab, Albuquerque
SAND79-1080
ONRR GPO: $3.75. NTIS.

A containment venting and filtration capability has been suggested as a means for reducing the risk from fuel melt accidents in light-water reactors. The risk reduction potential for such systems depends upon the dual function of venting containment to prevent overpressurization from the generation of steam and noncondensibles and filtering the effluent to limit the release of radioactive materials. This report addresses the major issues involved in such an accident mitigation system and discusses the engineering, technical, and economic questions that will have to be studied before judgments can be made regarding feasibility and effectiveness. A program plan is presented for research leading to the formulation of design requirements for vent-filter containment systems and to a comprehensive assessment of the values versus impacts of such systems.

NUREG/CR-1031

Stress and Duress Monitoring at NRC-Licensed Facilities.
January 1980.
Brookhaven National Lab
BNL/NUREG-51089
ONRR GPO: $2.00. NTIS.

Various current and near-future methods of detecting stress in humans are evaluated as to effectiveness and cost with a view to application as a screening mechanism at portals at NRC-licensed facilities. Also, similar and related techniques of stress detection and covert switches are evaluated for use by guards at NRC-licensed facilities as methods of informing the Central Alarm Station that the guard is under duress.

NUREG/CR-1032

Stress and Duress Monitoring at NRC-Licensed Facilities.
January 1980.
Brookhaven National Lab
BNL/NUREG-51090
ONRR GPO: $3.75. NTIS.

Various current and near-future methods of detecting stress in humans are evaluated as to effectiveness and cost with a view to application as a screening mechanism at portals at NRC-licensed facilities. Also, similar and related techniques of stress detection and covert switches are evaluated for use by guards at NRC-licensed facilities as methods of informing the Central Alarm Station that the guard is under duress.

NUREG/CR-1035

Water Reactor Safety Research Division Quarterly Progress Report. April-June 1979.
February 1980.
Brookhaven National Lab
BNL/NUREG-51081
ONRR GPO: $3.50. NTIS.

The Water Reactor Safety Research Programs Quarterly Report describes current activities and technical progress in the programs at Brookhaven National Laboratory sponsored by the USNRC Division of Reactor Safety Research. The projects reported each quarter are the following: LWR Thermal Hydraulic Development, RAMONA Code Evaluation, TRAC Code Assessment, and Stress Corrosion Cracking of PWR Steam Generator Tubing.

NUREG/CR-1036

Advanced Reactor Safety Research Division Quarterly Progress Report. April-June 1979.
February 1980.
Brookhaven National Lab
BNL/NUREG-51082
ONRR GPO: $4.00. NTIS.

The Advanced Reactor Safety Research Programs Quarterly Progress Report describes current activities and technical progress in the programs at Brookhaven National

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Laboratory sponsored by the USNRC Division of Reactor Safety Research. The projects reported each quarter are the following: HTGR Safety Evaluation, SSC Code Development, LMFBR Safety Experiments and Fast Reactor Safety Code Validation.

NUREG/CR-1037

Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites. Progress Report No. 10, July-September 1978.
February 1980.
Brookhaven National Lab
BNL/NUREG-51083
ONRR GPO: $2.25. NTIS.

A survey study of commercial low-level radioactive waste disposal sites was conducted for the purpose of selecting specific trenches at each site for in-depth studies. Several trenches from the West Valley, New York, and Maxey Flats, Kentucky, disposal sites were selected for further study due to their locations and the values of pH, specific conductance, dissolved organic carbon (DOC), alpha, beta, gamma, and tritium activities measured in the survey study. They represent extreme and medium values of these parameters and are located within and around the perimeter of the buried areas. This report provides inorganic, organic, and radiochemical data of these trench waters and describes the in-situ measurements made while collecting the samples.

NUREG/CR-1039

Estimated Radiation Doses from Thorium and Daughters Contained in Thoriated Welding Electrodes.
March 1980.
Oak Ridge National Lab
ORNL/NUREG/TM-344
OSD GPO: $3.50. NTIS.

Thoriated-tungsten welding electrodes, containing 1 to 2% thoria ($THO_2$) by weight, are potential sources of radiation exposure to members of the general public involved in gas tungsten-arc welding. Therefore, exposure scenarios were developed to estimate potential doses associated with the distribution, use and disposal of these commonly available consumer products. Source terms for both internal and external exposures were estimated on the basis of documented release rates of thorium and daughters from electrodes during welding, of known thoria ($ThO_2$) concentrations in electrodes, and on the basis of estimated production rates of thoron ($^{220}Rn$) in electrodes.

NUREG/CR-1053

Notch Ductility Degradation of Low Alloy Steels with Low-to-Intermediate Neutron Fluence Exposures.
February 1980.
Naval Research Lab
NRL Rpt 8357
ONRR GPO: $2.00. NTIS.

The extent and trend of Charpy-V ($C_v$) notch ductility changes in reactor vessel material with fluences of $1 \times 10^{18}$ n/cm² > 1MeV were investigated with several thick section steel plates and submerged arc weld deposits irradiated at 288° (550°F). The materials were fully representative of reactor vessels now in service and had copper contents ranging from 0.10 to 0.40% and phosphorus contents ranging from 0.008 to 0.20%. Material irradiations were performed in a 2-MW pool reactor. The steels with high radiation sensitivity indicated an onset of notch ductility changes at fluences of $\sim 1.5 \times 10^{18}$ n/cm² by an elevation in the ductile-to-brittle transition temperature. Reductions in upper shelf were not observed at this fluence level but were in the range of 0 to 15% at $\sim 4 \times 10^{18}$ n/cm² and between 15 to 44% at $\sim 8 \times 10^{18}$ n/cm². The data trend suggests a power law relationship of upper shelf reduction to fluence at low-to-intermediate fluences. The $C_v$ transition temperature elevation and upper shelf reduction with irradiation are compared to embrittlement projections by U.S. Nuclear Regulatory Commission Guide 1.99. A limited experimental comparison of radiation effects to dynamic fracture toughness and notch ductility is also presented.

| Report No. | Bibliographic Data |
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NUREG/CR-1060

Activities, Effects and Impacts of the Coal Fuel Cycle for a 1,000 MWE Electric Power Generating Plant.
February 1980.
Teknekron Research, McLean, VA
ONRR GPO: $10.00. NTIS.

Teknekron Research, Inc., under contract to the Nuclear Regulatory Commission, reviewed current literature pertaining to the health, ecological, economic, and social impacts of the coal fuel cycle. The computerized literature search revealed over 3,000 relevant citations. A review of their abstracts resulted in the selection of over 200 documents which were used to prepare this report. This data base was supplemented by publications of the most recent research results from a number of Federal agencies. The result is the summation of the adverse societal impacts of all segments of the coal fuel cycle associated with a 1,000-MWe power plant. This report does not describe in detail the effects and impacts that result from long-term operation of many power plants nationwide or in a particular region of the country, such as the "greenhouse" effect, acid rain, chronic health impacts of trace metals and carcinogens and long-range transport of sulfates. The report does, however, discuss the impacts of local air pollution, mining and transportation accidents, land disturbances, water pollution, and solid waste disposal. Socioeconomic impacts are also discussed.

NUREG/CR-1062

LMFBR Aerosol Release and Transport Program Quarterly Progress Report for April-June 1979.
January 1980.
Oak Ridge National Lab
ORNL/NUREG-TM-354
ONRR GPO: $3.50. NTIS.

This report summarizes progress for the Liquid-Metal Fast Breeder Reactor (LMFBR) Aerosol Release and Transport (ART) Program sponsored by the Division of Reactor Safety Research for the period April-June 1979. This program investigates radionuclide release and transport from LMFBRs for reactor events of severity up to and including hypothetical core-disruptive accidents (HCDAs). Topics discussed include recent capacitor discharge vaporization (CDV) tests in the Fuel Aerosol Simulant Test (FAST)/Containment Research Installation-III (CRI-III) facility, including underwater tests in the FAST vessel and underwater and vacuum tests in the CRI-III vessel; preliminary results from three Nuclear Safety Pilot Plant (NSPP) uranium oxide aerosol tests using the dc plasma metal-oxygen torch as an aerosol generator; results from CRI-II sodium oxide aerosol tests using the liquid-metal-spray sodium oxide aerosol generator; results from the normalization of the spiral-duct centrifuge calibration at different speeds; and the presentation of possible methods for calculating the rise in water temperature at the bubble vapor-liquid interface in FAST water experiments.

NUREG/CR 1063

Performance Testing of Personnel Dosimetry Services: Procedures Manual.
October 1977 - September 1979.
January 1980.
University of Michigan
OSD GPO: $4.75. NTIS.

The Procedures Manual describes the operational conditions of the 2-year pilot study conducted by the University of Michigan of the Health Physics Society Standards Committee (HPSSC) Standard titled, "Criteria for Testing Personnel Dosimetry Performance." The Manual describes source calibrations (which were done or supervised by the National Bureau of Standards), irradiation geometries, quality control, record keeping, data analysis, and methods of receiving, handling, and returning large numbers of dosimeters.

NUREG/CR-1064

Performance Testing of Personnel Dosimetry Services: Final Report of a Two-Year Pilot Study, October 1977 - December 1979.
January 1980.
University of Michigan
OSD GPO: $3.75. NTIS.

The University of Michigan conducted a two-year pilot study of the Health Physics Society Standards Committee (HPSSC) Standard, "Criteria for Testing Personnel Dosimetry Performance." Pilot study objectives were: (1) to give processors an opportunity to correct any problems that are uncovered; (2) to develop operational and administrative procedures to be used later by a permanent testing laboratory; (3) to determine whether the proposed HPSSC Standard provides an adequate and practical test of dosimetry performance. Fifty-nine dosimetry processors volunteered to submit dosimeters for test irradiations according to the requirements of the HPSSC Standard. These tests satisfied the first objective of the pilot study. The operational and administrative procedures developed during the pilot study are described in the Procedures Manual issued on August 14, 1979, NUREG/CR-1063. The Procedures Manual satisfies the second objective of the pilot study. The Final Report discusses the feasibility of using the HPSSC Standard for a fut re mandatory testing program for personnel dosimetry processors, the third object ve of the pilot study. This report contains the results of the pilot study and recommendations for revisions in the Standard that will make a mandatory testing program useful to regulatory agencies, dosimetry processors, and radiation workers that use personnel dosimeters.

NUREG/CR-1070

An Evaluation of the Unloading Compliance Procedure for J Integral Testing in the Hot Cell.
January 1980.
Westinghouse and Oak Ridge National Lab
ORNL/Sub-7394/1
ONRR GPO: $3.75. NTIS.

A testing program was initiated by the Oak Ridge National Laboratories to develop a procedure for toughness testing of irradiated materials using J integral methods. An evaluation of presently used procedures and improved techniques was completed to increase the reliability of the unloading compliance method for single specimen $J_{Ic}$ determination. Three different heats of pressure vessel steels were used to evaluate the reliability and accuracy of the unloading compliance procedure. A new technique for improved sensitivity of displacement measurements was found to be successful on large 4T specimens. This technique consisted of using a small electrical linear actuator to continuously rezero a 0.635 mm (.025 inch) LVDT. The effects of specimen geometry on both $J_{Ic}$ and dJ/da were evaluated using various specimen sizes and varying amounts of side grooving.

NUREG/CR-1077

Precharacterization Report for Instrumented Nuclear Fuel Assembly IFA-513.
March 1980.
Pacific Northwest Lab
PNL-3152
ONRR GPO: $4.25. NTIS.

This report is a resource document covering the rationale, design, fabrication, and preirradiation characterization of instrumented fuel assembly IFA-513. This assembly is being irradiated in the Halden Boiling Water Reactor in Norway as part of the Verification of Steady-State Codes Program conducted by Pacific Northwest Laboratory and sponsored by the Fuel Behavior Research Branch of the U.S. Nuclear Regulatory Commission. Data from this assembly will be used to better understand light water reactor fuel behavior under normal operating conditions.

NUREG/CR-1078

An Investigation of the Distribution and Entrainment of ECC Water Injected into the Upper Plenum.
January 1980.
Dartmouth College
ORES GPO: $3.75. NTIS.

In this report, concerns related to the effectiveness of the injected water in the Westinghouse two-loop reactors during a postulated loss-of-coolant accident are addressed. Interaction of saturated water and steam in the presence of upper plenum internal structures is simulated on laboratory scale experimental apparatus using air and water as the working media. The tests were designed to explore primarily the hydrodynamic interactions. In the tests, the mechan sm that limits the water from reaching the upper core plate is deduced to be one of water droplet entrainment in air. Flooding at the upper core plate was not observed to take place. For the operating reactors, it is felt, however, that due to significant condensation of steam in the subcooled water jets, the role of entrainment and flooding may be reversed.

| Report No. | Bibliographic Data |
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| NUREG/CR-1081 | Characterization of Uranium Tailings Cover Materials for Radon Flux Reduction.<br>March 1980.<br>Ford, Bacon & Davis Utah, Salt Lake City<br>FBDU-218-2<br>ONRR GPO: $5.00. NTIS.<br><br>The attenuation of radon through uranium cover material is usually described with diffusion theory expressions. One of the main parameters characterizing the diffusion is the diffusion coefficient. Measured values of the diffusion coefficient for several Wyoming and New Mexico soils are presented. An interpretation of various approximations to the diffusion equation is also given. Finally, the diffusion coefficient dependence on moisture is presented and the data are represented by a simple correlation. |
| NUREG/CR-1099 | Depleted Uranium Dioxide Power Flow Through Very Small Openings.<br>February 1980.<br>Battelle Pacific Northwest Lab<br>PNL-3177<br>ONRR GPO: $6.00. NITS.<br><br>The objective of the report is to develop experimental data that will be used to formulate calculational techniques to assess the potential powder passage through very small openings in shipping containers faulted in an accident. Results of experiments are presented that measured the leakage of depleted uranium dioxide (DUO) powder through microorifices in a vessel approximately the same dimentions as a plutonium dioxide shipping container. Leaks were measured as a function of upstream pressure (15 psig to 1000 psig) and above and below the static powder level. An equation was developed to predict powder transmission from leaks using $\ell n\ (A\sqrt{P}) <$ 10.5 (A = area; P = pressure). Maximum DUO transmission values were calculated for leaks where $\ell n\ (A\sqrt{P}) < 10.5$. |
| NUREG/CR-1103 | Behavior of a Nine-Rod Fuel Assembly during Power-Cooling-Mismatch Conditions - Results of Test PCM-5.<br>January 1980.<br>EG & G Idaho<br>EGG-2002<br>ONRR GPO: $3.75. NITS.<br><br>This report presents the results of power-cooling-mismatch (PCM) Test PCM-5, an experiment designed to investigate the behavior of pressurized water reactor type fuel rods during postulated PCM accident conditions. The primary objective of the test was to subject a nine-rod fuel bundle to stable film boiling conditions to investigate the potential for rod-to-rod interactions. In addition, the previously established PCM single-rod data base was assessed for multiple-rod geometries. |
| NUREG/CR-1104 | Preliminary Analysis of the Containment Failure Probability by Steam Explosions Following a Hypothetical Core Meltdown in an LWR.<br>March 1980.<br>Sandia Lab, Albuquerque<br>SAND79-2002<br>ONRR GPO: $3.50. NTIS.<br><br>The Reactor Safety Study assessed the probability of containment failure via a steam explosion during a hypothetical core meltdown accident to be 0.01. Large uncertainty bands were attached to these probabilities and research has continued to reduce the uncertainty. This present work, a result of this subsequent research, has examined the heat transfer and structural models needed to analyze the transient and what is a reasonable course of events. Preliminary results indicate (1) that reactor vessel failure by spallation and missile generation due to a steam explosion is not likely; (2) during the expansion phase of the explosion, the work potential could be significantly reduced (factor of 1.5 to 5.0) due to heat transfer between the coolant vapor and cold liquid; (3) a large mass missile generated from the explosion-driven water impact may be precluded because of the upper internal structure. However, a small mass missile (e.g., control rod drive assembly) cannot be precluded at this time although rod buckling could mitigate the effects. |

| Report No. | Bibliographic Data |
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**NUREG/CR-1120, Vol.1**

Seismic Safety Margins Research Program (Phase I) Quarterly Progress Report No. 5.
January 1980.
Lawrence Livermore Lab
OGRSR GPO: $3.25. NTIS.

This document is a Quarterly Progress Report on the Seismic Safety Margins Research Program (SSMRP). The report gives a general description of the program, together with financial summaries and individual project details. Each project is summarized to show accomplishments, schedules, milestones and completion dates, budget and expenditures, and any concerns that may affect the project.

**NUREG/CR-1121**

Quarterly Process Report on Blowdown Heat Transfer Separate-Effects Program for July-September 1979.
January 1980.
Oak Ridge National Lab
ORNL/NUREG/TM-363
ONRR GPO: $1.75. NTIS.

Bundle 3 was installed into the Thermal-Hydraulic Test Facility (THTF) test section on June 26, 1979. This bundle is an 8 x 8 array of electric fuel pin simulators 9.5 mm in diameter. Each of the fuel in simulators has a flat power profile 3.6 m long. The rod-to-rod pitch is representative of a 17 x 17 pressurized-water reactor subassembly. Each fuel pin simulator has 16 thermocouples in groups of four (one centered and three aximuthal) at each of four levels. The THTF was modified to have a separate external downcomer which includes two new instrumented spool pieces. The current status of the THFT is called THTF-Mod II. Operational testing of the system will be completed with the first isothermal blowdown in November 1979. THTF-Mod II contains approximately 1200 measurement locations; the PDP-11/34-based data acquisition system has a scanning rate of 40,000 channels per second.

**NUREG/CR-1126**

Properties of Radioactive Wastes and Waste Containers.
March 1980.
Brookhaven National Lab
BNL/NUREG/CR-51101
ONRR GPO: $2.00. NTIS.

Methods for extrapolation experimental leach data to allow predictions for radionuclide release from full-scale waste forms by diffusional mass transport are discussed. In particular, finite and semi-infinite modeling techniques are considered with their respective strengths and limitations. A finite model with more general boundary conditions is presented which can take into account leachant dependent release rates. Modified IAEA leach test data is shown for the release of cobalt-60 from portland type II cement-boric acid concentrate waste forms. A low cobalt-60 release rate was observed, apparently due to absorption processes within the cement matrix. Expansion has been observed in portland cement waste forms incorporating ion exchange bead resin waste. Cation exchange resin swelling and shrinkage was found to be dependent upon the absorbed species.

**NUREG/CR-1127**

Fractographic and Microstructural Analyais of Stress Corrosion Cracking of A533 Grade B Class 1 Plate and A508 Class 2 Forging in Pressurized Reactor-Grade Water at 93°C.
February 1980.
Naval Research Lab
NRL Memo Rpt 4121
ONRR GPO: $1.75. NTIS.

Stress corrosion cracking (SCC) studies were conducted in two commonly used pressure vessel steels: A533 Grade B Class 1 (A533-B-1) plate and an equivalent A508 Class 2 (A508-2) forging. The purpose of these studies was to determine the response of the materials in simulated pressurized water reactor environment. A hydrogen-assisted cracking model has been proposed to explain the experimental results on A508-2 forging. Produced by a cathodic reaction and aided by the stress fields, hydrogen diffuses ahead of the crack tip to the inclusion sites. This causes a preferential decohesion at the inclusion matrix interfaces and subsequent cracking along inclusion bands. The absence of stress corrosion cracking in A533-B-1 plate tested under identical experimental conditions is mainly due to fewer inclusion and carbide particles and to the more refined bainitic microstructure of this steel. This type of microstructure is less susceptible to hydrogen-assisted cracking than the mainly pearlitic microstructure found in A508-2 forging.

| Report No. | Bibliographic Data |
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NUREG/CR-1128

Structural Integrity of Water Reactor Pressure Boundary Components. Annual Report - Fiscal Year 1979.
February 1980.
Naval Research Lab
NRL Memo Rpt 4122
ONRR GPO: $4.25. NTIS.

This report describes research progress for Fiscal Year 1979 in a continuing program to characterize material properties performance with respect to structural integrity of light-water reactor pressure boundary components. Progress under fracture mechanics investigations includes the first J-R curves from irradiated A533-B weld deposit. A dynamic finite element analysis was also performed to verify the NRL experiment procedure for dynamic fracture toughness, $K_{Id}$. Work in corrosion fatigue has investigated the effects of waveform and temperature on cyclic crack growth in reactor vessel steel; a hydrogen embrittlement model has been proposed. Research in radiation sensitivity has characterized the notch ductility of vessel steels at low fluence. Also investigated was the postirradiation notch ductility of vessels in a coordinated IAEA program. The effects of postirradiation annealing and reirradiation are described in terms of Charpy V-notch ductility and J-R curves. In addition, a survey of embrittlement recovery by postirradiation heat treatment has been prepared. Abstracts of reports prepared under this program in FY79 are also included.

NUREG/CR-1132

Survey of Potential Light Water Reactor Fuel Rod Failure Mechanisms and Damage Limits.
March 1980.
Pacific Northwest Lab
PNL-2787
ONRR GPO: $2.50. NTIS.

This report documents the findings and conclusions of a survey to evaluate current information applicable to the development of fuel rod damage and failure limits for light-water-reactor fuel elements. The survey includes a review of past fuel failures, and identifies potential damage and failure mechanisms for both steady-state operating conditions and postulated accident events. Possible relationships between the various damage and failure mechanisms are also proposed. The report identifies limiting criteria where possible, but concludes that sufficient data are not currently available in many important areas.

NUREG/CR-1136

High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research Quarterly Progress Report, July 1 - September 30, 1979.
January 1980.
Oak Ridge National Lab
ORNL/NUREG/TM-366
ONRR GPO: $1.75. NTIS.

Further development work was done on the ORTAP and BLAST codes. A new and improved model of the Fort St. Vrain (FSV) reactor turbine-generator plant (ORTURB) was developed for use both as a stand-alone code and as a part of the ORTAP system code. Additional work was done on FSV licensing questions. The intermediate heat transfer experiment for investigating FSV upper-plenum reverse-flow plumes was assembled and checked, and an online computer was set up to acquire and analyze the data.

NUREG/CR-1138

Diffusion and Exhalation of Radon from Uranium Tailings.
March 1980.
Pacific Northwest Lab
PNL-3207
ONRR GPO: $4.50. NTIS.

The objectives of this program were to develop and apply an absolute method for determining randon emissions from uranium tailings. Briefly, $^{226}Ra$ and $^{222}Rn$ ($^{214}Pb$) concentration gradients as a function of depth were measured in situ by gamma-ray spectrometry, which was accomplished by lowering a calibrated intrinsic germanium detector to discrete levels within a sealed and cased test well hole and accumulating the gamma-ray spectrum with a multichannel analyzer. Differences between the vertical distributions of radium and radon were used to calculate a radon diffusion coefficient, the fraction of emanating radon and the flux of radon across the tailings-air interface. A diffusional model was developed that accounted for the nonuniform radium concentrations that occur with depth in tailings piles.

| Report No. | Bibliographic Data |
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| NUREG/CR-1139 | An Evaluation of the In-Pile Pressure Data from Instrumented Fuel Assemblies IFA-431 and IFA-432.<br>March 1980.<br>Pacific Northwest Lab<br>PNL-3206<br>ONRR GPO: $3.75. NTIS.<br><br>This report includes results of the examination of the in-pile pressure data from instrumented test assemblies IFA-431 and 432. The pressure data have been used to estimate the fission gas release fraction as a function of the fuel burnup. Included are comparisons of the estimated release functions and those predicted by three fission gas release models using the experimental temperature histories of the fuel rods. These comparisons show that fuel temperature is the primary factor in determining fission gas release and that burnup-enhanced fission gas release is not important in $UO_2$ fuels irradiated to 1700 GJ/kgU (20,000 MWd/MTM). |
| NUREG/CR-1140 Vol.1 | Aggregated Systems Model of Nuclear Safeguards Executive Summary, Vol. 1.<br>February 1980.<br>Lawrence Livermore Lab<br>UCRL-52712, Vol. 1<br>ONRR GPO: $2.25. NTIS.<br><br>When setting the performance criteria for systems which safeguard special nuclear material (SNM), decision makers must consider chracteristics of the adversaries who attempt to divert SNM, safeguards responses to these attempts, costs of safeguards systems, and the consequences of diverted SNM. This report describes an Aggregated Systems Model, which is designed to assist decision makers in integrating and evaluating these diverse factors consistently. The report summarizes the results obtained from applying the model to safeguards decision making in areas such as the hardware or procedures required, substitution of electronics for human safeguards, and overall performance criteria for safeguards systems. New performance criterion designed to measure how safeguards systems deter adversary attempts are also described. |
| NUREG/CR-1140 Vol.2 | Aggregated Systems Model of Nuclear Safeguards, Vol. II.<br>February 1980.<br>Lawrence Livermore Lab<br>UCRL-52712, Vol. 2<br>ONRR GPO: $4.25. NTIS.<br><br>When setting the performance criteria for systems that safeguard special nuclear material (SNM), decision makers must consider characteristics of the adversaries who attempt to divert SNM, safeguards responses to these attempts, costs of safeguards systems, and the consequences of diverted SNM. This report describes an Aggregated Systems Model that is designed to assist decision makers in integrating and evaluating these diverse factors consistently. The report summarizes the results obtained from applying the procedures required, substitution of electronics for human safeguards, and overall performance criteria for safeguards systems. New performance criteria designed to measure how safeguards systems deter adversary attempts are also described. |
| NUREG/CR-1142 | Remote Response Mechanisms.<br>April 1980.<br>Benard Johnson, Houston<br>Y/DS-99<br>OSD GPO: $3.75. NTIS.<br><br>This report on remote response mechanisms for nuclear facility security systems was prepared for use in the Safeguard Regulatory Program of the United States Nuclear Regulatory Commission. The report includes information that will be useful to those responsible for the planning, design, and implementation of physical security systems. It discusses mechanisms that may be controlled remotely to extend the period of time during which security forces can implement effective responses to intruders. Techniques and mechanisms designed to reduce the intruder's capability to perform tasks and/or increase the tasks necessary to accomplish the intruder's objective are described and their characteristics discussed. The importance of the synergism between remote response mechanisms and other components (especially physical barriers) of safeguard system is emphasized. The methods reviewed in this report are those which might be considered for application. The more exotic and futuristic techniques have been |

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included for completeness and to serve systems designers the effort of repeating the literature search required to evaluate them. Inclusion of a particular method does not necessarily represent recommendation or approval for use.

NUREG/CR-1145

Experiment Data Report for LOFT Nuclear Small Break Experiment L3-1.
March 1980.
EG & G Idaho
EGG-2007
ONRR GPO: $5.50. NTIS.

Experiment L3-1 was the first small break loss-of-coolant experiment conducted at the Loss-of-Fluid-Test (LOFT) facility with the nuclear core at power. The primary objectives of the experiment were to obtain date for analytical code assessment and to further understand the thermal-hydraulic behavior which occurs during a postulated loss-of-coolant accident in a presurized water reactor. Other objectives of the experiment were to determine emergency core cooling system performance, determine any unexpected thresholds or events, determine how effectively typical process instrumentation indicated the true system condition, and define variations in system design or plant operation that could mitigate small break transient phenomena. The experiment successfully accomplished the objectives.

NUREG/CR-1147

Econometric Model for the Disaggregation of State-Level Electricity Demand Forecasts to the Service Area.
February 1980.
Oak Ridge National Lab
ORNL/NUREG/TM-359
ONRR GPO: $4.00. NTIS.

An econometric model for the disaggregation of state-level electricity demand forecasts to the service area is presented. Based on demand models for the service area and the remainder of the state in which it is located, a model which explains the service area's share of the state's demand is developed and estimated for six service areas using econometric techniques. The share is then forecasted and combined with the forecasts for state demand presented in Regional Econometric Model for Forecasting Electricity Demand by Sector and by State to obtain service area forecasts to 1990.

NUREG/CR-1149

Emergency Response Scenarios for Transportation Accidents Involving Radioactive Materials.
March 1980.
Sandia Lab, Albuquerque
SAND79-2017
ONRR GPO: $2.50. NTIS.

To assist emergency response personnel in evaluating the unique hazards presented to them by transportation accidents involving radioactive materials, this study examines several limiting scenarios for such accidents in terms of their potential radiological impacts. The radiological consequences of five accident scenarios are analyzed in terms of dose and dose rate as a function of distance, and potential health effects. Additionally, the emergency response implications of possible sabotage events are discussed.

NUREG/CR-1150

Impact of Dispersion Parameters on Calculated Reactor Accident Consequences.
March 1980.
Sandia Lab, Albuquerque
SAND79-2081
ONRR GPO: $2.50. NTIS.

Numerous recommendations have been made concerning the choice of appropriate disper[illegible] parameters of pollutants released from a point source. A series of calculations has been performed to determine the impact of these recommendations on the calculated consequences of large reactor accidents. Results of those calculations indicate that predicted accident consequences are in general not strongly sensitive to assumed dispersion parameters. However, adjustment of the Pasquill-Gifford dispersion curves for release duration and surface roughness, or use of the Vogt rather than Pasquill-Gifford curves, can have a significant impact on calculated early fatalities.

| Report No. | Bibliographic Data |
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| NUREG/CR-1151 | Infiltration of Particulate Matter into Buildings.<br>March 1980.<br>Sandia Lab, Albuquerque<br>SAND79-2079<br>ONRR GPO: $1.75. NTIS.<br><br>This study was undertaken to estimate experimentally the protection afforded by being indoors against inhalation of particulates of outdoor origin. Results of the study suggest average protection factors for homes of approximately 4 for large (~5 micron) particles and 2 for submicron particles. Protection factors are somewhat higher for larger building types. As part of Sandia's FY77-78 reactor accident emergency response planning program, a ventilating model was developed to estimate the potential effectiveness of sheltering in reducing the dose due to inhaled radionuclides. Use of "best estimate" values for the model parameters suggested a protection factor of 1.5. |
| NUREG/CR-1158 | Tensile Properties of Irradiated and Unirradiated Welds of A533 Steel Plate and A508 Forgings.<br>February 1980.<br>Hanford Engineering Development Lab<br>HEDL/TME-79-51<br>ONRR GPO: $2.25. NTIS.<br><br>The tensile properties of welds of base metals ASTM A533, Grade B, Class 1 steel plate and ASTM A508, Class 1 forgings were evaluated in irradiated (3 to 21 x $10^{18}$ n/cm$^2$) and unirradiated conditions. Yield strength and ultimate strength both increased with increasing fluence, while small ductility losses were generally independent of fluence. Yield strength was found to be more sensitive to irradiation than ultimate strength for all welds. The strength and ductility responses to irradiation varied between the weld materials. These variations were attributed to differences in chemical constituents of the welds. |
| NUREG/CR-1160 | LMFBR Safety 7. Review of Current Issues and Bibliography of Literature (1978).<br>January 1980<br>Oak Ridge National Lab<br>ORNL/NUREG/TM-367<br>ONRR GPO: $3.75. NTIS.<br><br>This report discusses the current status of liquid-metal fast breeder reactor (LMFBR) development. Selected bibliographic information on LMFBRs relative to the development and safety of the breeder reactor is presented for the year 1978. The bibliography consists of approximately 127 abstracts covering research and development, operating experience, and design practices. Keyword, author, and permuted-title indexes are included for completeness. |
| NUREG/CR-1164 | Light-Water-Reactor Safety Research Program: Quarterly Progress Report.<br>April - June 1979.<br>January 1980.<br>Argonne National Lab<br>ANL-79-81<br>ONRR GPO: $2.25. NTIS.<br><br>This progress report summarized the Argonne National Laboratory work performed during April, May, and June 1979 on water-reactor-safety problems. The following research and development areas are covered: (1) Loss-of-Coolant Accident Research: Heat Transfer and Fluid Dynamics; (2) Transient Fuel Response and Fission-Product Release Program; and (3) Mechanical Properties of Zircaloy Containing Oxygen. |
| NUREG/CR-1169, Vol.2 | Safeguard Vulnerability Analysis Program (SVAP) Data-Gathering Handbook. Volume II.<br>January 1980.<br>Lawrence Livermore National Lab<br>UCRL-52731, Vol. 2<br>ONRR GPO: $3.50. NTIS.<br><br>Blank data collection forms for Safeguard Vulnerability Analysis Program (SVAP) are provided to the Nuclear Regulatory Commission. These forms may be reproduced by the NRC as necessary in conducting SVAP analyses at nuclear facilities. |

Report No. | Bibliographic Data

NUREG/CR-1171

The Effect of Crack Length and Side Grooves on the Ductile Fracture Toughness Properties of ASTM A533 Steel.
February 1980.
Hanford Engineering Development Lab
ORNL/SUB-79/50917/3
ORES GPO: $3.75. NTIS.

The ductile fracture toughness, $J_{Ic}$, and tearing modulus, T, of ASTM A533, Grade B, Class 1 steel were evaluated by the unloading compliance method for determining J-R curves. These properties were measured for a matrix of IT specimens in which the relative crack length, a/W, and the depth of side grooving were systematically varied to determine their individual effects. In addition, the applicability of an LVDT extensometer system was investigated for use in the unloading compliance method for J-R curve determination.

NUREG/CR-1172

Delayed Beta- and Gamma-Ray Production due to Thermal-Neutron Fission of 239Pu: Tabular Graphical Spectral Distributions for Times After Fission Between 2 and 1400 Sec.
January 1980.
Oak Ridge National Lab
ORNL/NUREG-66
ONRR GPO: $5.00. NTIS.

Fission-product decay energy-release rates have been measured for thermal-neutron fission ($^{239}Pu$). Samples of mass 1 and 5 ugm were irradiated for 1 to 100 sec using the fast pneumatic-tube facility at the Oak Ridge Research Reactor. The resulting beta- and gamma-ray emissions were separately counted for times-after-fission between 2 and 14,000 sec, giving spectral distributions $N(E_g)$ vs $E_g$ and $N(E_b)$ vs $E_b$. The gamma-ray spectra were obtained using a NaI detector, and the beta-ray spectra were obtained using NE-110 detector with an anticoincidence mantle. The raw data were unfolded to provide spectral distributions of moderate resolution. These distributions are given in graphical and tabular form as differential spectral intensity I(E) ($MeV^{-1}$ $fission^{-1}$) averaged over gamma-ray energy intervals ranging from 10 keV for $E_g$ < 0.18 MeV to 100 keV for $E_g$ > 6.8 MeV, and beta-ray energy intervals ranging from 20 keV for $E_B$ < 0.25 MeV to 160 keV for $E_B$ < 6.4 MeV. Counting time intervals ranged from 1 sec for times-after-fission ($t_w$) □6 sec to 4000 sec for $t_w \sim 10^4$ sec. For comparisons the graphical representations show calculated spectra obtained using the CINDER-10 summation code and the ENDF/B-IV fission yield and decay scheme data base.

NUREG/CR-1178

Sabres II: Code Description and Users Manual.
March 1980.
Sandia Lab, Livermore
SAND79-8268
ONRR GPO: $4.75. NTIS.

SABRES II is an individual resolution computer simulation of combat between two small groups using small arms weapons. The model is designed to simulate a battle between the protective force of a special nuclear material (SNM) transportation convoy and an adversary force attempting to steal the material. The usefulness of the model lies in its ability to compare, in a consistent manner, the relative value of the variations in the characteristics of an SNM transportation physical protection system. Two versions of the SABRES II model are described. An interactive version, allowing the user to control the movement of all combatants, aids in the generation of plans and tactics to be followed by each force during the conflict. A Monte Carlo version of the model, using the interactive plans and tactics derived from multiple simulations of the scenario, generates statistical analysis of the simulated conflict.

NUREG/CR-1182

Process Notebook for Aquatic Ecosystem Simulation.
January 1980.
University of Washington
ONRR GPO: $5.50. NTIS.

This notebook contains a detailed comparison of 14 models of fish growth, energetics, population dynamics and feeding. It is a basic document for the evaluation of these models' usefulness for impact assessment. Model equations are categorized into 18 subprocesses comprising the major processes of consumption, predation, metabolic processes, growth, fecundity and mortality. The model equations are compared in a standard notation and the equation rationales are considered and put into a historical framework with historical precedence charts. Model parameters are computed in standard units and data sources and techniques used for parameter estimation are identified. A translator compares standard notation with the notation used in the models. For the first time, fish models are arrayed with their assumptions laid bare and their parameter values compared, allowing elucidation of model differences and evaluation of model behavior and data needs by using the process notebook as a base for further simulation comparison.

NUREG/CR-1186

A Study of New England Seismicity with Emphasis on Massachusetts and New Hampshire.
January 1980.
MIT
ONRR GPO: $5.00. NTIS.

Teleseismic P-wave arrival times recorded by the North-Eastern Seismic Network are used to invert for lateral crust and upper mantle structure to depths of 500 km. Three-dimensional inversion of the travel time data between two ancient orogenic provinces suggests that structures down to possibly 200 km and greater can be correlated with surficial geologic and tectonic features. This has the important implication that major orogenic belts have effects that reach well into the lithosphere which are stable for extended periods of time, perhaps as long as 1 billion years. The crust beneath the Palaeozoic Appalachian Province is characterized by slightly greater thicknesses and lower average velocities than that of the Precambrian Grenville Province. The higher average velocities associated with the Grenville province extend to depths of 200 km and appear to be maximum beneath the Adirondack dome. A relatively low velocity anomaly extending to depths in excess of 200 km and dipping to the northwest shows a spatial correlation with the Bronson Hill - Boundary Mountain Anticlinorium in central New Hampshire and Maine. These structures occupy the sites of a complex series of island arc sequences last active in Early Devonian time prior to the Acadian orogeny. This low velocity region may represent subducted oceanic lithosphere which has undergone post-orogenic radioactive heating.

NUREG/CR-1191

Conformation of the Original Qualification Test for Electrical Connectors Used at Browns Ferry Nuclear Power Plant Unit 3.
March 1980.
Sandia Lab, Albuquerque
SAND79-2311
ONRR GPO: $1.75. NTIS.

Following the Union of Concerned Scientists' petition of November 4, 1977, the NRC Commissioners directed the NRC staff to: "Arrange for a repeat of the tests to obtain data for the verification of current methodology for environmental qualification of electrical components. These tests should be performed with a representative sample of commercially available electrical connectors qualified in accordance with IEEE-323 (1974) and in use in nuclear power reactor safety systems. When available, the test results are to be promptly provided to the Commission." The NRC staff interpreted this action to be aimed at providing information on the methodology of qualifiction testing using electrical connectors which meet the provisions of IEEE-323.

NUREG/CR-1193

Transportation of Radioactive Material in Illinois. June 6, 1978-June 6, 1979.
March 1980.
Dept. of Public Health, Illinois
OSP GPO: $3.50. NTIS.

This report describes the second-year study conducted by the State of Illinois on the transportation of radioactive material into, with and through the State from June 6, 1978 to June 6, 1979. During this period, sixteen State Police troopers monitored from time to time the main highways in vehicles equipped with radiation detection instrumentation. The data collected during the monitoring process are contained in this report. The data relate to personnel radiation exposures, condition of packages, handling practices, and general adherence to regulations governing the safe transort of radioactive materials. The results of the first year's study, covering the period June 6, 1977 to June 6, 1978, are contained in NUREG/CR-0756.

Report No. | Bibliographic Data

NUREG/CR-1199

Subcompartment Analysis Procedures.
February 1980.
Los Alamos Scientific Lab
LA-8169-MS
ONRR GPO: $3.75. NTIS.

Procedures for the performance of nuclear reactor subcompartment analysis are presented. The purpose of this presentation is to normalize the analysis procedures and provide a standard approach for such analyses. As a result, differences in the manner of performing subcompartment analyses hopefully will be minimized and more readily understood and evaluated by others. The procedures were developed within the constraint of current code capability for the performance of such analyses and the current USNRC guidelines. A wide gamut of the effect of input and modeling variations on calculated forces and moments were studied. The studies were primarily for representative reactor cavity geometries. The COMPARE subcompartment analysis code was used for the studies.

NUREG/CR-1200

Accident Delineation and Evaluation of the High-Temperature Gas-Cooled Reactor System Concepts.
February 1980.
Las Alamos Scientific Lab
LA-8170-MS
ONRR GPO: $6.50. NTIS.

A methodology of accident delineation and analysis is developed for application to an evaluation of the conceptual design of a high-temperature gas-cooled reactor (HTGR). The conceptual design of the 3000-MW(t) HTGR is studied and probabilities of possible accident sequences are provided. Latent hazard indices are developed for the accident sequences to identify quantitatively the sequences having the greatest potential impact on the public safety.

NUREG/CR-1201

Nuclear Reactor Safety. Quarterly Progress Report. July - September 1979.
February 1980.
Los Alamos Scientific Lab
LA-8171-MS
ONRR GPO: $6.50. NTIS.

This quarterly report summarizes technical progress from a continuing nuclear reactor safety research program conducted at the Los Alamos Scientific Laboratory (LASL). The reporting period is from July 1 to September 30, 1979. This research effort concentrates on providing an accurate and detailed understanding of the response of nuclear reactor systems to a broad range of postulated accident conditions. The report is mainly organized according to reactor type. Major sections deal with Light-Water Reactors (LWRs), Liquid Metal Fast Breeder Reactors (LMFBRs), High-Temperature Gas-Cooled Reactors (HTGRs), and Gas-Cooled Fast Reactors (GCFRs).

NUREG/CR-1202

COPS - A Model for Estimating Local Law Enforcement Agent Availability.
March 1980.
Sandia Lab, Livermore
SAND78-8237
ONRR GPO: $2.25. NTIS.

One element in the Physical Protection of Nuclear Material in Transient Program is a determination of the number of local law enforcement agents that might be available to support the transportation safeguards system. A computer model, COPS, has been developed to help address this problem. The model provides an inexpensive means for identifying areas along a route where the police coverage may be relatively low. It may also be used to compare alternate routes between locations and help identify those routes with better police coverage. COPS has been used to analyze several routes used for the transportation of highly enriched uranium. Examples of these analyses are presented.

Report No. | Bibliographic Data

NUREG/CR-1203

Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research.
October - December 1979.
March 1980.
EG & G Idaho
EGG-2012
ONRR GPO: $2.25. NTIS.

Water reactor research performed by EG&G Idaho, Inc., during October through December 1979 is reported. The Semiscale Program conducted the first three tests in the Semiscale small-break test series to investigate the phenomena that occur in a small break loss-of-coolant experiment. The Loss-of-Fluid Test (LOFT) Experimental Program performed and reported results of the first in a series of small-break tests in the LOFT nuclear test reactor. The Thermal Fuels Behavior Program completed (a) three loss-of-coolant accident (LOCA) tests in the Power Burst Facility, (b) scoping tests with a fission product measurement system, and (c) the first in a series of internal fuel rod fill gas composition tests (with xenon and helium mixtures) in the Halden reactor in Norway. The Code Development and Analysis Program progressed in the development of advanced computer codes, including checkout of the BEACON/MOD3 containment analysis code and development of a capability in the TRAC code for analysis of LOCA transients in a boiling water reactor. The Code Assessment and Applications Program performed calculations in support of the NRC review of the Three Mile Island accident and performed various reactor vendors' small-break analyses. The 3-D Experiment Project made progress in its instrument project for a test facility in Japan.

NUREG/CR-1204

The Design and Construction of a $D_2O$-Moderated 252Cf Source for Calibrating Neutron Personnel Dosimeters Used at Nuclear Power Reactors.
January 1980.
National Bureau of Standards
OSD GPO: $2.00. NTIS.

The major goal was to develop a moderated neutron source whose spectrum simulates the neutron spectrum found at power reactors. The intended use of such a source is to test neutron dosimeter processor performance, as well as to calibrate dosimeters and remmeters. The neutron spectra used as models to develop the moderated 252Cf source was a neutron spectrum measured at the Alabama Power and Light, Farley Nuclear Plant and a calculated spectrum for the Arkansas Power and Light reactor. It was concluded that 15 centimeters of $D_2O$ was the best moderator to simulate a reactor neutron spectrum using a 252Cf source. It somewhat compromises the need for a realistic spectrum on one hand, and m[illegible] size for calibration purposes on the other. An additional virtue of the $D_2O$ moderated spectrum is that it provides a substantial nuetron flux over the whole energy range from ~10 eV to ~5 MeV. Hence, this spectrum would prove generally useful for calibration and processor performance testing of a wide variety of dosimeters and remmeters. The NBS prototype moderated Cf source is a 30-cm inside-diameter stainless steel shell (0.8-mm-thick walls) containing the $D_2O$. This shell is covered with a (removable) 0.5-mm cadmium shell, which may be used to absorb thermal neutrons.

NUREG/CR-1205

Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978.
January 1980.
EG&G Idaho, Inc.
EGG-EA-5044
ONRR GPO: $8.50. NTIS.

This report describes the results of an analysis of nuclear plant pump failures. The data used for this analysis were the Licensee Event Reports (LERs). The LERs are written reports filed with the NRC whenever certain failures or incidences occur concerning nuclear plant safety systems. The pump failures or incidences contained in the LERs were evaluated and categorized as to the type of failure or problem and were used to calculate summary pump failure rate statistics. The report includes a variety of different statistics calculated to highlight or show important failure modes or other failure information. In addition to the quantitative failure rate information, there is also considerable qualitative information tabulated to allow the user to make additional pump failure rate calculations or inferences.

| Report No. | Bibliographic Data |
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| NUREG/CR-1206 | Critical Pathways of Radionuclides to Man from Agro - Ecosystem.<br>January 1980.<br>Savannah River Ecology Lab, Aiken, SC<br>ONRR GPO: $2.00. NTIS. |

Progress has been considerable within the past year. Both plutonium and curium analyses have been completed on the first year's samples of the greenhouse uptake experiment. A manuscript is near completion for submission for publication in the open literature. Results from this experiment have shown: (1) minimal uptake of plutonium by agricultural crops; (2) greater uptake of curium; (3) immobilization of both radionuclides by lime addition; and (4) partial counteraction of the lime immobilization by chelate addition. Also, dose to man from consumption of edible portions of crops grown in these soils is very small. Initiation of the field uranium studies has begun. These studies will allow for an evaluation of the comparative environmental behavior of Pu and U. Winter wheat and rye crops were successfully produced. Additional crops are anticipated for the current year.

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| NUREG/CR-1207 | Microbial Aerosols from Cooling Towers and Cooling Sprays: A Pilot Study.<br>January 1980.<br>Argonne National Lab<br>ANL/ES-83<br>ONRR GPO: $4.00. NTIS. |

In 1975, the U.S. Nuclear Regulatory Commission sponsored a pilot study by Argonne National Laboratory, working with an aerobiological team from the U.S. Army Dugway Proving Ground, to investigate cooling towers and cooling sprays as sources of potential human pathogenic bacteria. Conclusions of the study are: (1) the potential human health hazard from cooling towers using treated sewage effluent (reclaimed water) may not be as great as from those using polluted surface waters; (2) current chlorination practice for bacterial slime control at cooling towers is inadequate to disinfect against potential pathogens; (3) the potential human health hazard from cooling-tower aerosols appears to be greater from opportunistic bacteria than from enteric pathogens; (4) aerosols from cooling sprays will pose little hazard downwind if droplet sizes are similar to those found at the study site; (5) E. coli is probably not a good indicator of cooling-tower microbial hazard; and (6) the bacterium of Legionnaires' disease can find optimum temperature and pH conditions for survival in most cooling-tower environments, although it is unlikely that there is adequate substrate for rapid growth of the organism in cooling towers.

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| NUREG/CR-1213 | ANS 5.4. A Computer Subroutine for Predicting Fission Gas Release.<br>January 1980.<br>Pacific Northwest Lab<br>PNL-3077<br>ONRR GPO: $1.75. NTIS. |

The ANS 5.4 committee developed fission gas release models for stable and radioactive species for both low and high temperatures. This report reproduces these models and presents a FORTRAN coding of the models. The coding comprises subroutine ANS 5.4 which fall in a format suitable for direct insertion into a code like FRAPCON. Also shown are comparisons of model predictions with light water reactor low burnup data and liquid metal fast breeder reactor high burnup data that were used in the model development.

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| NUREG/CR-1214, Vol.1 | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process.<br>February 1980.<br>Mound Facility, Miamisburg, OH<br>MLM-NURG-2532, Vol. 1<br>OSD GPO: $6.00. NTIS. |

The Controllable Unit Approach (CUA) to Material Control is a systematic method to control nuclear materials by measurements. CUA was developed to test the use of performance regulations in controlling nuclear materials. In this study, CUA was applied to a 200-metric-ton mixed-oxide (4% $PuO_2$ in $UO_2$) process scheduled for completion in the 1980's. The performance criterion used for this study was to detect a loss (single or cumulative) of 2 kg of $PuO_2$ from the mixed-oxide process over a two-month inventory period with a detection probability of 97.5%, and to detect the loss within one day of reaching the 2-Kg magnitude.

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NUREG/CR-1214, Vol.2

The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process.
February 1980.
Mound Facility, Miamisburg, OH
MLM-NUREG-2532, Vol. 2
OSD GPO: $6.00. NTIS.

The Controllable Unit Approach (CUA) to Material Control is a systematic method to control nuclear materials by measurements. CUA was developed to test the use of performance regulations in controlling nuclear materials. In this study, CUA was applied to a 200-metric-ton mixed-oxide (4% $PuO_2$ in $UO_2$) process scheduled for completion in the 1980's. The performance criterion used for this study was to detect a loss (single or cumulative) of 2 kg of $PuO_2$ from the mixed-oxide process over a two-month inventory period with a detection probability of 97.5%, and to detect the loss within one day of reaching the 2-Kg magnitude.

NUREG/CR-1215

The Social and Economic Effects of the Accident at Three Mile Island - Findings to Date.
January 1980.
Mountain West Research, Tempe, AZ
ONRR GPO: $3.75. NTIS.

This report covers the social and economic effects of the accident at Three Mile Island during the first 6 months following the accident. A variety of data sources were utilized including published documents and statistics, household surveys, newspaper files, interviews, and other research about the accident. The findings can be grouped into effects on (1) the regional economy, (2) institutions, and (3) individuals. Direct economic effects during the emergency period following the accident were interrupted local production and reduced local income and employment. Losses were conspicuous during the first week of April but subsequently very minor. There is no evidence of any continuing interruption of activity because of the accident. However, there is concern within the business community about the effect of the accident on the continued growth and development of the area. Major institutional effects were a strain on the emergency preparedness network in the area and an increased focus on the issue of the TMI plant by the local populace. Major effects on individuals were the evacuation itself and increased stress during the accident period. For most people, the effects of the accident were short-lived, but for others, the accident has caused a more permanent change in their day-to-day activities.

NUREG/CR-1216

Radioisotopic Composition of Yellowcake: An Estimation of Stack Release Rate.
January 1980.
Argonne National Lab
ANL/ES-84
ONRR GPO: $2.00. NTIS.

Uranium concentrate (yellowcake) composites from four mills (Anaconda, Kerr-McGee, Highland, and Uravan) were analyzed for U-238, U-235, U-234, Th-230, Ra-226 and Pb-210. The ratio of specific activities of U-238 to U-234 in the composites suggested that secular radioactive equilibrium exists in the ore. The average activity ratios in the yellowcake were determined to be $2.7 \times 10^{-3}$ (Th-230/U-238), $5 \times 10^{-4}$ (Ra-226/U-238) and $2 \times 10^{-4}$ (Pb-210/U-238). Based on earlier EPA measurements of the release rates from the stacks, the amount of yellowcake released was determined to be 0.1% of the amount processed.

NUREG/CR-1217

Central Virginia Regional Seismic Network: Crustal Velocity Structure in Central and Southwestern Virginia.
January 1980
Virginia Polytechnic Institute
VPI-SU-77-134-7, 8
ONRR GPO: $5.50. NTIS.

CENTRAL VIRGINIA: A crustal velocity model was derived by analysis of P-wave travel times for seven earthquakes which occurred within or near the Piedmont province of Virginia. In addition, upper crustal P- and S-wave velocities were determined using an 80 km reversed refraction profile. The P-wave velocity of the deep crust was determined using a network of three stations in central Virginia. The preferred crustal velocity structure model consists of two layers. The upper layer is 15 km thick, with P- and S-wave velocities of 6.09 and 3.53 km/sec. The intermediate layer

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has P- and S-wave velocities of 6.5 and 3.79 km/sec. Upper mantle velocities appropriate for the region are 8.18 and 4.73 km/sec for P- and S-waves. Crustal thickness varies from 39 km beneath the Blue Ridge Mountains to a minimum value of 31 km beneath the central Piedmont. SOUTHWESTERN VIRGINIA: Two crustal models are presented as possible interpretations of unreversed refraction profiles and earthquake-phase arrival data. The preferred crustal has a 49 km thickness. The layer velocities (km/sec) and thicknesses (km) are: Vp1 = h1 = 10, h2 = 39, and upper mantle Pn and Sn velocities of 8.18 and 4.79. The alternate model is: Vp1 = 5.63, Vs1 = 3.44, Vp2 = 6.05, Vs2 = 3.52, Vp3 = 5.63, Vs3 = 3.84, h1 = 5.7, h2 = 9, h3 = 36, and upper mantle Pn and Sn velocities of 8.18 km/sec and 4.79. A velocity anisotropy (10%) was measured in the Valley and Ridge province using quarry blast data. The P-wave velocity was 6.03 km/sec parallel to structural trend and 5.45 perpendicular to it.

NUREG/CR-1218

Gravity Dominated Two Phase Flows in Vertical Rod Bundles.
January 1980.
MIT
DRSR GPO: $4.75. NTIS.

The fluid flow configuration through the core during reflood in a Loss-of-Coolant Accident in a reactor is investigated. Both cold leg injection and the cold leg plus upper plenum injection are studied. The core behavior is dependent primarily on whether the core is unflooded, partially flooded, or flooded. This work deals mainly with the unflooded core. For an unflooded core, the flow transition from countercurrent slug flow to countercurrent annular flow is inlet geometry dependent and occurs due to vapor bubble trapping. A method is developed to predict the range of down-flow liquid velocities for which the bubble is trapped. Flow reversal (or dump) through the core occurs if pressure in the upper plenum becomes more than the pressure in the lower plenum. Due to high down flow liquid velocities in flow reversal, the upper plenum is drained of liquid very quickly and an oscillation down flow develops. In most of the cases, a one-dimensional analysis, using an average heat flux, provides reasonably accurate predictions.

NUREG/CR-1219

Analysis of the Three Mile Island Accident and Alternative Sequences.
January 1980.
Battelle Columbus Lab
ONRR GPO: $4.00. NTIS.

Analyses were performed with the MARCH computer code to examine a number of variations in system operation in the TMI accident to evaluate their effect on the extent of core damage. The results indicate that:
(1) The throttling of HPI had a major effect on core damage. If the system had been permitted to operate at high flow, the core would not have been uncovered.
(2) Earlier closure of the block valve would have prevented core damage. An additiona delay of one hour in closing the valve would have resulted in severe core damage and possibly core meltdown.
(3) The delay in operation of the emergency feedwater system had little effect on the extent of core damage. However, a delay of one hour in the delivery of emergency feedwater would probably have resulted in more severe core damage and possibly core meltdown.

In addition to the analysis of alternative accident sequences: a consistent interpretation of the thermal and hydraulic behavior during the first sixteen hours has been developed; potential containment failure modes for accidents involving complete core meltdown have been examined; and the vulnerabil y of different types of containment designs to hydrogen explosion has been evaluatec

NUREG/CR-1220, Vol.1

Further Results on the Subject of Tearing Instability - Vol. I.
January 1980.
Washington University
ONRR GPO: $4.50. NTIS.

Additional results on the subject of tearing instability are presented as a compilation of five technical papers by several authors. Collectively, the papers add to the development of tearing instability as originally presented in NUREG-0311, published in 1977. The papers strengthen the basis of tearing instability theory and illustrate the broad applicability of the underlying J-integral methodology. The second paper also deals with a method for J-R curve determination by load-displacement test record analysis which is of immediate importance to fracture mechanics experimental lists.

| Report No. | Bibliographic Data |
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NUREG/CR-1220, Vol.2

Further Results on the Subject of Tearing Instability - Vol. II.
January 1980.
Washington University
ONRR GPO: $4.75. NTIS.

This report is a continuation of the series on tearing instability starting with NUREG-0311 in 1977 and immediately preceded by Vol. I of this NUREG. Five papers were assembled; all were prepared by A. Zahoor and P. C. Paris to support the practical applicability of tearing instability. They carry the subject from a theoretical discussion of dJ in order to enhance the utility of the tearing modulus, T, through analyses of center-cracked panels under tension, including work-hardening considerations, to approximate analyses of flawed pressure vessels, tested to failure.

NUREG/CR-1221

Tearing Instability Analysis Handbook (Formulas and Curves).
January 1980.
Washington University
ONRR GPO: $4.00. NTIS.

Information in handbook format is presented for tearing instability formulas on typical test configurations. A short introductory section addresses the use of the tearing modulus, T, in assessing crack growth instability. Three configurations are included in this, the initial publication of the handbook f[illegible], namely: center-cracked panel, compact tension specimen, and three-point bend specimen.

NUREG/CR-1222

Techniques of Analysis of Load-Displacement Records by J-Integral Methods.
January 1980.
Washington Unviersity
ORSR GPO: $6.00. NTIS.

Exact methods are used to obtain J from a single load displacement record for different configurations based on dimensional analysis and deformation theory of plasticity to apply the dependence of the load on crack length and displacement. The methods also permit the evaluation of the crack length increment a, and hence complete J-R curves can be constructed. Formulae for $T_{mat}$ and $T_{app}$ are presented as well. Conditions for the existence of the so-called $\eta$-factor are stated, and subsequent simplifications of the general method by its use are shown. Results of analysis of experimental work for different configuration material combinations are presented showing the adequacy of the general methodology.

NUREG/CR-1230

Local Drag Laws in Dispersed Two-Phase Flow.
March 1980.
Argonne National Lab
ANL-79-105
ONRR GPO: $2.25. NTIS.

The present report develops constitutive relations for the drag force for bubbly, droplet, and particulate flows by a unified method. Simple drag-similarity criteria and a mixture-viscosity model are introduced in the analysis. The present drag correlations cover all concentration ranges and wide Reynolds-number ranges, from the Stokes regime up to the Newton's regime or the churn-turbulent-flow regime. This unified and consistent model gives an improved understanding of the mechanisms of interfacial momentum transfer in dispersed two-phase flows. The result can be useful for the predictions of void fractions, interfacial area, particle residence time, and occurrences of flooding or concentration shock waves.

NUREG/CR-1231

Remote Sensing for Detection and Monitoring of Salt Stress On Vegetation: Evaluation and Guidelines. September 10, 1976 - March 30, 1979.
March 1980.
INTERA Environmental Consult., Houston
ONRR GPO: $4.50. NTIS.

A study was conducted over a 3-year period to investigate the utility of various remote sensing techniques for detection and monitoring of salt stress on vegetation. Predictive drift modeling was used for selecting areas which should be monitored around salt or brackish water cooling towers. Experimental vegetation plots with controlled salt mist applications were used to study the relationships between salt deposition, salt stress symptom development, and detectability of the salt stress using remote sensing. Remote sensing techniques were also tested around operating cooling towers. False color infrared (FCIR) aerial photographs gave the best results of the methods tested and areas of salt stress were found to be identifiable in the imagery. Using the FCIR imagery, the salt deposition threshold for detection of stress from NaCl salt on sensitive species was in the range of 0.9 to 1.9 kg/ha-day. Guidelines are presented for the use of FCIR aerial photography as a screening tool for detection of salt stress around salt and brackish water cooling towers.

NUREG/CR-1232

A Three-Dimensional Fluid Finite Element.
January 1980.
MIT
ONRR GPO: $2.00. NTIS.

A three-dimensional fluid finite element compatible with the solid elements in the ADINA finite element program has been developed for the analysis of wave propagation in fluid and fluid-structural systems. The fluid element can model inviscid compressible fluids with constant bulk modulus. Although the final discretized equations of motion are valid for general loading and displacement conditions, the numerical computations only admit relatively small fluid displacements unless mesh rezoning would be used.

NUREG/CR-1235

Respirator Studies for the Nuclear Regulatory Commission. Evaluation and Performance of Open Circuit Breathing Apparatus. Progress Report, October 1977 - September 1978.
February 1980.
Las Alamos Scientific Lab
LA-8188-PR
ONRR GPO: $1.75 NTIS.

The overall performance and protection factors provided by 12 NIOSH approved, __ minute duration, self-contained breathing apparatus were determined while the respirators were worn by a panel of anthropometrically selected test subjects. Demand-type units provide much lower protection factors than do pressure-demand types. Observations on facepiece pressure, sound level of end-of-life alarms, weight, and comfort are also recorded.

NUREG/CR-1236

Fuel Swelling due to Retained Fission Gas in Molten Fuel During High Temperature Transients.
March 1980.
EG & G Idaho
EGG-2014
ONRR GPO: $3.50. NTIS.

The behavior of light-water-reactor fuel elements is being studied under postulated accident conditions. As part of this program, pressurized-water-reactor-type fuel rods are tested under power-cooling-mismatch (PCM) conditions in the Power Burst Facility (PBF) to study fuel rod behavior of both previously irradiated and unirradiated rods under film boiling conditions. During these integral in-reactor experiments, the fuel rod diameter increased in the film boiling region to a greater extent for previously irradiated rods than for unirradiated rods. The purpose of the study described in this report was to investigate and assess the mechanisms and causes of fuel swelling in irradiated rods and to evaluate the applicability of an analytical fission gas behavior computer code such as GRASS-SST. The overall swelling calculated by summing the contribution of the proposed mechanisms and the GRASS-SST computer code calculated values are compared with measured values, showing good agreement.

NUREG/CR-1237

Best-Estimate LOCA Radiation Signature.
June 1980.
Sandia Lab, Albuquerque
SAND79-2143
ONRR GPO: $5.50 NTIS.

The principal objective of this study was to detail the "best estimate" of the time progressive accident radiation releases into the containment building. This study considers an unterminated LOCA condition as the basis for the calculations. The work is significant in that it represents a conservative, but realistic, best estimate of the resultant radiation signature. The effects of the uncertainties in the release fractions as well as variations in the time dependence of the releases were investigated; as might be expected, there is a wide uncertainty band on the best-estimate releases. Section 6 of the Part III report summarizes specific conclusions developed in the course of the study.

NUREG/CR-1247

Stably Stratified Building Wakes. January 1976 - July 1979.
January 1980.
Colorado State University
ONRR GPO: $5.00 NTIS.

The velocity and temperature wake behind an isolated building placed in a stably stratified turbulent boundary layer has been investigated utilizing wind tunnel tests and mathematical analysis. The mean velocity and mean temperature decrease but turbulence intensity and temperature fluctuation intensity increase as a result of the momentum wake. However, the vortex wake increase mean velocity and mean temperature and decreases turbulence intensity and temperature fluctuation intensity along the centerline of the wake. A wind tunnel study of the wakes behind six surface-mounted rectangular building models in a stably stratified turbulent boundary layer was performed for wind direction perpendicular to one face of the building. Measurements of wind velocity, mean temperature, turbulence intensity, temperature fluctuation intensity, velocity-temperature correlations, spectra of velocity and temperature were measured at a Reynolds number greater than $2.0 \times 10^4$ with and without the buildings in place. An analytical technique for prediction of the temperature field in the wake of a building using the energy equation for turbulence flow was developed and considers momentum and vortex wake effects to determine the mean temperature in the wake of a building.

NUREG/CR-1250, Vol.1

Three Mile Island: A Report to the Commission and to the Public.
January 1980
Rogovin, Stern & Huge, Washington, D.C.
NRC GPO: $5.00. NTIS.

Within weeks of the March 28, 1979 accident at Three Mile Island, the Nuclear Regulatory Commission (NRC) decided to institute a Special Inquiry to review and report on the accident. The principal objectives of the inquiry were to determine what happened and why, to assess the actions of utility and NRC personnel before and during the accident, and to identify deficiencies in the system and areas where further investigation might be warranted. The work of the Special Inquiry was not intended to duplicate the efforts of the President's Commission on the Accident at Three Mile Island. It was designed to enable the NRC to fulfill its regulatory responsibilities by achieving the fullest possible understanding of the accident, both from a technical point of view and from the standpoint of how the NRC's own regulatory processes functioned. Recognizing the potential conflict of interest problems involved if the inquiry were directed and undertaken solely by NRC staff, the Commission in mid-June 1979 contracted with our law firm, Rogovin, Stern & Huge, to conduct the inquiry, specifying that it would have full independence in carrying out the work. Neither the law firm nor any of its members had any prior involvement with nuclear energy issues. This report presents the results of the investigation.

NUREG/CR-1252

Physics of Reactor Safety Quarterly Progress Report. July - September 1979.
February 1980.
Argonne National Lab
ANL-79-98
ONRR GPO: $2.25. NTIS.

This Quarterly progress report summarizes work done during the months of July-September 1979 in Argonne National Laboratory's Applied Physics and Components Technology Divisions for the Division of Reactor Safety Research (RSR) of the U.S. Nuclear Regulatory Commission. The work in the Applied Physics Division includes reports on reactor-safety research and technical coordination of the RSR safety analysis program by members of the Reactor Safety Appraisals Group, Monte Carlo analysis of safety-related critical assembly experiments by members of the Reactor Computations Group, and planning of safety-related critical experiments by members of the Zero Power Reactor (ZPR) Planning and Experiments Group. Work on reactor core thermal-hydraulics was performed in ANL's Components Technology Division.

| Report No. | Bibliographic Data |
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NUREG/CR-1255

SKIRMISH and AMBUSH - Tactical Board Games for Development and Evaluation of Road Transit Physical Protection Systems.
March 1980.
Sandia Lab, Livermore
ONRR GPO: $3.50 NTIS.

Two manual play board games, SKIRMISH and AMBUSH, are described in detail. They are part of a broad-based research effort to develop evaluative methodologies for transportation safeguards systems. These games help provide insight into the value of additional vehicles, guards, cargo barriers, equipment, and tactics. The tasks that are executed at any game turn are based on a human interpretation of the current overall situation and on which strategies appear to optimize the chance of success. Due to the strong dependence of the outcome on the sequence of player decisions, the games may prove useful as a training device for a transportation guard force. An advantage over computer-based combat simulation models is that both SKIRMISH and AMBUSH are easily transportable and relatively inexpensive. The development of SKIRMISH (a simplified or introductory version of AMBUSH) is complete. A set of SKIRMISH rules, together with sufficient materials for the assembly of a complete game, are contained in the Appendix. A preliminary version of AMBUSH has also been completed, and the differences between the two games are discussed.

NUREG/CR-1257

Summary of FY79 - Progress on Refill Effects Program - Quarterly Progress Report. July 1 - September 30, 1979.
March 1980.
Creare, Hanover, NH
Creare TIV-312
NRC GPO: $2.00 NTIS.

This Quarterly Progress Report reviews progress on the Creare Refill Effects Program during the year FY79 and describes our plans for FY80. As studies of countercurrent flow and superheated wall phenomena were completed in FY78, so have studies of lower plenum voiding and condensation-induced transients been completed in FY79. Design and demonstration of a two-phase flow topography instrumentation system was also completed in FY79 by the development, testing and refinement of numerical and graphic data analysis techniques. Analysis and experiments on flashing transients and development of an integrated model of vessel refill continue in FY80.

NUREG/CR-1258, Vol.1

Inspection Methods for Physical Protection Project: Quarterly Report, September-November 1979.
February 1980.
Lawrence Livermore National Lab
UCID-18123-79-3
ONRR GPO: $1.75. NTIS.

This is the third quarterly report to the U.S. Nuclear Regulatory Commission (NRC) of progress at Lawrence Livermore Laboratory in the Inspection Methods for Physical Protection project. Besides presenting the activities and findings of the project's third quarter, this report details additional changes in the tasks and deliverables as requested by the NRC Offices of Nuclear Regulatory Research (RES) and Inspection and Enforcement (IE).

NUREG/CR-1259

The Correlation of Response Spectral Amplitudes with Seismic Intensity.
February 1980.
Computer Sciences Corp, Falls Church, VA
ORES GPO: $4.00. NTIS.

Statistical correlations relating response spectral amplitudes to seismic intensity are derived and shown to be most consistent over selected period bands. In general, these correlations possess lower levels of statistical uncertainty than those incorporating peak amplitude measurements. An analysis of individual effects listed under a general intensity level also indicates a strong dependence on the frequency content of the recorded seismic signal. The correlations were shown to be relatively insensitive to earthquake magnitude, epicentral distance and local site geology. A comparison of the relationships with spectral data not used in the analysis indicates that the model results in fairly good predictions of the spectral amplitude levels over the period bands judged to be most consistent. These comparison studies also indicate that the spectral amplitude/intensity correlation may be independent of the geographic location of the earthquake and, therefore, directly applicable to design problems in the Eastern United States.

Report No. — Bibliographic Data

**NUREG/CR-1263**

Compilation of State Laws and Regulations on the Transportation of Radioactive Materials.
January 1980.
Federal State Reports, Falls Church, VA
OSD GPO: $8.00. NTIS.

This document is a compilation of State laws and regulations that deal with the transport of radioactive materials. Local government initiatives are not included.

**NUREG/CR-1264**

Measurement of Aerosol Deposition Rates in Turbulent Flows.
January 1980.
Battelle Columbus Lab
BMI-2041
ONRR GPO: $2.00. NTIS.

The transport of fission products through reactor primary systems under postulated accident conditions is dependent on the deposition of aerosol particles onto various surfaces. Of particular interest is the case where deposition is from turbulent gas flow since this condition is expected to exist with significant frequency, and available theories for predicting transport under these conditions were largely unverified experimentally. Therefore, in this study, deposition rates of aerosol particles onto pipe walls under turbulent flow conditions were measured. Moderately monodispersed dioctyl phthalate aerosols of six different mean sizes ranging from 0.035 to 1.3 μm were used in gas flows whose Reynolds numbers were controlled at selected values between 5500 and 15,500. The measured results show that among the existing theories, those by Davies and Wells and by Chamberlain predict deposition by Brownian diffusion reasonably well, while the theory of Friedlander and Johnstone is suitable for correlating the data for deposition by inertial impaction. However, these theories alone were not satisfactory for estimating deposition rates in the neighborhood of the minimum aerosol deposition regime where both Brownian diffusional and inertial deposition mechanisms operate simultaneously. An empirical correlation equation which accommodates the measured data in that regime was developed to supplement the existing theories and provide a suitable method for predicting deposition rates.

**NUREG/CR-1267**

FRAP-T4 Best Estimate Sensitivity Study.
February 1980.
EG&G Idaho
ONRR GPO: $3.75. NTIS.

Sensitivity studies of five postulated reactor transient and accident events were performed with the FRAP-T4 transient fuel behavior code. The purpose of the study was to determine the influence of variation in selected input parameters on fuel performance characteristics. The range of the parameters perturbed during the study were supplied by the major U.S. reactor vendors. The five different events examined in the analysis were: (1) locked rotor, (2) control element ejection, (3) steam line break, (4) loss of flow, and (5) turbine trip without bypass.

**NUREG/CR-1270, Vol.1**

Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2.
January 1980.
Essex Corp, Alexandria, VA
NRC GPO: $4.75. NTIS.

This report describes a study of human factors engineering aspects of the Three Mile Island-2 (TMI-2) accident on 28 March 1979. The objective of the study was to evaluate the c[illegible]ntributions, if any, of operator performance and effects on operator perfor[illegible] (1) control room design; (2) operator training; and (3) emergency proc[illegible]e topic of the current report is the degree to which operator errors were, in turn, caused by human factors engineering aspects of the control room design, operator training, and emergency procedures.

| Report No. | Bibliographic Data |
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| NUREG/CR-1270, Vol.2 | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. Appendices, Part 1.<br>January 1980.<br>Essex Corp, Alexandria, VA<br>NRC GPO: $10.00. NTIS.<br><br>This report describes a study of human factors engineering aspects of the Three Mile Island-2 (TMI-2) accident on 28 March 1979. The objective of the study was to evaluate the causal contributions, if any, of operator performance and effects on operator performance of: (1) control room design; (2) operator training; and (3) emergency procedures. The topic of the current report is the degree to which operator errors were, in turn, caused by human factors engineering aspects of the control room design, operator training, and emergency procedures. |
| NUREG/CR-1270, Vol.3 | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. Appendices, Part 2.<br>January 1980.<br>Essex Corp, Alexandria, VA<br>NRC GPO: $5.00. NTIS.<br><br>This report describes a study of human factors engineering aspects of the Three Mile Island-2 (TMI-2) accident on 28 March 1979. The objective of the study was to evaluate the causal contributions, if any, of operator performance and effects on operator performance of: (1) control room design; (2) operator training; and (3) emergency procedures. The topic of the current report is the degree to which operator errors were, in turn, caused by human factors engineering aspects of the control room design, operator training, and emergency procedures. |
| NUREG/CR-1273 | Investigation of Radon-222 Emissions from Underground Uranium Mines. Progress Report 2.<br>March 1980.<br>Battelle Pacific Northwest Lab<br>PNL-3262<br>ONRR GPO: $4.25. NTIS.<br><br>A reliable estimate of radon emissions to the environment from underground uranium mines was obtained through measurements of radon in ventilation exhaust air at 24 uranium mines and estimates of radon release from ore piles and waste piles at mines and in water pumped from mines. Mine characteristics and production data were obtained from interviews with owners of mines representing more than half of 1978 production from underground uranium mines. Ore production and average grade as a composite of 27 mines in the study were furnished by the Grand Junction Office of the Department of Energy. Daily fluctuations of radon with barometric pressure and the statistically significant relationship between radon released per year from a mine and the cumulative ore production at the time of radon measurement were observed. The linear relationship between Ci/yr of radon and cumulative ore accounted for about half the variability. |
| NUREG/CR-1279 | The Insider Threat to Secure Facilities--A Synopsis of Nine Interviews.<br>March 1980.<br>Lawrence Livermore Lab<br>UCRL-52729<br>ONRR GPO: $3.75. NTIS.<br><br>A series of nine interviews was conducted with recognized experts in the field of institutional internal security, for the purpose of gaining insight into the insider threat to nuclear facilities. The results of these interviews included the defining of fundamental problems in internal security, the identification of typical bases of conspiracy formation, and the naming of possible options for improving the quality of nuclear safeguards, both through personnel policies and internal controls. |

| Report No. | Bibliographic Data |
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NUREG/CR-1280

Power Plant Staffing.
January 1980.
Basic Energy Technology Associates
BETA-103
NRC GPO: $4.00. NTIS.

This report outlines the results of a comparative review of current NRC requirements, licensed nuclear power plant practices and the Naval Nuclear Propulsion Program procedures for the selection, training and qualification of personnel involved in nuclear plant operation and maintenance. It also contains recommendations to improve the NRC requirements and civilian practices.

NUREG/CR-1285

Computational Results for a 7-pin Hexagonal Fuel Assembly During a Flow Rundown Transient Using the COMMIX-1A Computer Code.
March 1980.
Argonne National Lab
ANL-CT-80-10
ONRR GPO: $3.50. NTIS.

The numerical simulation of a 7-pin LMFBR fuel assembly undergoing flow rundown transients has been carried out using the COMMIX-1A computer code. This code performs three-dimensional, transient, single-phase thermal-hydraulic simulations. Transient results have been obtained for two different operating conditions. The computed transient temperatures compare favorably with the experimental measurements.

NUREG/CR-1286

Rancho Seco Wake Effects on Atmospheric Difussion: Simulation in a Meteorological Wind Tunnel.
January 1980.
Colorado State University
ONRR GPO: $5.50. NTIS.

Wind tunnel diffusion tests were conducted on 1:500 scale models of the Rancho Seco Nuclear Power Station, California; surrounding buildings, hyperbolic cooling towers, and terrain were similarly modeled in the Meteorological Wind Tunnel at Colorado State University. The purpose was to quantify the effects on diffusion of buildings perturbing the mean flow. The test program consisted of three gaseous tracer releases of gases having no appreciable plume rise from ground, building, and containment vessel top heights. The program was repeated for eight wind directions and cases of unstable, neutral, and stable atmospheric stratification conditions. Results show that the buildings significantly perturb the dispersion patterns from the flat terrain isolated source release case; hence buildings, hyperbolic towers, and terrain in the immediate vicinity of the release have a major effect. Maximum ground level normalized concentrations occurred during stable stratification. Upwind or downwind presence of the hyperbolic cooling towers was felt by the shift of ground level concentration values toward conditions approximately two categories more unstable than that suggested by the Pasquill-Gifford curves for that background flow stability.

NUREG/CR-1290

Application of the Key Curve Method to Determining J-R Curves for A533B Steel.
January 1980.
U.S. Naval Academy
ORES GPO: $2.50. NTIS.

This report describes the experimental development of a key curve for compact specimens of A533B steel and the use of this experimental key curve to generate the J-Resistance curve directly from the load displacement records without obtaining crack length estimates from unloading compliance, ultrasonics, electric potential or other techniques. In fact two complete key curve functions were developed, the first using subsized fatigue precracked specimens, the second using subsized but machine notched specimens. In each case eight 1/2 T compact specimens with crack lengths from a/W = 0.5 to 0.9 were used to generate a series of digital load displacement records which were assembled in a computer file as the key curve for geometrically similar compact specimens. This key curve can be thought of as defining the locus of load displacement records expected for geometrically similar compact specimens of this material for similar loading conditions if no crack extension were to take place. Deviations between the key curve function and the load displacement record for a particular specimen can then be attributed to crack extension and a calculation for the amount of crack extension can be made. The key curve also allows corrections to be made to J values to account for effects of this crack extension.

Report No. | Bibliographic Data

NUREG/CR-1291

Pressure Vessel Surveillance Dosimetry Improvement Program -- 1979 Annual Report.
January 1980.
Hanford Engineering Development Lab
ORES GPO: $3.50. NTIS.

Aging light-water reactor pressure vessels (LWR-PV) are accumulating significant neutron fluence exposures, with consequent changes in their steel embrittlement characteristics. Recognizing that accurate and validated measurements and data analysis procedures are needed to periodically evaluate the metallurgical condition of these reactor vessels, the U.S. Nuclear Regulatory Commission has established the LWR-PV Surveillance Dosimetry Improvement Program. The primary concern of this program is to improve, standardize, and maintain dosimetry, damage correlation, and the associated reactor analysis procedures used for predicting the integrated effect of neutron exposure to LWR pressure vessels. A vigorous research effort attacking the same measurement and analysis problems exists worldwide, and strong cooperative links between the NRC-supported activities at HEDL, ORNL, and NBS and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Julich, Germany), and several U.K. laboratories have been established. The major benefit of this program will be a significant improvement in the accuracy of the assessment of the remaining safe operating lifetime of light water reactor pressure vessels.

NUREG/CR-1294

Multiple Zone Aerosol Behavior Model.
January 1980.
Battelle Columbus Lab
BMI-2042
ONRR GPO: $2.25. NTIS.

The localized nature of the aerosol source of some LMFBR accident scenarios, such as a sodium pool fire in the containment of an LMFBR, contradicts a basic assumption of all presently employed nuclear aerosol behavior codes. This assumption was tested by creating a cellular model, the Zone code, which accounts for spatial inhomogeneities in an approximate fashion. Calculations using ZONE and a homogeneous (QUICK code) model show that the homogeneous model gives consistently conservative results. This conservatism never exceeds a factor of about 2 for full-scale reactor containments. Once the sodium fire extinguishes, the cellular and homogeneous models agree very closely.

NUREG/CR-1307

Fixed Site Neutralization Model User's Manual.
March 1980.
Sandia Lab, Albuquerque
SAND79-2241
ONMSS GPO: $3.75. NTIS.

The Fixed Site Neutralization Model (FSNM) is a stochastic, time-stepped simulation of an engagement process whereby an adversary force attempts to steal or sabotage sensitive (e.g., nuclear) materials being guarded by a security force on a fixed site and a response force that is offsite. It is anticipated that the FSNM will assist regulatory bodies of the U.S. Government in evaluating fixed-site physical protection systems at various installations in a variety of scenarios. In resolution, the model has representations of individual activities, plans, perceptions, psychological profiles, skills, and equipment. The forces simulated may involve as many as 30 individuals. For purposes of efficiency, most data input to the Fixed Site Neutralization Model are unformatted -- i.e., in machine-readable form.

NUREG/CR-1308

Fixed Site Neutralization Model Programmers Manual - Vol. 1.
March 1980.
Sandia Lab, Albuquerque
SAND79-2242
ONRR GPO: $9.00. NTIS.

The Fixed Site Neutralization Model (FSNM) is a stochastic, time-stepped simulation of an engagement process whereby an adversary force attempts to steal or sabotage sensitive (e.g., nuclear) materials being guarded by a security force on a fixed site and a response force that is offsite. It is anticipated that the FSNM will assist regulatory bodies of the U.S. Government in evaluating fixed-site physical protection systems at various installations in a variety of scenarios. In resolution, the model has representations of individual activities, plans, perceptions, psychological profiles, skills, and equipment. The forces simulated may involve as many as 50 individuals. For purposes of efficiency, most data input to the Fixed Site Neutralization Model are in binary form. Both preprocessors and the FSNM itself are written in FORTRAN.

NUREG/CR-1309

Markov and Semi-Markov Modelling of Small Engagements.
March 1980.
Sandia Lab, Albuquerque
SAND79-2013
ONRR GPO: $2.25. NTIS.

A modified-Markov model of small combat engagements with reinforcements is developed, and the mathematical basis underlying its development is discussed. General assumptions and computational schemes related to the combatant attrition rates used in the model are also presented.

NUREG/CR-1310

Human Effects Aspects in Simulating Hostile Attacks Against Nuclear Facilities.
March 1980.
Sandia Lab, Albuquerque
SAND79-2237
ONRR GPO: $3.50. NTIS.

The psychological, analytic, and programmatic aspects of a computer simulation subroutine are presented. This subroutine was designed to add the realism of human effects elements to, and to be compatible with, the Brief Adversary-Threat Loss-Estimator (BATLE) Model previously developed by Sandia Laboratories for simulating attack on and defense of a nuclear facility. The logic and features of the subroutine are explained, and the results of sensitivity tests are presented.

NUREG/CR-1312

Methods for Evaluating the Leak Tightness of Spent Fuel Container Closures.
March 1980.
Lawrence Livermore Lab
UCRL-52738
ONRR GPO: $6.50. NTIS.

For the Nuclear Regulatory Commission, Lawrence Livermore Laboratory undertook a project to develop and improve methods for assessing the leak tightness of general seal designs for nuclear waste shipping casks. Our objective was to develop and verify analytical tools for predicting leak rates of spent fuel cask closures. We built quarter- and half-scale configurations of five different seal-flange hardware configurations, including: polymer O-rings, silver-coated hollow-metal O-rings, Conoseals, Grayloc seals, and Batzer seals. Using helium as a tracer gas, we conducted leak-rate tests under conditions simulating normal use of spent fuel containers. Leak rates were then correlated with such measured and calculated parameters as temperature, bolt load, seal-flange interface stress, and the differential pressure across the seal. Computer codes were developed; analyses of the seal-flange configuration related the bolt closure force to both seal-flange interface contact stress and surface area.

NUREG/CR-1313

Rationale for a Perturbation Method for Analyzing Fluid-Structure Interactions in BWR Pressure-Suppression Containment Systems.
January 1980.
MIT
ONRR GPO: $3.75. NTIS.

A formal justification is developed for a method in which hydrodynamic data for a transient in a rigid-wall system (derived, for example, from a small-scale experimental simulation) is used as input in a linear computation for the perturbation flow field due to actual wall flexibility. The method is useful in problems where the basic flow transient is so complex that it can be quantified only empirically, and where the fluid-structure interaction is too complex for the fluid side to be represented by a priori defined equivalent mass.

| Report No. | Bibliographic Data |
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NUREG/CR-1314

Gravity Reflood Oscillations in a Pressurized Water Reactor.
February 1980.
MIT
ORES GPO: $3.75. NTIS.

The thermal hydraulics of reflood oscillations in a pressurized water reactor is studied. Violent steam generation beneath the core water level and subsequent expulsion of the coolant are proposed as the physical mechanisms responsible for driving the osc llations. A computer model of the gravity reflood process is formulated based on a simplified boiling curve and one-dimensional fluid mechanics. In general, model calculations compare favorably with experiments. The core coolant level, however, cannot be calculated with certainty because the model does not account, in sufficient details, for interactions beyond the reactor core. Calculated vapor velocities at the core exit indicate that draining of carryover coolant from the upper plenum is possible.

NUREG/CR-1317

Response of a 4-Inch Nuclear Power Plant Valve to Dynamic Excitation.
February 1980.
University of Southern California
CE79-13
ONRR GPO: $7.50. NTIS.

Experimental laboratory studies of a valve for nuclear power plant application were conducted to determine characteristic structural dynamic response relationships between various magnitudes and types of forced excitation. A 4-inch gate valve weighing approximately 160 lb was investigated. The forced dynamic excitation consisted of swept-sine, sine dwell, random, and shock base acceleration that were generated by an electrodynamic shaker or a shock machine. For each type of excitation, different magnitudes and directions of acceleration were applied. Acceleration and strain data were recorded on analog tape and subsequently analyzed, including the determination of transmissibility ratio, power spectral density, time histories, and shock spectra. The valve was found to have a complicated dynamic response characterized by multiple resonances and nonlinear behavior. This report comprises complete l atory data obtained from this investigation, but does not include analysis. Detailed analysis of the data will follow in future documents.

NUREG/CR-1319

Final Report on Cold Leg Integrity Evaluation.
February 1980.
Battelle Columbus Lab
BMI-2044
ORES GPO: $9.00. NTIS.

The objective of this study was to evaluate the margin of safety against a large break in the cold leg piping system of a Pressurized Water Reactor (PWR) power plant. The program scope was such that the study was conducted on the basis of existing technology using principally vendor-supplied stress analyses and published material property data. Consequently, no significant developments related to either fracture mechanics models or fatigue crack growth models were included. The only development undertaken was related to an as yet unverified model for calculating leak rates as a function of crack area.

NUREG/CR-1320

Measured Concentrations of Radioactive Particles in Air in the Vicinity of the Anaconda Uranium Mill.
March 1980.
Argonne National Lab
ANL/ES-89
ONRR GPO: $2.25. NTIS.

Concentrations of radioactive particles (U-238, Th-230, Ra-226, and Pb-210) in air were measured in the vicinity of the Anaconda Uranium Mill, Bluewater, New Mexico. Airborne particles were collected at three stations for about two-thirds of a year using a continuous collection method at a sampling rate of 10 L/min, and also were measured in monthly composites collected periodically at four stations using "high volume" air samplers at a sampling rate of 1400 L/min. The average concentrations calculated from the measurements made in this study suggest that releases from the Anaconda mill were well within the existing limits of the maximum permissible concentrations for inhalation exposure of the general public (10 CFR Part 20, Appendix B).

| Report No. | Bibliographic Data |
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| NUREG/CR-1325 | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites - Quarterly Progress Report Oct.-Dec. 1979.<br>March 1980.<br>Brookhaven National Lab<br>BNL/NUREG-51147<br>ONRR GPO: $2.00. NTIS. |
| | Decreasing radionuclide sorption, $K_d$, was observed for Am-241, Sr-85, and Co-60 when organic substances were added to well water and shale from the Maxey Flats, Kentucky disposal site. Ethylenediaminetetraacetic acid (EDTA) caused the greatest decrease in $K_d$. Several reference clays were also used for comparison. Only montmorillonite maintained its sorption capability in the presence of EDTA. Experiments were performed to establish the existence of organoradionuclide complexes in trench waters from the low-level radioactive waste disposal sites. Fractionations of trench waters were accomplished by gel filtration chromatography. Preliminary results indicated that cesium isotopes in the trench water from West Valley, New York, may be associated with organic molecules as species with molecular weight less than 700, and that it is unlikely an EDTA complex. |
| NUREG/CR-1326 | Properties of Radioactive Wastes and Waste Containers Quarterly Progress Report Oct - Dec. 1979.<br>March 1980.<br>Brookhaven National Lab<br>BNL/NUREG-51148<br>ONRR GPO. $1.7. NTIS. |
| | The Nuclear Regulatory Commission has proposed that solidified waste forms meet certain regulatory requirements. One technique that can be used to meet these NRC requirements is the development of a process control program which would investigate process parameters and determine boundary conditions (envelopes) within which satisfactory solidification could be reasonably assured. This report presents the data generated from the ongoing experimental program at Brookhaven National Laboratory. In the second part of the report, the results of the impact strength of simulated urea-formaldehyde waste forms testing program are presented. |
| NUREG/CR-1328 | Regional Seismicity and Tectonics of Eastern Nebraska, Annual Report June 1, 1978 to May 30, 1979.<br>February 1980.<br>Nebraska Geological Su. . y<br>ONRR GPO: $4.50. NTIS. |
| | This report presents and interprets information that pertains to the geology, structure, tectonics, and seismicity of eastern Nebraska with emphasis on the vicinity of the intersection of the Union and Humboldt Fault zones. Some of the information presented results from a combination of studies begun in earlier years, but the greater part results from studies begun in 1978. The scope of the studies is summarized as follows:<br>1. Rock outcrops in southeastern Cass and northeastern Otoe counties were reexamined and reevaluated, and 39 test holes were drilled to determine the altitude of the upper surface of the Kereford Limestone of Pennsylvanian age;<br>2. Three new seismographs were installed in eastern Nebraska;<br>3. Gravity surveys in eastern Nebraska were extended;<br>4. Ground magnetic surveys in southeastern Cass and northeastern Otoe counties were made and evaluated.<br>Discussion of the results of these studies constitutes the remainder of this report. |
| NUREG/CR-1330 | Review of Methods and Criteria for Dynamic Combinations in Piping Systems.<br>March 1980.<br>Brookhaven National Lab<br>ONRR GPO: $10.00. NTIS. |
| | Structures for nuclear power plant facilities etc., must be designed to resist safely and effectively all types of load combinations that may be expected during their lifetime. The basic problem involves the combination of two or more responses caused by the application of concurrent dynamic loads with random time lag. The probabilistic outcome of the combination results is represented by a cumulative distribution function (CDF), obtained by using a Monte Carlo simulation procedure. This is a report detailing the finding of four specific items related to the problem of combinations of dynamic responses. These include: (a) a parametric study of combination characteristics, |

(b) a generic study of the methods and criteria needed for appropriate combinations in piping components, (c) a study of the validity, adequacy, limitations and applicability of the two criteria for response combinations given in the General Electric (GE) Report (NEDO-24010-2) entitled "Basis of Criteria for Combination of Earthquake and Other Transient Responses by the Square Root of the Sum of the Squares Method" (The Kennedy-Newmark Criteria), and (d) an evaluation of Mark II response data supplied by GE under this work. A comparison of GE and BNL procedures for generating CDF curves was also made.

NUREG/CR-1331

Data Summaries of Licensee Event Reports of Control Rods and Drive Mechanisms at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978.
February 1980.
EG&G Idaho
EGG-EA-5079
ONRR GPO: $8.00. NTIS.

This report describes the results of an analysis of nuclear plant control rod and drive mechanism failures. The data used for this analysis were the Licensee Event Report (LERs). The LERs are written reports filed with the NRC whenever certain failures or incidences occur concerning nuclear plant safety systems. The control rod failures or incidences contained in the LERs were evaluated and categorized as to type of failure or problem and were used to calculate summary control rod failure rate statistics. The report includes a variety of different statistics calculated to highlight or show important failure modes or other failure information. In addition to the quantitative failure rate information, there is also considerable qualitative information tabulated to allow the user to make additional control rod failure rate calculations or inferences.

NUREG/CR-1336

The Bioaccumulation Factor for Phosphorus-32 in Edible Fish Tissue.
March 1980.
Georgia Institute of Technology
ONMSS GPO: $4.25. NTIS.

Information used to derive the bioaccumulation factor for P-32 in edible portions of fish from water was reviewed to evaluate the currently recommended values of 100,000 in fresh water and 29,000 in sea water that are applied in generic calculations of radiation doses to persons from nuclear power reactor effluents. A phosphorus bioaccumulation factor of 70,000 was calculated for larger rivers and estuarine waters on the basis of geometric mean phosphorus concentrations of 2 mg/g wet weight in fish muscle and 0.03 mg/l dissolved in water. A bioaccumulation factor for P-32 of 3,000 was computed by multiplying the phosphorus bioaccumulation factor by the ratio of the biological to the effective turnover rate in fish muscle. A biological turnover rate in muscle of 0.2 percent per day was estimated from phosphorus balances as a long-term average for large fish, although more rapid turnovers have been observed for brief periods. Large deviations from these selected generic bioaccumulation factors occur because of differences in phosphorus concentrations and turnover rates. Bioaccumulation of this magnitude is due to P-32 concentration at lowest trophic levels in the food web, not by concentration in fish, hence the availability of concentrating organisms determines whether this bioaccumulation factor is reached. Several other conditions that affect the P-32 bioaccumulation factor have not been quantified but are suggested for study. Measurement programs are recommended to determine site-specific P-32 bioaccumulation factors and enlarge the data base for the generic values.

NUREG/CR-1337

Light-Water-Reactor Safety Research Program: Quarterly Progress Report July - September 1979.
March 1980.
Argonne National Lab
ANL-79-108
ONRR GPO: $2.00. NTIS.

This progress report summarizes the Argonne National Laboratory work performed during July, August, and September 1979 on water-reactor-safety problems. The research and development areas covered are: (1) Loss-of-Coolant Accident Research: Heat Transfer and Fluid Dynamics and (2) Transient Fuel Response and Fission-Product Release Program.

| Report No. | Bibliographic Data |
|---|---|
| NUREG/CR-1341 | Earthquake-Induced Liquefaction near Lake Amatitlan, Guatemala.<br>February 1980.<br>University of California<br>ONRR GPO: $2.25. NTIS.<br><br>An analysis of data gathered from a field survey, including bore hole cores, and from laboratory measurement of soil particles, penetration resistance and other parameters, leads to the development of a soil profile and the determination of conditions which caused the soil liquefaction resulting from the earthquake motion. Field data are very consistent with experimental-analytical theory. Boundary conditions between liquefaction and non-liquefaction zones are identified and correlated with physical conditions of particle size, water content, etc. |
| NUREG/CR-1350 | Calculation of Factors Affecting the Toxicity of Chlorine to Aquatic Organisms.<br>March 1980.<br>University of Washington<br>ONRR GPO: $4.75. NTIS.<br><br>This report provides a state-of-the-art review of 111 of the most recent available references on the interactive effects of chlorine and various environmental factors on saltwater and freshwater aquatic organisms. The chemistry of chlorine in freshwater and seawater is discussed as it relates to the evaluation of toxicity studies. Those factors found to affect the toxicity of chlorine are concentration, exposure time, temperature, chemical species of chlorine and biotic factors such as species, life stage, and size of organism. Other factors such as pH and metal pollutants modify the toxicity of chlorine; however, the two factors of primary importance in the determination of toxicity are concentration and exposure time. Further research is needed on the reaction products of chlorine in both aquatic environments which will require development of new analytical methodology. Toxicity testing which simulates the transient low-level exposures of organisms in cooling water mixing zones is needed to provide more accurate estimates of conditions in the receiving water. References defining the additive, synergistic or antagonistic effects of chlorine with one or more other chemical pollutants were very limited. |
| NUREG/CR-1351 | Fatigue of Weldments in Nuclear Pressure Vessels and Piping.<br>March 1980.<br>Oak Ridge National Lab<br>ORNL/NUREG-64<br>ONRR GPO: $3.75. NTIS.<br><br>The current American Society of Mechanical Engineers (ASME) Code fatigue design rules for nuclear pressure vessels and piping include no special considerations for weldments other than purely geometric factors. Some research programs aimed at nonnuclear applications have found weldments to display fatigue behavior inferior to that of pure base material. For this reason we reviewed available information on fatigue of weldments relevant to nuclear pressure vessels and piping and determined what (if any) changes in the current design rules appear to be dictated by the available information. |
| NUREG/CR-1352 | High Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research Quarterly Progress Report, October-December 1979.<br>March 1980.<br>Oak Ridge National Lab<br>ORNL/NUREG/TM-383<br>ONRR GPO: $1.75. NTIS.<br><br>Development work continued on the ORECA, BLAST, and ORTAP codes; and verification studies were continued on the ORECA and BLAST codes. Data from the upper-plenum reverse-flow plume heat transfer experiment indicated strong tendencies for the plumes to be unstable. This means that plumes formed in a postulated loss-of-forced-convection accident would be less likely to cause damage to the upper-plenum cover plates. |

| Report No. | Bibliographic Data |
|---|---|
| NUREG/CR-1354 | A Microprocessor Based Underwater Data Acquisition System.<br>February 1980.<br>University of Florida<br>UFL/COEL/TR/038<br>NRC GPO: $4.25. NTIS.<br><br>A self-contained, underwater, microprocessor controlled data acquisition system is described. The system was designed, constructed, programmed, and then successfully operated in the ocean off the coast of Florida. Particular to the design is very low power consumption allowing operation from internal rechargeable batteries, control of system functions by firmware, internal digital cassette storage, and the ability to perform two-way communications with a shore-based computer terminal or remotely located computer via telephone data communications. The system was programmed to operate in two modes: a field station mode in which the data are collected by a remote computer and a storm mode in which data are automatically recorded internally. |
| NUREG/CR-1362 | Data Summaries of Licensee Event Reports of Diesel Generators of U.S. Commercial Nuclear Power Plants from January 1, 1976, to December 31, 1978.<br>March 1980.<br>EG&G Idaho<br>EGG-EA-5092<br>ONRR GPO: $8.00. NTIS.<br><br>This report describes the results of an analysis of nuclear plant Diesel Generator failures. The data used for this analysis were the Licensee Event Reports (LERs). The LERs are written reports filed with the NRC whenever certain failures or incidents occur concerning nuclear plant safety systems. The Diesel General failures or incidents contained in the LERs were evaluated and categorized as to type of failure or problem and were used to calculate summary Diesel Generator failure rate statistics. The report includes a variety of different statistics calculated to highlight or show important failure modes or other failure information. In addition to the quantitative failure rate information, there is also considerable qualitative information tabulated to allow the user to make additional Diesel Generator failure rate calculations or inferences. |

| Keyword Listing A | Report Title | Report No. |
|---|---|---|
| Abnormal Occurrences | Report to Congress on Abnormal Occurrences. July-September 1979. | NUREG-0090, Vol.2, No.3 |
| Accident | Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177OFA Operating Plant. | NUREG-0565 |
| Accident | Impact of Dispersion Parameters on Calculated Reactor Accident Consequences. | NUREG/CR-1150 |
| Accident | Accident Delineation and Evaluation of the High-Temperature Gas-Cooled Reactor System Concepts. | NUREG/CR-1200 |
| Accident | The Social and Economic Effects of the Accident at Three Mile Island - Findings to Date. | NUREG/CR-1215 |
| Accident | Analysis of the Three Mile Island Accident and Alternative Sequences. | NUREG/CR-1219 |
| Accidents | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications. | NUREG-0626 |
| Accidents | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants. | NUREG-0635 |
| Accidents | Emergency Response Scenarios for Transportation Accidents Involving Radioactive Materials. | NUREG/CR-1149 |
| Acquisition | A Microprocessor Based Underwater Data Acquisition System. | NUREG/CR-1354 |
| Action Plans | Task Action Plan for Unresolved Safety Issues Related to Nuclear Power Plants. | NUREG-0649 |
| Activities | New Madrid Seismotectonic Study, Activities During Fiscal Year 1979. | NUREG/CR-0977 |

| Keyword Listing A | Report Title | Report No. |
|---|---|---|
| Activities | Activities, Effects and Impacts of the Coal Fuel Cycle for a 1,000 MWe Electric Power Generating Plant. | NUREG/CR-1060 |
| Adjudication | Report of the Advisory Committee on Construction During Adjudication. | NUREG-0646 |
| Adsorption | Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. | NUREG/CR-0803 |
| Advance Notice | NRC Staff Analysis of Public Comments on Advance Notice of Proposed Rulemaking on Emergency Planning. | NUREG-0628 |
| Advanced | Advanced Reactor Safety Research - Quarterly Report April-June 1979. | NUREG/CR-0984 |
| Advanced | Advanced Reactor Safety Research Division Quarterly Progress Report. April-June 1979. | NUREG/CR-1036 |
| Advisory Committee | Report of the Advisory Committee on Construction During Adjudication. | NUREG-0646 |
| Aerosol | LMFBR Aerosol Release and Transport Program Quarterly Progress Report for April-June 1979. | NUREG/CR-1062 |
| Aerosol | Measurement of Aerosol Deposition Rates in Turbulent Flows. | NUREG/CR-1264 |
| Aerosol | Multiple Zone Aerosol Behavior Model. | NUREG/CR-1294 |
| Aerosols | Microbial Aerosols from Cooling Towers and Cooling Sprays: A Pilot Study. | NUREG/CR-1207 |
| Agent | COPS - Model for Estimating Local Law Enforcement Agent Availability. | NUREG/CR-1202 |

| Keyword Listing A | Report Title | Report No. |
|---|---|---|
| Aggregated | Aggregated Systems Model of Nuclear Safeguards Executive Summary. | NUREG/CR-1140, Vol.1 |
| Aggregated | Aggregated Systems Model of Nuclear Safeguards, Vol. II. | NUREG/CR-1140, Vol.2 |
| Air | Measured Concentrations of Radioactive Particles in Air in the Vicinity of the Anaconda Uranium Mill. | NUREG/CR-1320 |
| Alloy Steels | Notch Ductility Degradation of Low Alloy Steels with Low-to-Intermediate Neutron Fluence Exposures. | NUREG/CR-1053 |
| Alternative | Analysis of the Three Mile Island Accident and Alternative Sequences. | NUREG/CR-1219 |
| Alternative Methods | Study of Alternative Methods for the Management of Liquid Scintillation Counting Wastes. | NUREG-0656 |
| AMBUSH | SKIRMISH and AMBUSH - Tactical Board Games for Development and Evaluation of Road Transit Physical Protection Systems. | NUREG/CR-1255 |
| Americium | Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Np in Marine and Freshwater Systems. | NUREG/CR-0803 |
| Amplitudes | The Correlation of Response Spectral Amplitudes with Seismic Intensity. | NUREG/CR-1259 |
| Anaconda | Measured Concentrations of Radioactive Particles in Air in the Vicinity of the Anaconda Uranium Mill. | NUREG/CR-1320 |
| Analysis | MATPRO II Rev. I: A Handbook of Material Properties for use in the Analysis of Light Water Reactor Fuel Rod Behavior. | NUREG/CR-0497, Rev.1 |
| Analysis | Subcompartment Analysis Procedures. | NUREG/CR-1199 |

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| Keyword Listing A | Report Title | Report No. |
|---|---|---|
| Analysis | Analysis of the Three Mile Island Accident and Alternative Sequences. | NUREG/CR-1219 |
| Analysis Code | PINSIM-MOD1: A Nuclear Fuel Pin/Electric Fuel Pin Simulator Transient Analysis Code. | NUREG/CR-0575 |
| ANS 5.4 | ANS 5.4. A Computer Subroutine for Predicting Fission Gas Release. | NUREG/CR-1213 |
| Anticipated | Anticipated Transients Without Scram for Light Water Reactors. | NUREG-0460, Vol.4 |
| Apollo, PA | A Safeguards Case Study of the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 ES |
| Apollo, PA | A Safeguards Case Study of the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 MR |
| Appellate System | Study of the Nuclear Regulatory Commission's Appellate System. | NUREG-0648 |
| Application | Facilities License Application Record. | NUREG-0652 |
| Application | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.1 |
| Application | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.2 |
| Application | Application of the Key Curve Method to Determining J-R Curves for A533B Steel. | NUREG/CR-1290 |
| Applications | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications. | NUREG-0626 |

| Keyword Listing A | Report Title | Report No. |
|---|---|---|
| Aquatic | Process Notebook for Aquatic Ecosystem Simulation. | NUREG/CR-1182 |
| Aquatic Environments | Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. | NUREG/CR-0803 |
| Aquatic Organisms | Calculation of Factors Affecting the Toxicity of Chlorine to Aquatic Organisms. | NUREG/CR-1350 |
| Argo - Ecosystem | Critical Pathways of Radionuclides to Man from Agro - Ecosystem. | NUREG/CR-1206 |
| Assumptions | Disassembly Phase Energetics: An Examination of the Impact of Simmer Models and Assumptions. | NUREG/CR-1027 |
| ASTM A533 | The Effect of Crack Length and Side Grooves on the Ductile Fracture Toughness Properties of ASTM A533 Steel. | NUREG/CR-1171 |
| Atmospheric | Rancho Seco Wake Effects on Atmospheric Diffusion: Simulation in a Meteorological Wind Tunnel. | NUREG/CR-1286 |
| Attacks | Human Effects Aspects in Simulating Hostile Attacks Against Nuclear Facilities. | NUREG/CR-1310 |
| ATWS | Anticipated Transients Without Scram for Light Water Reactors. | NUREG-0460,Vol. 4 |
| Availability | COPS - A Model for Estimating Local Law Enforcement Agent Availability. | NUREG/CR-1202 |
| Axial | Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2. | NUREG/CR-1011 |

| Keyword Listing B | Report Title | Report No. |
|---|---|---|
| Babcock & Wilcox | Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177OFA Operating Plant. | NUREG-0565 |
| Behavior | Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177OFA Operating Plant. | NUREG-0565 |
| Behavior | MATPRO II Rev. I: A Handbook of Material Properties for use in the Analysis of Light Water Reactor Fuel Rod Behavior. | NUREG/CR-0497, Rev.1 |
| Behavior | The Thermal and Mechanical Behavior of Xenon-Filled Fuel Rod as a Function of Burnup. | NUREG/CR-0749 |
| Behavior | Behavior of a Nine-Rod Fuel Assembly During Power-Cooling-Mismatch Conditions - Results of Test PCM-5. | NUREG/CR-1103 |
| Behavior Model | Multiple Zone Aerosol Behavior Model. | NUREG/CR-1294 |
| Best-Estimate | Best-Estimate LOCA Radiation Signature. | NUREG/CR-1237 |
| Beta-Ray | Delayed Beta- and Gamma-Ray Production due to Thermal-Neutron Fission of 239Pu: Tabular Graphical Spectral Distributions for Times After Fission Between 2 and 1400 Sec. | NUREG/CR-1172 |
| Bibliography | LMFBR Safety 7. Review of Current Issues and Bibliography of Literature (1978). | NUREG/CR-.[illegible]0 |
| Bioaccumulation | The Bioaccumulation Factor for Phosphorus-32 in Edible Fish Tissue. | NUREG/CR-1336 |
| Blowdown | Quarterly Process Report on Blowdown Heat Transfer Separate-Effects Program for July-September 1979. | NUREG/CR-1121 |
| Bluewater, NM | Measured Concentrations of Radioactive Particles in A[illegible] in the Vicinity of the Anaconda Uranium Mill. | NUREG/CR-1320 |

| Keyword Listing B | Report Title | Report No. |
|---|---|---|
| Board Games | SKIRMISH and AMBUSH - Tactical Board Games for Development and Evaluation of Road Transit Physical Protection Systems. | NUREG/CR-1255 |
| Boundary Components | Structural Integrity of Water Reactor Pressure Boundary Components. Annual Report - Fiscal Year 1979. | NUREG/CR-1128 |
| Break | Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177OFA Operating Plant. | NUREG-0565 |
| Break | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications. | NUREG-0626 |
| Break | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants. | NUREG-0635 |
| Break | Experiment Data Report for LOFT Nuclear Small Break Experiment L3-1. | NUREG/CR-1145 |
| Breathing Apparatus | Respirator Studies for the Nuclear Regulatory Commission. Evaluation and Performance of Open Circuit Breathing Apparatus, October 1977 - September 1978. | NUREG/CR-1235 |
| Browns Ferry | Conformation of the Original Qualification Test for Electrical Connectors Used at Browns Ferry Nuclear Power Plant Unit 3. | NUREG/CR-1191 |
| Budget Estimates | U.S. Nuclear Regulatory Commission Budget Estimates Fiscal Year 1981. | NUREG-0629 |
| Building | Stably Stratified Building Wakes. | NUREG/CR-1247 |
| Buildings | Infiltration of Particulate Matter into Buildings. | NUREG/CR-1151 |
| Bulletins | Report of the Bulletins & Orders Task Force of the Office of Nuclear Reactor Regulation. | NUREG-0645, Vol.1 |

| Keyword Listing B | Report Title | Report No. |
|---|---|---|
| Bulletins | Report of the Bulletins & Orders Task Force of the Office of Nuclear Reactor Regulation. Appendices. | NUREG-0645, Vol.2 |
| Bundles | Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2. | NUREG/CR-1011 |
| Bundles | Gravity Dominated Two Phase Flows in Vertical Rod Bundles. | NUREG/CR-1218 |
| Burial | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites. Progress Report No. 10, July-September 1978. | NUREG/CR-1037 |
| Burnup | The Thermal and Mechanical Behavior of Xenon-Filled Fuel Rod as a Function of Burnup. | NUREG/CR-0749 |
| Burst Test | Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2. | NUREG/CR-1011 |
| BWR | Rationale for a Perturbation Method for Analyzing Fluid-Structure Interactions in BWR Pressure-Suppression Containment Systems. | NUREG/CR-1313 |

| Keyword Listing C | Report Title | Report No. |
|---|---|---|
| Calculated | Impact of Dispersion Parameters on Calculated Reactor Accident Consequences. | NUREG/CR-1150 |
| Calculation | An Improvement in the Calculation of Turbulent Friction in Smooth Concentric Annuli. | NUREG/CR-0933 |
| Calculation | Calculation of Factors Affecting the Toxicity of Chlorine to Aquatic Organisms. | NUREG/CR-1350 |
| Calibrating | The Design and Construction of a $D_2O$-Moderated $^{252}Cf$ Source for Calibrating Neutron Personnel Dosimeters Used at Nuclear Power Reactors. | NUREG/CR-1204 |
| Californium 252 | The Design and Construction of a $D_2O$-Moderated $^{252}Cf$ Source for Calibrating Neutron Personnel Dosimeters Used at Nuclear Power Reactors. | NUREG/CR-1204 |
| Case Study | A Safeguards Case Study of the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 ES |
| Case Study | A Safeguards Case Study of the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 MR |
| Centerline | Procedure for the Qualitative Interpretation of Fuel Centerline Thermocouple Response to Step-Power Decreases. | NUREG/CR-1012 |
| Cesium 137 | Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. | NUREG/CR-0803 |
| Characterization | Evaluation of Selected Signal Processing Methods for the Characterization of Steam Generator Eddy Current Signals. | NUREG/CR-1007 |
| Characterization | Characterization of Uranium Tailings Cover Materials for Radon Flux Reduction. | NUREG/CR-1081 |
| Charts | U.S. Nuclear Regulatory Commission Functional Organization Charts. | NUREG-0325, Rev. II |

| Keyword Listing C | Report Title | Report No. |
|---|---|---|
| Chlorine | Calculation of Factors Affecting the Toxicity of Chlorine to Acquatic Organisms. | NUREG/CR-1350 |
| Cladding | Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2. | NUREG/CR-1011 |
| Coal | Activities, Effects and Impacts of the Coal Fuel Cycle for a 1,000 MWe Electric Power Generating Plant. | NUREG/CR-1060 |
| Code Description | Sabres II: Code Description and Users Manual. | NUREG/CR-1178 |
| Coefficients | Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. | NUREG/CR-0803 |
| Cold Leg | Final Report on Cold Leg Integrity Evaluation. | NUREG/CR-1319 |
| Combat | Sabres II: An Individual Resolution Small Arms Combat Simulation Model. | NUREG/CR-0929 |
| Combinations | Review of Methods and Criteria for Dynamic Combinations in Piping Systems. | NUREG/CR-1330 |
| Combinatorial | KENO-IV/CG, The Combinatorial Geometry Version of the Keno Monte Carlo Criticality Safety Program. | NUREG/CR-0709 |
| Combustion Engineering | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants. | NUREG-0635 |
| Commercial | Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978. | NUREG/CR-1205 |
| Commercial | Data Summaries of Licensee Event Reports of Control Rods and Drive Mechanisms at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978. | NUREG/CR-1331 |

| Keyword Listing C | Report Title | Report No. |
|---|---|---|
| Commercial | Data Summaries of Licensee Event Reports of Diesel Generators of U.S. Commerci Nuclear Power Plants from January 1, 1976, to December 31, 1978. | NUREG/CR-1362 |
| Commercially | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites. Progress Report No. 10, July-September 1978. | NUREG/CR-1037 |
| Commercially | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites - Quarterly Progress Report Oct.-Dec. 1979. | NUREG/CR-1325 |
| COMMIX-1A | Computational Results for a 7-pin Hexagonal Fuel Assembly During a Flow Rundown Transient Using the COMMIX-1A Computer Code. | NUREG/CR-1285 |
| Compilation | Compilation of State Laws and Regulations on the Transportation of Radioactive Materials. | NUREG/CR-1263 |
| Compliance | Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR Part 190). | NUREG-0543 |
| Compliance Procedure | An Evaluation of the Unloading Compliance Procedure for J-Integral Testing in the Hot Cell. | NUREG/CR-1070 |
| Computational | Computational Results for a 7-pin Hexagonal Fuel Assembly During a Flow Rundown Transient Using the COMMIX-1A Computer Code. | NUREG/CR-1285 |
| Concentrations | Measured Concentrations of Radioactive Particles in Air in the Vicinity of the Anaconda Uranium Mill. | NUREG/CR-1320 |
| Concentric Annuli | An Improvement in the Calculation of Turbulent Friction in Smooth Concentric Annuli. | NUREG/CR-0933 |
| Conceptual | Program Plan for the Investigation of Vent-Filtered Containment Conceptual Designs for Light Water Reactors. | NUREG/CR-1029 |
| Conditions | Behavior of a Nine-Rod Fuel Assembly During Power-Cooling-Mismatch Conditions - Results of Test PCM-5. | NUREG/CR-1103 |

| Keyword Listing C | Report Title | Report No. |
|---|---|---|
| Congress | Report to Congress on Abnormal Occurrences July-September 1979. | NUREG-0090, Vol.2, No.3 |
| Connectors | Conformation of the Original Qualification Test for Electrical Connectors Used at Browns Ferry Nuclear Power Plant Unit 3. | NUREG/CR-1191 |
| Consequences | Impact of Dispersion Parameters on Calculated Reactor Accident Consequences. | NUREG/CR-1150 |
| Construction | Report of the Advisory Committee on Construction During Adjudication. | NUREG-0646 |
| Construction | The Design and Construction of a $D_2O$-Moderated $^{252}Cf$ Source for Calibrating Neutron Personnel Dosimeters Used at Nuclear Power Reactors. | NUREG/CR-1204 |
| Container | Methods for Evaluating the Leak Tightness of Spent Fuel Container Closures. | NUREG/CR-1312 |
| Containers | Properties of Radioactive Wastes and Waste Containers Quarterly Progress Report Oct. - Dec. 1979. | NUREG/CR-1326 |
| Containment | Program Plan for the Investigation of Vent-Filtered Containment Conceptual Design for Light Water Reactors. | NUREG/CR-1029 |
| Containment | Preliminary Analysis of the Containment Failure Probability by Steam Explosions Following a Hypothetical Core Meltdown in an LWR. | NUREG/CR-1104 |
| Containment | Rationale for a Perturbation Method for Analyzing Fluid-Structure Interactions in BWR Pressure-Suppression Containment Systems. | NUREG/CR-1313 |
| Control Rods | Data Summaries of Licensee Event Reports of Control Rods and Drive Mechanisms at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978. | NUREG/CR-1331 |
| Control Room | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. | NUREG/CR-1270, Vol.1 |

| Keyword Listing C | Report Title | Report No. |
|---|---|---|
| Control Room | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. Appendices, Pt. 1. | NUREG/CR-1270, Vol.2 |
| Control Room | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island. Appendices, Pt. 3. | NUREG/CR-1270, Vol.3 |
| Controllable | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.1 |
| Controllable | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.2 |
| Cooling Towers | Microbial Aerosols from Cooling Towers and Cooling Sprays: A Pilot Study. | NUREG/CR-1207 |
| Cooling-Mismatch | Behavior of a Nine-Rod Fuel Assembly During Power-Cooling-Mismatch Conditions - Results of Test PCM-5. | NUREG/CR-1103 |
| Cooperation | Federal-State Cooperation in Nuclear Power Plant Licensing. | NUREG-0398 |
| COPS | COPS - Model for Estimating Local Law Enforcement Agent Availability. | NUREG/CR-1202 |
| Core | Preliminary Analysis of the Containment Failure Probability by Steam Explosions Following a Hypothetical Core Meltdown in a LWR. | NUREG/CR-1104 |
| Core Barrel | FLX: A Shell Code for Coupled Fluid-Structure Analysis of Core Barrel Dynamics. | NUREG/CR-0957 |
| Correlation | The Correlation of Response Spectral Amplitudes with Seismic Intensity. | NUREG/CR-125[illegible] |
| Corrosion | Fractographic and Microstructural Analysis of Stress Corrosion Cracking of A533 Grade B Class 1 Plate and A508 Class 2 Forging in Pressurized Reactor-Grade Water at 93°C. | NUREG/CR-1127 |

| Keyword Listing C | Report Title | Report No. |
|---|---|---|
| Counting | Study of Alternative Methods for the Management of Liquid Scintillation Counting Wastes. | NUREG-0656 |
| Coupled | FLX: A Shell Code for Coupled Fluid-Structure Analysis of Core Barrel Dynamics. | NUREG/CR-0957 |
| Coupled | Generalized Sensitivity Theory for Systems of Coupled Nonlinear Equations. | NUREG/CR-1003 |
| Crack Length | The Effect of Crack Length and Side Grooves on the Ductile Fracture Toughness Properties of ASTM A533 Steel. | NUREG/CR-1171 |
| Cracking | Stress Corrosion Cracking of Inconel 600 Tubing in Deaerated High Temperature Water. | NUREG/CR-0858 |
| Cracking | Fractographic and Microstructural Analysis of Stress Corrosion Cracking of A533 Grade B Class 1 Plate and A508 Class 2 Forging in Pressurized Reactor-Grade Water at 93°C. | NUREG/CR-1127 |
| Criteria | Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants. | NUREG-0654 |
| Criteria | Review of Methods and Criteria for Dynamic Combinations in Piping Systems. | NUREG/CR-1330 |
| Critical Pathways | Critical Pathways of Radionuclides to Man from Agro - Ecosystem. | NUREG/CR-1206 |
| Criticality | KENO-IV/CG, The Combinatorial Geometry Version of the Keno Monte Carlo Criticality Safety Program. | NUREG/CR-0709 |
| Crustal | Central Virginia Regional Seismic Network: Crustal Velocity Structure in Central and Southwestern Virginia. | NUREG/CR-1217 |
| CSNI Specialists | CSNI Specialists' Meeting on Plastic Tearing Instability. | NUREG/CP-0010 |

| Keyword Listing C | Report Title | Report No. |
|---|---|---|
| Current Issues | LMFBR Safety 7. Review of Current Issues and Bibliography of Literature (1978). | NUREG/CR-1160 |
| Curves | Tearing Instability Analysis Handbook (Formulas and Curves). | NUREG/CR-1221 |
| Curves | Application of the Key Curve Method to Determining J-R Curves for A533B Steel. | NUREG/CR-1290 |

| Keyword Listing D | Report Title | Report No. |
|---|---|---|
| Damage Assessment | Simplified Damage Assessment of Nuclear Power Plants Objected to Turbine Fragments. | NUREG/CR-0966 |
| Damage Limits | Survey of Potential Light Water Reactor Fuel Rod Failure Mechanisms and Damage Limits | NUREG/CR-1132 |
| Data Analysis | Post-Irradiation Data Analysis for NRC/PNL Halden Assembly IFA-431. | NUREG/CR-0797 |
| Data Report | Experiment Data Report for LOFT Nuclear Small Break Experiment L3-1. | NUREG/CR-1145 |
| Data Summaries | Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978. | NUREG/CR-1205 |
| Data-Gathering | Safeguard Vulnerability Analysis Program (SVAP) Data-Gathering Handbook. Volume II. | NUREG/CR-1169, Vol.2 |
| Daughters | Estimated Radiation Doses from Thorium and Daughters Contained in Thoriated Welding Electrodes. | NUREG/CR-1039 |
| Deaerated | Stress Corrosion Cracking of Inconel 600 Tubing in Deaerated High Temperature Water. | NUREG/CR-0858 |
| Decreases | Procedure for the Qualitative Interpretation of Fuel Centerline Thermocouple Response to Step-Power Decreases. | NUREG/CR-1012 |
| Deformed | Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2. | NUREG/CR-1011 |
| Degradation | Notch Ductility Degradation of Low Alloy Steels with Low-to-Intermediate Neutron Fluence Exposures. | NUREG/CR-1053 |
| Delineation | Accident Delineation and Evaluation of the High-Temperature Gas-Cooled Reactor System Concepts. | NUREG/CR-1200 |

| Keyword Listing D | Report Title | Report No. |
|---|---|---|
| Demonstrating | Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR Part 190). | NUREG-0543 |
| Depleted Uranium | Depleted Uranium Dioxide Power Flow Through Very Small Openings. | NUREG/CR-1099 |
| Deposition | Measurement of Aerosol Deposition Rates in Turbulent Flows. | NUREG/CR-1264 |
| Design | Preliminary Design of a Large Scale Graphite Oxidation Loop. | NUREG/CR-1006 |
| Design | The Design and Construction of a $D_2O$-Moderated $^{252}Cf$ Source for Calibrating Neutron Personnel Dosimeters Used at Nuclear Power Reactors. | NUREG/CR-1204 |
| Design | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. | NUREG/CR-1270, Vol.1 |
| Design | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. Appendices, Pt. 1. | NUREG/CR-1270, Vol.2 |
| Design | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island. Appendices, Pt. 3. | NUREG/CR-1270, Vol.3 |
| Designs | Program Plan for the Investigation of Vent-Filtered Containment Conceptual Design for Light Water Reactors. | NUREG/CR-1029 |
| Desorption | Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. | NUREG/CR-0803 |
| Detection | Remote Sensing for Detection and Monitoring of Salt Stress on Vegetation: Evaluation and Guidelines. | NUREG/CR-1231 |
| Determination | Report to the Nuclear Regulatory Commission from the Staff Panel on the Commission's Determination of an Extraordinary Nuclear Occurrence (ENO). | NUREG-0637 |

| Keyword Listing D | Report Title | Report No. |
|---|---|---|
| Determining | Application of the Kev Curve Method to Determining J-R Curves for A533B Steel. | NUREG/CR-1290 |
| Development | SKIRMISH and AMBUSH - Tactical Board Games for Development and Ev..luation of Road Transit Physical Protection Systems. | NUREG/CR-1255 |
| Diesel | Data Summaries of Licensee Event Reports of Diesel Generators of U.S. Commercial Nuclear Power Plants from January 1, 1976, to December 31, 1978. | NUREG/CR-1362 |
| Diffusion | Diffusion and Exhalation of Radon from Uranium Tailings. | NUREG/CR-1138 |
| Diffusion | Rancho Seco Wake Effects on Atmospheric Diffusion: Simulation in a Meteorological Wind Tunnel. | NUREG/CR-1286 |
| Dioxide | Depleted Uranium Dioxide Power Flow Through Very Small Openings. | NUREG/CR-1099 |
| Disaggregation | Econometric Model for the Disaggregation of State-Level Electricity Demand Forecast to the Service Area. | NUREG/CR-1147 |
| Disassembly Phase | Disassembly Phase Energetics: An Examination of the Impact of Simmer Models and Assumptions. | NUREG/CR-1027 |
| Dispersed | Local Drag Laws in Dispersed Two-Phase Flow. | NUREG/CR-1230 |
| Dispersion | Impact of Dispersion Parameters on Calculated Reactor Accident Consequences. | NUREG/CR-1150 |
| Displacement | Techniques of Analysis of Load-Displacement Records by J-Integral Methods. | NUREG/CR-1222 |
| Disposal | Radioactive Waste Processing and Disposal. | NUREG-0643 |

| Keyword Listing D | Report Title | Report No. |
|---|---|---|
| Disposal | Radioactive Waste Processing and Disposal. | NUREG-0644 |
| Disposal Sites | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites. Progress Report No. 10, July-September 1978. | NUREG/CR-1037 |
| Disposal Sites | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites - Quarterly Progress Report Oct.-Dec. 1979. | NUREG/CR-1325 |
| Distribution | Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. | NUREG/CR-0803 |
| Distribution | An Investigation of the Distribution and Entrainment of ECC Water Injected into the Upper Plenum. | NUREG/CR-1078 |
| Distributions | Delayed Beta- and Gamma-Ray Production due to Thermal-Neutron Fission of 239Pu: Tabular Graphical Spectral Distributions for Times After Fission Between 2 and 1400 Sec. | NUREG/CR-1172 |
| Docket Files | Evaluation of Docket Files for Terminated Source Material Licenses. | NUREG/CR-1010 |
| Dominated | Gravity Dominated Two Phase Flows in Vertical Rod Bundles. | NUREG/CR-1218 |
| Dosimeters | The Design and Construction of a $D_2O$-Moderated $^{252}Cf$ Source for Calibrating Neutron Personnel Dosimeters Used at Nuclear Power Reactors. | NUREG/CR-1204 |
| Dosimetry | Performance Testing of Personnel Dosimetry Services: Procedures Manual. | NUREG/CR-1063 |
| Dosimetry | Performance Testing of Personnel Dosimetry Services: Final Report of a Two-Year Pilot Study. | NUREG/CR-1064 |
| Dosimetry | Pressure Vessel Surveillance Dosimetry Improvement Program -- 1979 Annual Report. | NUREG/CR-1291 |

| Keyword Listing D | Report Title | Report No. |
|---|---|---|
| Drag Laws | Local Drag Laws in Dispersed Two-Phase Flow. | NUREG/CR-1230 |
| Drive Mechanisms | Data Summaries of Licensee Event Reports of Control Rods and Drive Mechanisms at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978. | NUREG/CR-1331 |
| Ductile | The Effect of Crack Length and Side Grooves on the Ductile Fracture Toughness Properties of ASTM A533 Steel. | NUREG/CR-1171 |
| Ductility | Notch Ductility Degr dation of Low Alloy Steels with Low-to-Intermediate utron Fluence Exposures. | NUREG/CR-1053 |
| Duress | Stress and Duress Monitoring at NRC-Licensed Facilities. | NUREG/CR-1031 |
| Duress | Stress and Duress Monitoring at NRC-Licensed Facilities. | NUREG/CR-1032 |
| Dynamic | Response of a 4-Inch Nuclear Power Plant Valve to Dynamic Excitation. | NUREG/CR-1317 |
| Dynamic | Review of Methods and Criteria for Dynamic Combinations in Piping Systems. | NUREG/CR-1330 |
| Dynamics | FLX: A Shell Code for Coupled Fluid-Structure Analysis of Core Barrel Dynamics. | NUREG/CR-0957 |
| $D_2O$-Moderated | The Design and Construction of a $D_2O$-Moderated $^{252}Cf$ Source for Calibrating Neutron Personnel Dosimeters Used at Nuclear Power Reactors. | NUREG/CR-1204 |

| Keyword Listing E | Report Title | Report No. |
| --- | --- | --- |
| Earthquake-Induced | Earthquake-Induced Liquefaction near Lake Amatitlan, Guatemala. | NUREG/CR-1341 |
| ECC | An Investigation of the Distribution and Entrainment of ECC Water Injected into the Upper Plenum. | NUREG/CR-1078 |
| Econometric Model | Econometric Model for the Disaggregation of State-Level Electricity Demand Forecast to the Service Area. | NUREG/CR-1147 |
| Economic | The Social and Economic Effects of the Accident at Three Mile Island - Findings to Date. | NUREG/CR-1215 |
| Ecosystem | Process Notebook for Aquatic Ecosystem Simulation. | NUREG/CR-1182 |
| Eddy Current | Evaluation of Selected Signal Processing Methods for the Characterization of Steam Generator Eddy Current Signals. | NUREG/CR-1007 |
| Edible | The Bioaccumulation Factor for Phosphorus-32 in Edible Fish Tissue. | NUREG/CR-1336 |
| Effects | Activities, Effects and Impacts of the Coal Fuel Cycle for a 1,000 MWe Electric Power Generating Plant. | NUREG/CR-1060 |
| Effects | The Social and Economic Effects of the Accident at Three Mile Island - Findings to Date. | NUREG/CR-1215 |
| Effects | Summary of FY79 - Progress on Refill Effects Program - Quarterly Progress Report. July 1 - September 30, 1979. | NUREG/CR-1257 |
| Electric Fuel Pin | PINSIM-MOD1: A Nuclear Fuel Pin/Electric Fuel Pin Simulator Transient Analysis Code. | NUREG/CR-0575 |
| Electric Power | Activities, Effects and Impacts of the Coal Fuel Cycle for a 1,000 MWe Electric Power Generating Plant. | NUREG/CR-1060 |

| Keyword Listing E | Report Title | Report No. |
|---|---|---|
| Electrical Connectors | Conformation of the Original Qualification Test for Electrical Connectors Used at Browns Ferry Nuclear Power Plant Unit 3. | NUREG/CR-1191 |
| Electricity Demand | Econometric Model for the Disaggregation of State-Level Electricity Demand Forecasts to the Service Area. | NUREG/CR-1147 |
| Electrodes | Estimated Radiation Doses from Thorium and Daughters Contained in Thoriated Welding Electrodes. | NUREG/CR-1039 |
| Element | A Three-Dimensional Fluid Finite Element. | NUREG/CR-1232 |
| Emergency | Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants. | NUREG-0654 |
| Emergency Planning | NRC Staff Analysis of Public Comments on Advance Notice of Proposed Rulemaking on Emergency Planning. | NUREG-0628 |
| Emergency Response | Emergency Response Scenarios for Transportation Accidents Involving Radioactive Materials. | NUREG/CR-1149 |
| Emissions | Investigation of Radon-222 Emissions from Underground Uranium Mines. Progress Report 2. | NUREG/CR-1273 |
| Emphasis | A Study of New England Seismicity with Emphasis on Massachusetts and New Hampshire. | NUREG/CR-1186 |
| Energetics | Disassembly Phase Energetics: An Examination of the Impact of Simmer Models and Assumptions. | NUREG/CR-1027 |
| Engagements | Markov and Semi-Markov Modelling of Small Engagements. | NUREG/CR-1309 |
| ENO | Report to the Nuclear Regulatory Commission From the Staff Panel on the Commission's Determination of an Extraordinary Nuclear Occurrence (ENO). | NUREG-0637 |

| Keyword Listing E | Report Title | Report No. |
|---|---|---|
| Entrainment | An Investigation of the Distribution and Entrainment of ECC Water Injected into the Upper Plenum. | NUREG/CR-1078 |
| Environmental Statement | Final Environmental Statement Related to the Operation of Split Rock Uranium Mill - Western Nuclear, Inc., Docket No. 40-1162. | NUREG-0639 |
| EPA Standard | Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR Part 190). | NUREG-0543 |
| Estimate | FRAP-T4 Best Estimate Sensitivity Study. | NUREG/CR-1267 |
| Estimated | Estimated Radiation Doses from Thorium and Daughters Contained in Thoriated Welding Electrodes. | NUREG/CR-1039 |
| Estimation | Radioisotopic Composition of Yellowcake: An Estimation of Stack Release Rate. | NUREG/CR-1216 |
| Evaluating | An Evaluation of the Unloading Compliance Procedure for J-Integral Testing in the Hot Cell. | NUREG/CR-1070 |
| Evaluation | Evaluation of Steam Generator Tube Rupture Events. | NUREG-0651 |
| Evaluation | Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants. | NUREG-0654 |
| Evaluation | Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program for Fiscal Year 1981. | NUREG-0657 |
| Evaluation | Evaluation of Selected Signal Processing Methods for the Characterization of Steam Generator Eddy Current Signals. | NUREG/CR-1007 |
| Evaluation | Evaluation of Docket Files for Terminated Source Material Licenses. | NUREG/CR-1 [illegible] |

| Keyword Listing E | Report Title | Report No. |
|---|---|---|
| Evaluation | Accident Delineation and Evaluation of the High-Temperature Gas-Cooled Reactor System Concepts. | NUREG/CR-1200 |
| Evaluation | Respirator Studies for the Nuclear Regulatory Commission. Evaluation and Performance of Open Circuit Breathing Apparatus, October 1977 - September 1978. | NUREG/CR-1235 |
| Evaluation | SKIRMISH and AMBUSH - Tactical Board Games for Development and Evaluation of Road Transit Physical Protection Systems. | NUREG/CR-1255 |
| Evaluation | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. | NUREG/CR-1270, Vol.1 |
| Evaluation | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. Appendices, Pt. 1. | NUREG/CR-1270, Vol.2 |
| Evaluation | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island. Appendices, Pt. 3. | NUREG/CR-1270, Vol.3 |
| Evaluation | Final Report on Cold Leg Integrity Evaluation. | NUREG/CR-1319 |
| Evaluation Program | Qualification Testing Evaluation Program Light Water Reactor Safety Research Quarterly Report January - March 1979. | NUREG/CR-0970 |
| Examination | Disassembly Phase Energetics: An Examination of the Impact of Simmer Models and Assumptions. | NUREG/CR-1027 |
| Excitation | Response of a 4-Inch Nuclear Power Plant Valve to Dynamic Excitation. | NUREG/CR-1317 |
| Exhalation | Diffusion and Exhalation of Radon from Uranium Tailings. | NUREG/CR-1138 |
| Experiment L3-1 | Experiment Data Report for LOFT Nuclear Small Break Experiment L3-1. | NUREG/CR-1145 |

| Keyword Listing E | Report Title | Report No. |
|---|---|---|
| Explosions | Preliminary Analysis of the Containment Failure Probability by Steam Explosions Following a Hypothetical Core Meltdown in an LWR. | NUREG/CR-1104 |
| Exterior | Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2. | NUREG/CR-1011 |
| Extraordinary | Report to the Nuclear Regulatory Commission from the Staff Panel on the Commission's Determination of an Extraordinary Nuclear Occurrence (ENO). | NUREG-0637 |

| Keyword Listing F | Report Title | Report No. |
|---|---|---|
| Facilities | Facilities License Application Record. | NUREG-0652 |
| Facilities | Stress and Duress Monitoring at NRC-Licensed Facilities. | NUREG/CR-1031 |
| Facilities | Stress and Duress Monitoring at NRC-Licensed Facilities. | NUREG/CR-1032 |
| Facilities | The Insider Threat to Secure Facilities--A Synopsis of Nine Interviews. | NUREG/CR-1279 |
| Factors | Calculation of Factors Affecting the Toxicity of Chlorine to Aquatic Organisms. | NUREG/CR-1350 |
| Failure | Preliminary Analysis of the Containment Failure Probability by Steam Explosions Following a Hypothetical Core Meltdown in an LWR. | NUREG/CR-1104 |
| Failure | Survey of Potential Light Water Reactor Fuel Rod Failure Mechanisms and Damage Limits. | NUREG/CR-1132 |
| Fatigue | Fatigue of Weldments in Nuclear Pressure Vessels and Piping. | NUREG/CR-1351 |
| Federal-State | Federal-State Cooperation in Nuclear Power Plant Licensing. | NUREG-0398 |
| Feedwater | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications. | NUREG-0626 |
| Feedwater | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants. | NUREG-0635 |
| Finite | A Three-Dimensional Fluid Finite Element. | NUREG/CR-1232 |

| Keyword Listing F | Report Title | Report No. |
|---|---|---|
| Fiscal Year | U.S. Nuclear Regulatory Commission Budget Estimates Fiscal Year 1981. | NUREG-0629 |
| Fish Tissue | The Bioaccumulation Factor for Phosphorus-32 in Edible Fish Tissue. | NUREG/CR-1336 |
| Fission | Delayed Beta- and Gamma-Ray Production due to Thermal-Neutron Fission of 239Pu: Tabular Graphical Spectral Distributions for Times After Fission Between 2 and 1400 Sec. | NUREG/CR-1172 |
| Fission | ANS 5.4. A Computer Subroutine for Predicting Fission Gas Release. | NUREG/CR-1213 |
| Fission Gas | Fuel Swelling due to Retained Fission Gas in Molten Fuel During High Temperature Transients. | NUREG/CR-1236 |
| Fission Product | Fission Product Release from Highly Irradiated LWR Fuel. | NUREG/CR-0722 |
| Fixed Site | Fixed Site Neutralization Model User's Manual. | NUREG/CR-1307 |
| Fixed Site | Fixed Site Neutralization Model Programmers Manual - Vol. 1. | NUREG/CR-1308 |
| Floating Plants | Supplement No. 3 to Safety Evaluation Report for Offshore Power Systems (Floating Nuclear Plants 1-8). | NUREG-0054 |
| Flow | Local Drag Laws in Dispersed Two-Phase Flow. | NUREG/CR-1230 |
| Flow Rundown | Computational Results for a 7-pin Hexagonal Fuel Assembly During a Flow Rundown Transient Using the COMMIX-1A Computer Code. | NUREG/CR-1285 |
| Flows | Gravity Dominated Two Phase Flows in Vertical Rod Bundles. | NUREG/CR-1218 |

| Keyword Listing F | Report Title | Report No. |
|---|---|---|
| Fluence Exposures | Notch Ductility Degradation of Low Alloy Steels with Low-to-Intermediate Neutron Fluence Exposures. | NUREG/CR-1053 |
| Fluid | A Three-Dimensional Fluid Finite Element. | NUREG/CR-1232 |
| Fluid-Structure | FLX: A Shell Code for Coupled Fluid-Structure Analysis of Core Barrel Dynamics. | NUREG/CR-0957 |
| Fluid-Structure | Rationale for a Perturbation Method for Analyzing Fluid-Structure Interactions in BWR Pressure-Suppression Containment Systems. | NUREG/CR-1313 |
| Flux Reductions | Characterization of Uranium Tailings Cover Materials for Radon Flux Reduction. | NUREG/CR-1081 |
| FLX | FLX: A Shell Code for Coupled Fluid-Structure Analysis of Core Barrel Dynamics. | NUREG/CR-0957 |
| Forecasts | Econometric Model for the Disaggregation of State-Level Electricity Demand Forecast to the Service Area. | NUREG/CR-1147 |
| Forgings | Tensile Properties of Irradiated and Unirradiated Welds of A533 Steel Plate and A508 Forgings. | NUREG/CR-1158 |
| Formulas | Tearing Instability Analysis Handbook (Formulas and Curves). | NUREG/CR-1221 |
| Fractographic | Fractographic and Microstructural Analysis of Stress Corrosion Cracking of A533 Grade B Class 1 Plate and A508 Class 2 Forging in Pressurized Reactor-Grade Water at 93°C. | NUREG/CR-1127 |
| Fracture | The Effect of Crack Length and Side Grooves on the Ductile Fracture Toughness Properties of ASTM A533 Steel. | NUREG/CR-1171 |
| Fragments | Simplified Damage Assessment of Nuclear Power Plants Objected to Turbine Fragments. | NUREG/CR-0966 |

| Keyword Listing F | Report Title | Report No. |
|---|---|---|
| FRAP-T4 | FRAP-T4 Best Estimate Sensitivity Study. | NUREG/CR-1267 |
| Freshwater | Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. | NUREG/CR-0803 |
| Friction | An Improvement in the Calculation of Turbulent Friction in Smooth Concentric Annuli. | NUREG/CR-0933 |
| FSNM | Fixed Site Neutralization Model User's Manual. | NUREG/CR-1307 |
| FSNM | Fixed Site Neutralization Model Programmers Manual - Vol. 1. | NUREG/CR-1308 |
| Fuel | Fission Product Release from Highly Irradiated LWR Fuel. | NUREG/CR-0722 |
| Fuel | Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2. | NUREG/CR-1011 |
| Fuel | Procedure for the Qualitative Interpretation of Fuel Centerline Thermocouple Response to Step-Power Decreases. | NUREG/CR-1012 |
| Fuel | Methods for Evaluating the Leak Tightness of Spent Fuel Container Closures. | NUREG/CR-1312 |
| Fuel Assemblies | An Evaluation of the In-Pile Pressure Data from Instrumented Fuel Assemblies IFA-431 and IFA-432. | NUREG/CR-1139 |
| Fuel Assembly | Precharacterization Report for Instrumented Nuclear Fuel Assembly IFA-513. | NUREG/CR-1077 |
| Fuel Assembly | Behavior of a Nine-Rod Fuel Assembly During Power-Cooling-Mismatch Conditions - Results of Test PCM-5. | NUREG/CR-1103 |

| eyword Listing F | Report Title | Report No. |
|---|---|---|
| uel Assembly | Computational Results for a 7-pin Hexagonal Fuel Assembly During a Flow Rundown Transient Using the COMMIX-1A Computer Code. | NUREG/CR-1285 |
| uel Cycle | Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR Part 190). | NUREG-0543 |
| uel Cycle | Activities, Effects and Impacts of the Coal Fuel Cycle for a 1,000 MWe Electric Power Generating Plant. | NUREG/CR-1060 |
| uel Rod | MATPRO II Rev. I: A Handbook of Material Properties for use in the Analysis of Light Water Reactor Fuel Rod Behavior. | NUREG/CR-0497, Rev.1 |
| uel Rod | The Thermal and Mechanical Behavior of Xenon-Filled Fuel Rod as a Function of Burnup. | NUREG/CR-0749 |
| uel Rod | Survey of Potential Light Water Reactor Fuel Rod Failure Mechanisms and Damage Limits. | NUREG/CR-1132 |
| uel Swelling | Fuel Swelling due to Retained Fission Gas in Molten Fuel During High Temperature Transients. | NUREG/CR-1236 |
| unction | The Thermal and Mechanical Behavior of Xenon-Filled Fuel Rod as a Function of Burnup. | NUREG/CR-0749 |
| unctional | U.S. Nuclear Regulatory Commission Functional Organization Charts. | NUREG-0325, Rev. II |
| unctions | A Review of NRC Regulatory Processes and Functions. | NUREG-0642 |

POOR ORIGINAL

| Keyword Listing G | Report Title | Report No. |
|---|---|---|
| Games | SKIRMISH and AMBUSH - Tactical Board Games for Development and Evaluation of Road Transit Physical Protection Systems. | NUREG/CR-1255 |
| Gamma-Ray | Delayed Beta- and Gamma-Ray Production due to Thermal-Neutron Fission of 239Pu: Tabular Graphical Spectral Distributions for Times After Fission Between 2 and 1400 Sec. | NUREG/CR-1172 |
| Gas Release | ANS 5.4. A Computer Subroutine for Predicting Fission Gas Release. | NUREG/CR-1213 |
| Gas-Cooled | High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research. Quarterly Progress Report, July 1 - September 30, 1979. | NUREG/CR-1136 |
| Gas-Cooled | Accident Delineation and Evaluation of the High-Temperature Gas-Cooled Reactor System Concepts. | NUREG/CR-1200 |
| Gas-Cooled | High Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research. Quarterly Progress Report, October-December 1979. | NUREG/CR-1352 |
| GE-Designed | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications. | NUREG-0626 |
| Generating Plant | Activities, Effects and Impacts of the Coal Fuel Cycle for a 1,000 MWe Electric Power Generating Plant. | NUREG/CR-1060 |
| Generator | Evaluation of Steam Generator Tube Rupture Events. | NUREG-0651 |
| Generator | Evaluation of Selected Signal Processing Methods for the Characterization of Steam Generator Eddy Current Signals. | NUREG/CR-1007 |
| Generators | Summary of Tube Integrity Operating Experience with Once-Through Steam Generators. | NUREG-0571 |
| Generators | Data Summaries of Licensee Event Reports of Diesel Generators of U.S. Commercial Nuclear Power Plants from January 1, 1976, to December 31, 1978. | NUREG/CR-1362 |

| Keyword Listing G | Report Title | Report No. |
|---|---|---|
| Generic Evaluation | Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177OFA Operating Plant. | NUREG-0565 |
| Generic Evaluation | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications. | NUREG-0626 |
| Generic Evaluation | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants. | NUREG-0635 |
| Geometry Version | KENO-IV/CG, The Combinatorial Geometry Version of the Keno Monte Carlo Criticality Safety Program. | NUREG/CR-0709 |
| Graphite | Preliminary Design of a Large Scale Graphite Oxidation Loop. | NUREG/CR-1006 |
| Gravity | Gravity Dominated Two Phase Flows in Vertical Rod Bundles. | NUREG/CR-1218 |
| Gravity | Gravity Reflood Oscillations in a Pressurized Water Reactor. | NUREG/CR-1314 |
| Guatemala | Earthquake-Induced Liquefaction near Lake Amatitlan, Guatemala. | NUREG/CR-1341 |
| Guidelines | Remote Sensing for Detection and Monitoring of Salt Stress on Vegetation: Evaluation and Guidelines. | NUREG/CR-1231 |

| Keyword Listing H | Report Title | Report No. |
|---|---|---|
| Halden Reactor | Post-Irrac[illegible] Data Analysis for NRC/PNL Halden Assembly IFA-431. | NUREG/CR-0797 |
| Handbook | MATPRO II Rev. I: A Handbook of Material Properties for use in the Analysis of Light Water Reactor Fuel Rod Behavior. | NUREG/CR-0497, Rev.1 |
| Handbook | Safeguard Vulnerability Analysis Program (SVAP) Data-Gathering Handbook. Volume II. | NUREG/CR-1169, Vol.2 |
| Handbook | Tearing Instability Analysis Handbook (Formulas and Curves). | NUREG/CR-1221 |
| Heat Transfer | A Radiative Heat Transfer Model for the TRAC Code. | NUREG/CR-0994 |
| Heat Transfer | Quarterly Process Report on Blowdown Heat Transfer Separate-Effects Program for July-September 1979. | NUREG/CR-1121 |
| Hexagonal | Computational Results for a 7-pin Hexagonal Fuel Assembly During a Flow Rundown Transient Using the COMMIX-1A Computer Code. | NUREG/CR-1285 |
| High Temperature | Fuel Swelling due to Retained Fission Gas in Molten Fuel During High Temperature Transients. | NUREG/CR-1236 |
| High Temperature | High Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research. Quarterly Progress Report, October-December 1979. | NUREG/CR-1352 |
| High-Temperature | High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research. Quarterly Progress Report, July 1 - September 30, 1979. | NUREG/CR-1136 |
| High-Temperature | Accident Delineation and Evaluation of the High-Temperature Gas-Cooled Reactor System Concepts. | NUREG/CR-1200 |
| Hostile | Human Effects Aspects in Simulating Hostile Attacks Against Nuclear Facilities. | NUREG/CR-1310 |

| Keyword Listing H | Report Title | Report No. |
|---|---|---|
| Hot Cell | An Evaluation of the Unloading Compliance Procedure for J-Integral Testing in the Hot Cell. | NUREG/CR-1070 |
| HTGR | Accident Delineation and Evaluation of the High-Temperature Gas-Cooled Reactor System Concepts. | NUREG/CR-1200 |
| Human Effects | Human Effects Aspects in Simulating Hostile Attacks Against Nuclear Facilities. | NUREG/CR-1310 |
| Human Factors | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. | NUREG/CR-1270, Vol.1 |
| Human Factors | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. Appendices, Pt. 1. | NUREG/CR-1270. Vol.2 |
| Human Factors | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island. Appendices, Pt. 3. | NUREG/CR-1270, Vol.3 |
| Hypothetical | Preliminary Analysis of the Containment Failure Probability by Steam Explosions Following a Hypothetical Core Meltdown in an LWR. | NUREG/CR-1104 |

| Keyword Listing I | Report Title | Report No. |
|---|---|---|
| IEEE-323-1974 | Conformation of the Original Qualification Test for Electrical Connectors Used at Browns Ferry Nuclear Power Plant Unit 3. | NUREG/CR-1191 |
| IFA-431 | Post-Irradiation Data Analysis for NRC/PNL Halden Assembly IFA-431. | NUREG/CR-0797 |
| IFA-431 | An Evaluation of the In-Pile Pressure Data from Instrumented Fuel Assemblies IFA-431 and IFA-432. | NUREG/CR-1139 |
| IFA-432 | An Evaluation of the In-Pile Pressure Data from Instrumented Fuel Assemblies IFA-431 and IFA-432. | NUREG/CR-1139 |
| IFA-513 | Precharacterization Report for Instrumented Nuclear Fuel Assembly IFA-513. | NUREG/CR-1077 |
| Illinois | Transportation of Radioactive Material in Illinois. | NUREG/CR-1193 |
| Impact | Disassembly Phase Energetics: An Examination of the Impact of Simmer Models and Assumptions. | NUREG/CR-1027 |
| Impact | Impact of Dispersion Parameters on Calculated Reactor Accident Consequences. | NUREG/CR-1150 |
| Impacts | Activities, Effects and Impacts of the Coal Fuel Cycle for a 1,000 MWe Electric Power Generating Plant. | NUREG/CR-1060 |
| Improvement | An Improvement in the Calculation of Turbulent Friction in Smooth Concentric Annuli. | NUREG/CR-0933 |
| Improvement | Pressure Vessel Surveillance Dosimetry Improvement Program -- 1979 Annual Report. | NUREG/CR-1291 |
| In-Pile | An Evaluation of the In-Pile Pressure Data from Instrumented Fuel Assemblies IFA-431 and IFA-432. | NUREG/CR-1139 |

| Keyword Listing I | Report Title | Report No. |
|---|---|---|
| Inconel 600 | Stress Corrosion Cracking of Inconel 600 Tubing in Deaerated High Temperature Water. | NUREG/CR-0858 |
| Individual | Sabres II: An Individual Resolution Small Arms Combat Simulation Model. | NUREG/CR-0929 |
| Infiltration | Infiltration of Particulate Matter into Buildings. | NUREG/CR-1151 |
| Injected | An Investigation of the Distribution and Entrainment of ECC Water Injected into the Upper Plenum. | NUREG/CR-1078 |
| Insider Threat | The Insider Threat to Secure Facilities--A Synopsis of Nine Interviews. | NUREG/CR-1279 |
| Inspection Methods | Inspection Methods for Physical Protection Project: Quarterly Report, September-November 1979. | NUREG/CR-1258, Vol.1 |
| Instability | CSNI Specialists' Meeting on Plastic Tearing Instability. | NUREG/CP-0010 |
| Instability | Further Results on the Subject of Tearing Instability - Vol. 1. | NUREG/CR-1220, Vol.1 |
| Instability | Further Results on the Subject of Tearing Instability - Vol. 2. | NUREG/CR-1220, Vol.2 |
| Instability | Tearing Instability Analysis Handbook (Formulas and Curves). | NUREG/CR-1221 |
| Instrumented | Precharacterization Report for Instrumented Nuclear Fuel Assembly IFA-513. | NUREG/CR-1077 |
| Instrumented | An Evaluation of the In-Pile Pressure Data From Instrumented Fuel Assemblies IFA-431 and IF-432. | NUREG/CR-1139 |

| Keyword Listing I | Report Title | Report No. |
|---|---|---|
| Integral | An Evaluation of the Unloading Compliance Procedure for J-Integral Testing in the Hot Cell. | NUREG/CR-1070 |
| Integral Methods | Techniques of Analysis of Load-Displacement Records by J-Integral Methods. | NUREG/CR-1222 |
| Integrity | Structural Integrity of Water Reactor Pressure Boundary Components. Annual Report - Fiscal Year 1979. | NUREG/CR-1128 |
| Integrity | Final Report on Cold Leg Integrity Evaluation. | NUREG/CR-1319 |
| Intensity | The Correlation of Response Spectral Amplitudes with Seismic Intensity. | NUREG/CR-1259 |
| Interactions | Rationale for a Perturbation Method for Analyzing Fluid-Structure Interactions in BWR Pressure-Suppression Containment Systems. | NUREG/CR-1313 |
| Intermediate Test | Test of 6-inch-thick Pressure Vessels, Series 3: Intermediate Test Vessel V-8. | NUREG/CR-0675 |
| Interpretation | Procedure for the Qualitative Interpretation of Fuel Centerline Thermocouple Response to Step-Power Decreases. | NUREG/CR-1012 |
| Interviews | The Insider Threat to Secure Facilities--A Synopsis of Nine Interviews. | NUREG/CR-1279 |
| Investigation | Program Plan for the Investigation of Vent-Filtered Containment Conceptual Design for Light Water Reactors. | NUREG/CR-1029 |
| Investigation | An Investigation of the Distribution and Entrainment of ECC Water Injected into the Upper Plenum. | NUREG/CR-1078 |
| Investigation | Investigation of Radon-222 Emissions from Underground Uranium Mines. Progress Report 2. | NUREG/CR-1273 |

| Keyword Listing I | Report Title | Report No. |
|---|---|---|
| Irradiated | Fission Product Release from Highly Irradiated LWR Fuel. | NUREG/CR-0722 |
| Irradiated | Tensile Properties of Irradiated and Unirradiated Welds of A533 Steel Plate and A508 Forgings. | NUREG/CR-1158 |
| Isotope | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites. Progress Report No. 10, July-September 1978. | NUREG/CR-1037 |
| Isotope | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites - Quarterly Progress Report Oct.-Dec. 1979. | NUREG/CR-1325 |

| Keyword Listing K | Report Title | Report No. |
|---|---|---|
| KENO-IV/CG | KENO-IV/CG, The Combinatorial Geometry Version of the Keno Monte Carlo Criticality Safety Program. | NUREG/CR-0709 |
| Key Curve | Application of the Key Curve Method to Determining J-R Curves for A533B Steel. | NUREG/CR-1290 |

| Keyword Listing L | Report Title | Report No. |
|---|---|---|
| Lake Amatitlan | Earthquake-Induced Liquefaction near Lake Amatitlan, Guatamala. | NUREG/CR-1341 |
| Land Burial | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites - Quarterly Progress Report Oct.-Dec. 1979. | NUREG/CR-1325 |
| Large Scale | Preliminary Design of a Large Scale Graphite Oxidation Loop. | NUREG/CR-1006 |
| Law Enforcement | COPS - A Model for Estimating Local Law Enforcement Agent Availability. | NUREG/CR-1202 |
| Laws | Local Drag Laws in Dispersed Two-Phase Flow. | NUREG/CR-1230 |
| Laws | Compilation of State Laws and Regulations on the Transportation of Radioactive Materials. | NUREG/CR-1263 |
| Leak Tightness | Methods for Evaluating the Leak Tightness of Spent Fuel Container Closures. | NUREG/CR-1312 |
| LER | Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978. | NUREG/CR-1205 |
| LER | Data Summaries of Licensee Event Reports of Control Rods and Drive Mechanisms at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978. | NUREG/CR-1331 |
| LER | Data Summaries of Licensee Event Reports of Diesel Generators of U.S. Commercial Nuclear Power Plants from January 1, 1976, to December 31, 1978. | NUREG/CR-1362 |
| License | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications. | NUREG-0626 |
| License | Facilities License Application Record. | NUREG-0652 |

| Keyword Listing L | Report Title | Report No. |
|---|---|---|
| Licensed Facilities | Stress and Duress Monitoring at NRC-Licensed Facilities. | NUREG/CR-1031 |
| Licensed Facilities | Stress and Duress Monitoring at NRC-Licensed Facilities. | NUREG/CR-1032 |
| Licensee Event | Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978. | NUREG/CR-1205 |
| Licensee Event | Data Summaries of Licensee Event Reports of Control Rods and Drive Mechanisms at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978. | NUREG/CR-1331 |
| Licensee Event | Data Summaries of Licensee Event Reports of Diesel Generators of U.S. Commercial Nuclear Power Plants from January 1, 1976, to December 31, 1978. | NUREG/CR-1362 |
| Licenses | Evaluation of Docket Files for Terminated Source Material Licenses. | NUREG/CR-1010 |
| Licensing | Federal-State Cooperation in Nuclear Power Plant Licensing. | NUREG-0398 |
| Lique faction | Earthquake-Induced Liquefaction near Lake Amatitlan, Guatemala. | NUREG/CR-1341 |
| Liquid | Study of Alternative Methods for the Management of Liquid Scintillation Counting Wastes. | NUREG-0656 |
| LMFBR | LMFBR Aerosol Release and Transport Program Quarterly Progress Report for April-June 1979. | NUREG/CR-1062 |
| LMFBR | LMFBR Safety 7. Review of Current Issues and Bibliography of Literature (1978). | NUREG/CR-1160 |
| Load-Displacement | Techniques of Analysis of Load-Displacement Records by J-Integral Methods. | NUREG/CR-1222 |

| Keyword Listing L | Report Title | Report No. |
|---|---|---|
| LOCA | Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 1770FA Operating Plant. | NUREG-0565 |
| LOCA | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications. | NUREG-0626 |
| LOCA | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants. | NUREG-0635 |
| LOCA | Best-Estimate LOCA Radiation Signature. | NUREG/CR-1237 |
| Local | COPS - Model for Estimating Local Law Enforcement Agent Availability. | NUREG/CR-1202 |
| LOFT | Experiment Data Report for LOFT Nuclear Small Break Experiment L3-1. | NUREG/CR-1145 |
| Loop | Preliminary Design of a Large Scale Graphite Oxidation Loop. | NUREG/CR-1006 |
| Loss-of-Coolant | Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 1770FA Operating Plant. | NUREG-0565 |
| Loss-of-Coolant | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications. | NUREG-0626 |
| Loss-of-Coolant | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants. | NUREG-0635 |
| Losses | Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2. | NUREG/CR-1011 |
| Low-Level | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites. Progress Report No. 10, July-September 1978. | NUREG/CR-1037 |

| Keyword Listing L | Report Title | Report No. |
|---|---|---|
| Low-Level | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites - Quarterly Progress Report Oct.-Dec. 1979. | NUREG/CR-1325 |
| LSCW | Study of Alternative Methods for the Management of Liquid Scintillation Counting Wastes. | NUREG-0656 |
| LWR | Anticipated Transients Without Scram for Light Water Reactors. | NUREG-0460, Vol.4 |
| LWR | Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR Part 190). | NUREG-0543 |
| LWR | MATPRO II Rev. I: A Handbook of Material Properties for use in the Analysis of Light Water Reactor Fuel Rod Behavior. | NUREG/CR-0497, Rev.1 |
| LWR | Fission Product Release from Highly Irradiated LWR Fuel. | NUREG/CR-0722 |
| LWR | Qualification Testing Evaluation Program Light Water Reactor Safety Research Quarterly Report January - March 1979. | NUREG/CR-0970 |
| LWR | Program Plan for the Investigation of Vent-Filtered Containment Conceptual Design for Light Water Reactors. | NUREG/CR-1029 |
| LWR | Preliminary Analysis of the Containment Failure Probability by Steam Explosions Following a Hypothetical Core Meltdown in an LWR. | NUREG/CR-1104 |
| LWR | Survey of Potential Light Water Reactor Fuel Rod Failure Mechanisms and Damage Limits. | NUREG/CR-1132 |
| LWR | Light-Water-Reactor Safety Research Program: Quarterly Progress Report. April - June 1979. | NUREG/CR-1164 |
| LWR | Light-Water-Reactor Safety Research Program: Quarterly Progress Report July - September 1979. | NUREG/CR-1337 |

| Keyword Listing M | Report Title | Report No. |
|---|---|---|
| Man | Critical Pathways of Radionuclides to Man from Agro - Ecosystem. | NUREG/CR-1206 |
| Management | Study of Alternative Methods for the Management of Liquid Scintillation Counting Wastes. | NUREG-0656 |
| Manual | Performance Testing of Personnel Dosimetry Services: Procedures Manual. | NUREG/CR-1063 |
| Manual | Sabres II: Code Description and Users Manual. | NUREG/CR-1178 |
| Manual | Fixed Site Neutralization Model User's Manual. | NUREG/CR-1307 |
| Manual | Fixed Site Neutralization Model Programmers Manual - Vol. 1. | NUREG/CR-1308 |
| Marine | Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. | NUREG/CR-0803 |
| Markov | Markov and Semi-Markov Modelling of Small Engagements. | NUREG/CR-1309 |
| Massachusetts | A Study of New England Seismicity with Emphasis on Massachusetts and New Hampshire. | NUREG/CR-1186 |
| Material Control | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.1 |
| Material Control | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.2 |
| Material Properties | MATPRO II Rev. 1: A Handbook of Material Properties for use in the Analysis of Light Water Reactor Fuel Rod Behavior. | NUREG/CR-0497, Rev.1 |

| Keyword Listing M | Report Title | Report No. |
|---|---|---|
| Materials | Characterization of Uranium Tailings Cover Materials for Radon Flux Reduction. | NUREG/CR-1081 |
| MATPRO II | MATPRO II Rev. I: A Handbook of Material Properties for use in the Analysis of Light Water Reactor Fuel Rod Behavior. | NUREG/CR-0497, Rev.1 |
| Measurement | Measurement of Aerosol Deposition Rates in Turbulent Flows. | NUREG/CR-1264 |
| Mechanical Behavior | The Thermal and Mechanical Behavior of Xenon-Filled Fuel Rod as a Function of Burnup. | NUREG/CR-0749 |
| Mechanisms | Survey of Potential Light Water Reactor Fuel Rod Failure Mechanisms and Damage Limits. | NUREG/CR-1132 |
| Mechanisms | Remote Response Mechanisms. | NUREG/CR-1142 |
| Meeting | CSNI Specialists' Meeting on Plastic Tearing Instability. | NUREG/CP-0010 |
| Meltdown | Preliminary Analysis of the Containment Failure Probability by Steam Explosions Following a Hypothetical Core Meltdown in an LWR. | NUREG/CR-1104 |
| Meteorological | Rancho Seco Wake Effects on Atmospheric Diffusion: Simulation in a Meteorological Wind Tunnel. | NUREG/CR-1286 |
| Microbial | Microbial Aerosols from Cooling Towers and Cooling Sprays: A Pilot Study. | NUREG/CR-1207 |
| Microprocessor | A Microprocessor Based Underwater Data Acquisition System. | NUREG/CR-1354 |
| Microstructural | Fractographic and Microstructural Analysis of Stress Corrosion Cracking of A533 Grade B Class 1 Plate and A508 Class 2 Forging in Pressurized Reactor-Grade Water at 93°C. | NUREG/CR-1127 |

| Keyword Listing M | Report Title | Report No. |
|---|---|---|
| Migration | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites. Progress Report No. 10, July-September 1978. | NUREG/CR-1037 |
| Migration | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites - Quarterly Progress Report Oct.-Dec. 1979. | NUREG/CR-1325 |
| Mixed Oxide | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.1 |
| Mixed Oxide | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.2 |
| Modelling | Markov and Semi-Markov Modelling or Small Engagements. | NUREG/CR-1309 |
| Molten Fuel | Fuel Swelling due to Retained Fission Gas in Molten Fuel During High Temperature Transients. | NUREG/CR-1236 |
| Monitoring | Stress and Duress Monitoring at NRC-Licensed Facilities. | NUREG/CR-1031 |
| Monitoring | Stress and Duress Monitoring at NRC-Licensed Facilities. | NUREG/CR-1032 |
| Monitoring | Remote Sensing for Detection and Monitoring of Salt Stress on Vegetation: Evaluation and Guidelines. | NUREG/CR-1231 |
| Monte Carlo | KENO-IV/CG, The Combinatorial Geometry Version of the Keno Monte Carlo Criticality Safety Program. | NUREG/CR-0709 |
| MRBT | Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2. | NUREG/CR-1011 |
| Multiple Zone | Multiple Zone Aerosol Behavior Model. | NUREG/CR-1294 |

| Keyword Listing M | Report Title | Report No. |
|---|---|---|
| Multirod | Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2. | NUREG/CR-1011 |

| Keyword Listing N | Report Title | Report No. |
|---|---|---|
| Near-Term Operating | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications. | NUREG-0626 |
| Nebraska | Regional Seismicity and Tectonics of Eastern Nebraska, Annual Report June 1, 1978 to May 30, 1979. | NUREG/CR-1328 |
| Neutralization | Fixed Site Neutralization Model User's Manual. | NUREG/CR-1307 |
| Neutralization | Fixed Site Neutralization Model Programmers Manual - Vol. 1. | NUREG/CR-1308 |
| Neutron | Notch Ductility Degradation of Low Alloy Steels with Low-to-Intermediate Neutron Fluence Exposures. | NUREG/CR-1053 |
| Neutron | Delayed Beta- and Gamm.-Ray Production due to Thermal-Neutron Fission of 239Pu: Tabular Graphical Spectral Distributions for Times After Fission Between 2 and 1400 Sec. | NUREG/CR-1172 |
| Neutron | The Design and Construction of a $D_2O$-Moderated $^{252}Cf$ Source for Calibrating Neutron Personnel Dosimeters Used at Nuclear Power Reactors. | NUREG/CR-1204 |
| New England | A Study of New England Seismicity with Emphasis on Massachusetts and New Hampshire. | NUREG/CR-1186 |
| New Hampshire | A Study of New England Seismcity with Emphasis on Massachusetts and New Hampshire. | NUREG/CR-1186 |
| New Madrid | New Madrid Seismotectonic Study, Activities During Fiscal Year 1979. | NUREG/CR-0977 |
| Nine-Rod | Behavior of a Nine-Rod Fuel Assembly During Power-Cooling-Mismatch Conditions - Results of Test PCM-5. | NUREG/CR-1103 |
| Nonlinear Equations | Generalized Sensitivity Theory for Systems of Coupled Nonlinear Equations. | NUREG/CR-1003 |

| Keyword Listing N | Report Title | Report No. |
|---|---|---|
| Notch | Notch Ductility Degradation of Low Alloy Steels with Low-to-Intermediate Eeutron Fluence Exposures. | NUREG/CR-1053 |
| Notebook | Process Notebook for Aquatic Ecosystem Simulation. | NUREG/CR-1182 |
| NRC | U.S. Nuclear Regulatory Commission Functional Organization Charts. | NUREG-0325, Rev. II |
| NRC | United States Nuclear Regulatory Commission Staff Practice & Procedure Digest. Supplement 1 to Digest No. 2. April 1, 1978 - September 30, 1978. | NUREG-0386, Supp.1 |
| NRC | NRC Staff Analysis of Public Comments on Advance Notice of Proposed Rulemaking on Emergency Planning. | NUREG-0628 |
| NRC | U.S. Nuclear Regulatory Commission Budget Estimates Fiscal Year 1981. | NUREG-0629 |
| NRC | Report to the Nuclear Regulatory Commission from the Staff Panel on the Commission's Determination of an Extraordinary Nuclear Occurrence (ENO). | NUREG-0637 |
| NRC | A Review of NRC Regulatory Processes and Functions. | NUREG-0642 |
| NRC | Study of the Nuclear Regulatory Commission's Appellate System. | NUREG-0648 |
| NRC | Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program for Fiscal Year 1981. | NUREG-0657 |
| NRC | Stress and Duress Monitoring at NRC-Licensed Facilities. | NUREG/CR-1031 |
| NRC | Stress and Duress Monitoring at NRC-Licensed Facilities. | NUREG/CR-1032 |

| Keyword Listing N | Report Title | Report No. |
|---|---|---|
| NRC | Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research. October - December 1979. | NUREG/CR-1203 |
| NRC | Respirator Studies for the Nuclear Regulatory Commission. Evaluation and Performance of Open Circuit Breathing Apparatus, October 1977 - September 1978. | NUREG/CR-1235 |
| NRC/PNL | Post-Irradiation Data Analysis for NRC/PNL Halden Assembly IFA-431. | NUREG/CR-0797 |
| NRR | Report of the Bulletins & Orders Task Force of the Office of Nuclear Reactor Regulation. | NUREG-0645, Vol.1 |
| NRR | Report of the Bulletins & Orders Task Force of the Office of Nuclear Reactor Regulation. Appendices. | NUREG-0645, Vol.2 |
| Nuclear Facilities | Human Effects Aspects in Simulating Hostile Attacks Against Nuclear Facilities. | NUREG/CR-1310 |
| Nuclear Fuel Pin | PINSIM-MOD1: A Nuclear Fuel Pin/Electric Fuel Pin Simulator Transient Analysis Code. | NUREG/CR-0575 |
| Nuclear Reactor | Nuclear Reactor Safety. Quarterly Progress Report. July - September 1979. | NUREG/CR-1201 |

| Keyword Listing O | Report Title | Report No. |
|---|---|---|
| Occurrence | Report to the Nuclear Regulatory Commission from the Staff Panel on the Commission's Determination of an Extraordinary Nuclear Occurrence (ENO). | NUREG-0637 |
| Occurrences | Report to Congress on Abnormal Occurrences. July-September 1979. | NUREG-0090, Vol.2, No.3 |
| Offshore Power | Supplement No. 3 to Safety Evaluation Report for Offshore Power Systems (Floating Nuclear Plants 1-8). | NUREG-0054 |
| Once-Through | Summary of Tube Integrity Operating Experience with Once-Through Steam Generators. | NUREG-0571 |
| Open Circuit | Respirator Studies for the Nuclear Regulatory Commission. Evaluation and Performance of Open Circuit Breathing Apparatus, October 1977 - September 1978. | NUREG/CR-1235 |
| Openings | Depleted Uranium Dioxide Power Flow Through Very Small Openings. | NUREG/CR-1099 |
| Operating Experience | Summary of Tube Integrity Operating Experience with Once-Through Steam Generators. | NUREG-0571 |
| Operation | Final Environmental Statement Related to the Operation of Split Rock Uranium Mill - Western Nuclear, Inc., Docket No. 40-1162. | NUREG-0639 |
| Operator | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. | NUREG/CR-1270, Vol.1 |
| Operator | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. Appendices, Pt. 1. | NUREG/CR-1270, Vol.2 |
| Operator | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island. Appendices, Pt. 3. | NUREG/CR-1270, Vol.3 |
| Orders | Report of the Bulletins & Orders Task Force of the Office of Nuclear Reactor Regulation. | NUREG-0645, Vol.1 |

| Keyword Listing O | Report Title | Report No. |
|---|---|---|
| Organization | U.S. Nuclear Regulatory Commission Functional Organization Charts. | NUREG-0325, Rev. II |
| Oscillations | Gravity Reflood Oscillations in a Pressurized Water Reactor. | NUREG/CR-1314 |
| Oxidation | Preliminary Design of a Large Scale Graphite Oxidation Loop | NUREG/CR-1006 |
| Oxide | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.1 |
| Oxide | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.2 |

| Keyword Listing P | Report Title | Report No. |
|---|---|---|
| Parameters | Impact of Dispersion Parameters on Calculated Reactor Accident Consequences. | NUREG/CR-1150 |
| Particles | Measured Concentrations of Radioactive Particles in Air in the Vicinity of the Anaconda Uranium Mill. | NUREG/CR-1320 |
| Particulate Matter | Infiltration of Particulate Matter into Buildings. | NUREG/CR-1151 |
| Pathways | Critical Pathways of Radionuclides to Man from Agro - Ecosystem. | NUREG/CR-1206 |
| Pennsylvania | A Safeguards Case Study of the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 ES |
| Pennsylvania | A Safeguards Case Study on the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 MR |
| Performance | Performance Testing of Personnel Dosimetry Services: Procedures Manual. | NUREG/CR-1063 |
| Performance | Performance Testing of Personnel Dosimetry Services: Final Report of a Two-Year Pilot Study. | NUREG/CR-1064 |
| Performance | Respirator Studies for the Nuclear Regulatory Commission. Evaluation and Performance of Open Circuit Breathing Apparatus, October 1977 - September 1978. | NUREG/CR-1235 |
| Performance | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. | NUREG/CR-1270, Vol.1 |
| Performance | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. Appendices, Pt. 1. | NUREG/CR-1270, Vol.2 |
| Performance | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island. Appendices, Pt. 3. | NUREG/CR-1270, Vol.3 |

| Keyword Listing P | Report Title | Report No. |
|---|---|---|
| Personnel | Performance Testing of Personnel Dosimetry Services: Procedures Manual. | NUREG/CR-1063 |
| Personnel | Performance Testing of Personnel Dosimetry Services: Final Report of a Two-Year Pilot Study. | NUREG/CR-1064 |
| Personnel | The Design and Construction of a $D_2O$-Moderated $^{252}Cf$ Source for Calibrating Neutron Personnel Dosimeters Used at Nuclear Power Reactors. | NUREG/CR-1204 |
| Perturbation | Rationale for a Perturbation Method for Analyzing Fluid-Structure Interactions in BWR Pressure-Suppression Containment Systems. | NUREG/CR-1313 |
| Phase | Disassembly Phase Energetics: An Examination of the Impact of Simmer Models and Assumptions. | NUREG/CR-1027 |
| Phosphorus-32 | The Bioaccumulation Factor for Phosphorus-32 in Edible Fish Tissue. | NUREG/CR-1336 |
| Physical Protection | SKIRMISH and AMBUSH - Tactical Board Games for Development and Evaluation of Road Transit Physical Protection Systems. | NUREG/CR-1255 |
| Physical Protection | Inspection Methods for Physical Protection Project: Quarterly Report, September-November 1979. | NUREG/CR-1258, Vol.1 |
| Physics | Physics of Reactor Safety Quarterly Progress Report. July - September 1979. | NUREG/CR-1252 |
| Pilot Study | Performance Testing of Personnel Dosimetry Services: Final Report of a Two-Year Pilot Study. | NUREG/CR-1064 |
| Pilot Study | Microbial Aerosols from Cooling Towers and Cooling Sprays: A Pilot Study. | NUREG/CR-1207 |
| PINSIM-MOD1 | PINSIM-MOD1: A Nuclear Fuel Pin/Electric Fuel Pin Simulator Transient Analysis Code. | NUREG/CR-0575 |

| Keyword Listing P | Report Title | Report No. |
|---|---|---|
| Piping | Fatigue of Weldments in Nuclear Pressure Vessels and Piping. | NUREG/CR-1351 |
| Piping Systems | Review of Methods and Criteria for Dynamic Combinations in Piping Systems. | NUREG/CR-1330 |
| Plastic | CSNI Specialists' Meeting on Plastic Tearing Instability. | NUREG/CP-0010 |
| Plutonium | Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. | NUREG/CR-0803 |
| Plutonium 239 | Delayed Beta- and Gamma-Ray Production due to Thermal-Neutron Fission of 239Pu: Tabular Graphical Spectral Distributions for Times After Fission Between 2 and 1400 Sec. | NUREG/CR-1172 |
| Post-Irradiation | Post-Irradiation Data Analysis for NRC/PNL Halden Assembly IFA-431. | NUREG/CR-0797 |
| Potential | Survey of Potential Light Water Reactor Fuel Rod Failure Mechanisms and Damage Limits. | NUREG/CR-1132 |
| Power Flow | Depleted Uranium Dioxide Power Flow Through Very Small Openings. | NUREG/CR-1099 |
| Power Plant | Power Plant Staffing. | NUREG/CR-1280 |
| Power Plant | Response of a 4-Inch Nuclear Power Plant Valve to Dynamic Excitation. | NUREG/CR-1317 |
| Power Plants | Task Action Plan for Unresolved Safety Issues Related to Nuclear Power Plants. | NUREG-0649 |
| Power Plants | Simplified Damage Assessment o[illegible] [illegible]er Plants Objected to Turbine Fragments. | NUREG/CR-0966 |

| Keyword Listing P | Report Title | Report No. |
|---|---|---|
| Power Plants | Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978. | NUREG/CR-1205 |
| Power Plants | Data Summaries of Licensee Event Reports of Control Rods and Drive Mechanisms at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978. | NUREG/CR-1331 |
| Power Plants | Data Summaries of Licensee Event Reports of Diesel Generators of U.S. Commercial Nuclear Power Plants from January 1, 1976, to December 31, 1978. | NUREG/CR-1362 |
| Practice | United States Nuclear Regulatory Commission Staff Practice & Procedure Digest. Supplement 1 to Digest No. 2. April 1, 1978 - September 30, 1978. | NUREG-0386, Supp.1 |
| Precharacterization | Precharacterization Report for Instrumented Nuclear Fuel Assembly IFA-513. | NUREG/CR-1077 |
| Predicting | ANS 5.4. A Computer Subroutine for Predicting Fission Gas Release. | NUREG/CR-1213 |
| Preparation | Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants. | NUREG-0654 |
| Preparedness | Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants. | NUREG-0654 |
| Pressure | Rationale for a Perturbation Method for Analyzing Fluid-Structure Interactions in BWR Pressure Suppression Containment Systems. | NUREG/CR-1313 |
| Pressure Data | An Evaluation of the In-Pile Pressure Data from Instrumented Fuel Assemblies IFA-431 and IFA-432. | NUREG/CR-1139 |
| Pressure Losses | Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2. | NUREG/CR-1011 |
| Pressure Vessel | Pressure Vessel Surveillance Dosimetry Improvement Program -- 1979 Annual Report. | NUREG/CR-1291 |

| Keyword Listing P | Report Title | Report No. |
|---|---|---|
| Pressure Vessels | Test of 6-inch-thick Pressure Vessels, Series 3: Intermediate Test Vessel V-8. | NUREG/CR-0675 |
| Pressure Vessels | Fatigue of Weldments in Nuclear Pressure Vessels and Piping. | NUREG/CR-1351 |
| Pressurized | Fractographic and Microstructural Analysis of Stress Corrosion Cracking of A533 Grade B Class 1 Plate and A508 Class 2 Forging in Pressurized Reactor-Grade Water at 93°C. | NUREG/CR-1127 |
| Probability | Preliminary Analysis of the Containment Failure Probability by Steam Explosions Following a Hypothetical Core Meltdown in an LWR. | NUREG/CR-1104 |
| Procedure | United States Nuclear Regulatory Commission Staff Practice & Procedure Digest. Supplement 1 to Digest No. 2. April 1, 1978 - September 30, 1978. | NUREG-0386, Supp.1 |
| Procedures | Subcompartment Analysis Procedures. | NUREG/CR-1199 |
| Procedures Manual | Performance Testing of Personnel Dosimetry Services: Procedures Manual. | NUREG/CR-1063 |
| Process | A Review of NRC Regulatory Processes and Functions. | NUREG-0642 |
| Processing | Radioactive Waste Processing and Disposal. | NUREG-0643 |
| Processing | Radioactive Waste Processing and Disposal. | NUREG-0644 |
| Processing Plant | A Safeguards Case Study of the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 ES |
| Processing Plant | A Safeguards Case Study of the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 MR |

| Keyword Listing P | Report Title | Report No. |
|---|---|---|
| Programmers Manual | Fixed Site Neutralization Model Programmers Manual - Vol. 1. | NUREG/CR-1308 |
| Progress Report | Summary of FY79 - Progress on Refill Effects Program - Quarterly Progress Report. July 1 - September 30, 1979. | NUREG/CR-1257 |
| Properties | Properties of Radioactive Wastes and Waste Containers. | NUPEG/CR-0619 |
| Properties | Properties of Radioactive Wastes and Waste Containers Progress Report No. 10, July-September 1978 | NUREG/CR-0857 |
| Propert.es | Properties of Radioactive Wastes and Waste Containers. | NUREG/CR-1126 |
| Properties | The Effect of Crack Length and Side Grooves on the Ductile Fracture Toughness Properties of ASTM A533 Steel. | NUREG/CR-1171 |
| Properties | Properties of Radioactive Wastes and Waste Containers Quarterly Progress Report Oct. - Dec. 1979. | NUREG/CR-1326 |
| Proposed | NRC Staff Analysis of Public Comments and Advance Notice of Proposed Rulemaking on Emergency Planning. | NUREG-0628 |
| Protection | Inspection Methods for Physical Protection Project: Quarterly Report, September-November 1979. | NUREG/CR-1258, Vol.1 |
| Protection Systems | SKIRMISH and AMBUSH - Tactical Board Games for Development and Evaluation of Road Transit Physical Protection Systems. | NUREG/CR-1255 |
| Public Comments | NRC Staff Analysis of Public Comments on Advance Notice of Proposed Rulemaking on Emergency Planning. | NUREG-0628 |
| Pumps | Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978. | NUREG/CR-1205 |

| Keyword Listing P | Report Title | Report No. |
|---|---|---|
| PWR | Gravity Reflood Oscillations in a Pressurized Water Reactor. | NUREG/CR-1314 |

| Keyword Listing Q | Report Title | Report No. |
|---|---|---|
| Qualification | Qualification Testing Evaluation Program Light Water Reactor Safety Research Quarterly Report January - March 1979. | NUREG/CR-0970 |
| Qualification Test | Conformation of the Original Qualification Test for Electrical Connectors Used at Browns Ferry Nuclear Power Plant Unit 3. | NUREG/CR-1191 |
| Qualitative | Procedure for the Qualitative Interpretation of Fuel Centerline Thermocouple Response to Step-Power Decreases. | NUREG/CR-1012 |

| Keyword Listing R | Report Title | Report No. |
|---|---|---|
| Radiation | Best-Estimate LOCA . ·ation Signature. | NUREG/CR-1237 |
| Radiation Doses | Estimated Radiation Doses from Thorium and Daughters Contained in Thoriated Weld ng Electrodes. | NUREG/CR-1039 |
| Radiative | A Radiative Heat Transfer Model for the TRAC Code. | NUREG/CR-0994 |
| Radioactive | Radioactive Waste Processing and Disposal. | NUREG-0643 |
| Radioactive | Radioactive Waste Processing and Disposal. | NUREG-0644 |
| Radioactive | Measured Concentrations of Radioactive Particles in Air in the Vicinity of the Anaconda Uranium Mill. | NUREG/CR-1320 |
| Radioactive Material | Transportation of Radioactive Material in Illinois. | NUREG/CR-1193 |
| Radioactive Materials | Emergency Response Scenarios for Transportation Accidents Involving Radioactive Materials. | NUREG/CR-1149 |
| Radioactive Materials | Compilation of State Laws and Regulations on the Transportation of Radioactive Materials. | NUREG/CR-1263 |
| Radioactive Waste | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites. Progress Report No. 10, July-September 1978. | NUREG/CR-1037 |
| Radioactive Waste | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites - Quarterly Progress Report Oct.-Dec. 1979. | NUREG/CR-1325 |
| Radioactive Wastes | Properties of Radioactive Wastes and Waste Containers. | NUREG/CR-0619 |

| Keyword Listing R | Report Title | Report No. |
|---|---|---|
| Radioactive Wastes | Properties of Radioactive Wastes and Waste Containers Progress Report No. 10, July-September 1978. | NUREG/CR-0857 |
| Radioactive Wastes | Properties of Radioactive Wastes and Waste Containers. | NUREG/CR-1126 |
| Radioactive Wastes | Properties of Radioactive Wastes and Waste Containers Quarterly Progress Report Oct. - Dec. 1979. | NUREG/CR-1326 |
| Radioisotopic | Radioisotopic Composition of Yellowcake: An Estimation of Stack Release Rate. | NUREG/CR-1216 |
| Radiological | Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants. | NUREG-0654 |
| Radionuclides | Distribution Coefficients for Radionuclides in Aquatic Environment III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. | NUREG/CR-0803 |
| Radionuclides | Critical Pathways of Radionuclides to Man from Agro - Ecosystem. | NUREG/CR-1206 |
| Radon | Characterization of Uranium Tailings Cover Materials for Radon Flux Reduction. | NUREG/CR-1081 |
| Radon | Diffusion and Exhalation of Radon from Uranium Tailings. | NUREG/CR-1138 |
| Radon-222 | Investigation of Radon-222 Emissions from Underground Uranium Mines. Progress Report 2. | NUREG/CR-1273 |
| Rancho Seco | Rancho Seco Wake Effects on Atmospheric Diffusion: Simulation in a Meteorological Wind Tunnel. | NUREG/CR-1286 |
| Rates | Measurement of Aerosol Deposition Rates in Turbulent Flows. | NUREG/CR-1264 |

| Keyword Listing R | Report Title | Report No. |
|---|---|---|
| Rationale | Rationale for a Perturbation Method for Analyzing Fluid-Structure Interactions in BWR Pressure-Suppression Containment Systems. | NUREG/CR-1313 |
| Reactor Concepts | Accident Delineation and Evaluation of the High-Temperature Gas-Cooled Reactor System Concepts. | NUREG/CR-1200 |
| Reactor Pressure | Structural Integrity of Water Reactor Pressure Boundary Components. Annual Report - Fiscal Year 1979. | NUREG/CR-1128 |
| Reactor Safety | Reactor Safety Research Programs. | NUREG/CR-0962 |
| Reactor Safety | Advanced Reactor Safety Research - Quarterly Report April-June 1979. | NUREG/CR-0984 |
| Reactor Safety | Water Reactor Safety Research Division Quarterly Progress Report. April-June 1979. | NUREG/CR-1035 |
| Reactor Safety | Advanced Reactor Safety Research Division Quarterly Progress Report. April-June 1979. | NUREG/CR-1036 |
| Reactor Safety | High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research. Quarterly Progress Report, July 1 - September 30, 1979. | NUREG/CR-1136 |
| Reactor Safety | Light-Water-Reactor Safety Research Program: Quarterly Progress Report. April - June 1979. | NUREG/CR-1164 |
| Reactor Safety | Nuclear Reactor Safety. Quarterly Progress Report. July - September 1979. | NUREG/CR-1201 |
| Reactor Safety | Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research. October - December 1979. | NUREG/CR-1203 |
| Reactor Safety | Physics of Reactor Safety Quarterly Progress Report. July - September 1979. | NUREG/CR-1252 |

| Keyword Listing R | Report Title | Report No. |
|---|---|---|
| Reactor Safety | Light-Water-Reactor Safety Research Program: Quarterly Progress Report July - September 1979. | NUREG/CR-1337 |
| Reactor Safety | High Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research. Quarterly Progress Report, October-December 1979. | NUREG/CR-1352 |
| Reactor-Grade | Fractographic and Microstructural Analysis of Stress Corrosion Cracking of A533 Grade B Class 1 Plate and A508 Class 2 Forging in Pressurized Reactor-Grade Water at 93°C. | NUREG/CR-1127 |
| Refill | Summary of FY79 - Progress on Refill Effects Program - Quarterly Progress Report. July 1 - September 30, 1979. | NUREG/CR-1257 |
| Reflood | Gravity Reflood Oscillations in a Pressurized Water Reactor. | NUREG/CR-1314 |
| Regional | Central Virginia Regional Seismic Network: Crustal Velocity Structure in Central and Southwestern Virginia. | NUREG/CR-1217 |
| Regional | Regional Seismicity and Tectonics of Eastern Nebraska, Annual Report June 1, 1978 to May 30, 1979. | NUREG/CR-1328 |
| Regulations | Compilation of State Laws and Regulations on the Transportation of Radioactive Materials. | NUREG/CR-1263 |
| Regulatory | A Review of NRC Regulatory Processes and Functions. | NUREG-0642 |
| Release | Fission Product Release from Highly Irradiated LWR Fuel. | NUREG/CR-0722 |
| Release | LMFBR Aerosol Release and Transport Program Quarterly Progress Report for April-June 1979. | NUREG/CR-1062 |
| Release Rate | Radioisotopic Composition of Yellowcake: An Estimation of Stack Release Rate. | NUREG/CR-1216 |

| Keyword Listing R | Report Title | Report No. |
|---|---|---|
| Remote | Remote Response Mechanisms. | NUREG/CR-1142 |
| Remote Sensing | Remote Sensing for Detection and Monitoring of Salt Stress on Vegetation: Evaluation and Guidelines. | NUREG/CR-1231 |
| Research Program | Seismic Safety Margins Research Program (Phase I) Quarterly Progress Report No. 5. | NUREG/CR-1120, Vol.1 |
| Research Programs | Reactor Safety Research Programs. | NUREG/CR-0962 |
| Resolution | Sabres II: An Individual Resolution Small Arms Combat Simulation Model. | NUREG/CR-0929 |
| Respirator | Respirator Studies for the Nuclear Regulatory Commission. Evaluation and Performance of Open Circuit Breathing Apparatus, October 1977 - September 1978. | NUREG/CR-1235 |
| Response | Procedure for the Qualitative Interpretation of Fuel Centerline Thermocouple Response to Step-Power Decreases. | NUREG/CR-1012 |
| Response | Remote Response Mechanisms. | NUREG/CR-1142 |
| Response | The Correlation of Response Spectral Amplitudes with Seismic Intensity. | NUREG/CR-1259 |
| Response | Response of a 4-Inch Nuclear Power Plant Valve to Dynamic Excitation. | NUREG/CR-1317 |
| Response Plans | Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants. | NUREG-0654 |
| Results | Further Results on the Subject of Tearing Instability - Vol. 1. | NUREG/CR-1220, Vol.1 |

| Keyword Listing R | Report Title | Report No. |
|---|---|---|
| Results | Further Results on the Subject of Tearing Instability - Vol. 2. | NUREG/CR-1220, Vol.2 |
| Review | A Review of NRC Regulatory Processes and Functions. | NUREG-0642 |
| Review | Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program for Fiscal Year 1981. | NUREG-0657 |
| Review | LMFBR Safety 7. Review of Current Issues and Bibliography of Literature (1978). | NUREG/CR-1160 |
| Review | Review of Methods and Criteria for Dynamic Combinations in Piping Systems. | NUREG/CR-1330 |
| Road Transit | SKIRMISH and AMBUSH - Tactical Board Games for Development and Evaluation of Road Transit Physical Protection Systems. | NUREG/CR-1255 |
| Rod Bundles | Gravity Dominated Two Phase Flows in Vertical Rod Bundles. | NUREG/CR-1218 |
| Rogovin Report | Three Mile Island: A Report to the Commission and to the Public. | NUREG/CR-1250, Vol.1 |
| Rulemaking | NRC Staff Analysis of Public Comments on Advance Notice of Proposed Rulemaking on Emergency Planning. | NUREG-0628 |
| Rupture Events | Evaluation of Steam Generator Tube Rupture Events. | NUREG-0651 |
| Ruthenium 106 | Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. | NUREG/CR-0803 |
| Ruthenium 237 | Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. | NUREG/CR-0803 |

| Keyword Listing S | Report Title | Report No. |
|---|---|---|
| Sabres II | SABRES II: An Individual Resolution Small Arms Combat Simulation Model. | NUREG/CR-0929 |
| Sabres II | Sabres II: Code Description and Users Manual. | NUREG/CR-1178 |
| Safeguard | Safeguard Vulnerability Analysis Program (SVAP) Data-Gathering Handbook. Volume II. | NUREG/CR-1169, Vol.2 |
| Safeguards | A Safeguards Case Study of the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 ES |
| Safeguards | A Safeguards Case Study of the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 MR |
| Safeguards | Aggregated Systems Model of Nuclear Safeguards Executive Summary. | NUREG/CR-1140, Vol.1 |
| Safeguards | Aggregated Systems Model of Nuclear Safeguards, Vol. II. | NUREG/CR-1140, Vol.2 |
| Safety | LMFBR Safety 7. Review of Current Issues and Bibliography of Literature (1978). | NUREG/CR-1160 |
| Safety | Nuclear Reactor Safety. Quarterly Progress Report. July - September 1979. | NUREG/CR-1201 |
| Safety | Physics of Reactor Safety Quarterly Progress Report. July - September 1979. | NUREG/CR-1252 |
| Safety Evaluation | Supplement 1 to Safety Evaluation Report for Sequoyah Nuclear Plant, Units 1 and 2. | NUREG-0011 |
| Safety Evaluation | Supplement 1 to Safety Evaluation Report for Sequoyah Nuclear Plant, Units 1 and 2. Docket Nos. 50-327 and 50-328. | NUREG-0011, Supp. 1 |

| Keyword Listing S | Report Title | Report No. |
|---|---|---|
| Safety Evaluation | Supplement No. 3 to Safety Evaluation Report for Offshore Power Systems (Floating Nuclear Plants 1-8). | NUREG-0054 |
| Safety Issues | Task Action Plan for Unresolved Safety Issues Related to Nuclear Power Plants. | NUREG-0649 |
| Safety Margins | Seismic Safety Margins Research Program (Phase I) Quarterly Progress Report No. 5. | NUREG/CR-1120, Vol 1 |
| Safety Program | KENO-IV/CG, The Combinatorial Geometry Version of the Keno Monte Carlo Criticality Safety Program. | NUREG/CR-0709 |
| Safety Programs | Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research. October - December 1979. | NUREG/CR-1203 |
| Safety Research | Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program for Fiscal Year 1981. | NUREG-0657 |
| Safety Research | Reactor Safety Research Programs. | NUREG/CR-0962 |
| Safety Research | Qualification Testing Evaluation Program Light Water Reactor Safety Research Quarterly Report January - March 1979. | NUREG/CR-0970 |
| Safety Research | Advanced Reactor Safety Research - Quarterly Report April-June 1979. | NUREG/CR-0984 |
| Safety Research | Water Reactor Safety Research Division Quarterly Progress Report. April-June 1979. | NUREG/CR-1035 |
| Safety Research | Advanced Reactor Safety Research Division Quarterly Progress Report. April-June 1979. | NUREG/CR-1036 |
| Safety Research | Light-Water-Reactor Safety Research Program: Quarterly Progress Report. April - June 1979. | NUREG/CR-1164 |

| Keyword Listing S | Report Title | Report No. |
|---|---|---|
| Safety Research | Light-Water-Reactor Safety Research Program: :arterly Progress Report July - September 1979. | NUREG/CR-1337 |
| Safety Research | High Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research. Quarterly Progress Report, October-December 1979. | NUREG/CR-1352 |
| Safety Studies | High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research. Quarterly Progress Report, July 1 - September 30, 1979. | NUREG/CR-1136 |
| Salt Stress | Remote Sensing for Detection and Monitoring of Salt Stress on Vegetation: Evaluation and Guidelines. | NUREG/CR-1231 |
| SCC | Stress Corrosion Cracking of Inconel 600 Tubing in Deaerated High Temperature Water. | NUREG/CR-0858 |
| Scintillation | Study of Alternative Methods for the Management of Liquid Scintillation Counting Wastes. | NUREG-0656 |
| Scram | Anticipated Transients Without Scram for Light Water Reactors. | NUREG-0460, Vol.4 |
| Secure Facilities | The Insider Threat to Secure Facilities--A Synopsis of Nine Interviews. | NUREG/CR-1279 |
| Seismic | Seismic Safety Margins Research Program (Phase I) Quarterly Progress Report No. 5. | NUREG/CR-1120, Vol.1 |
| Seismic Intensity | The Correlation of Response Special Amplitudes with Seismic Intensity. | NUREG/CR-1259 |
| Seismic Network | Central Virginia Regional Seismic Network: Crustal Velocity Structure in Central and Southwestern Virginia. | NUREG/CR-1217 |
| Seismicity | A Study of New England Seismicity with Emphasis on Massachusetts and New Hampshire. | NUREG/CR-1186 |

| Keyword Listing S | Report Title | Report No. |
|---|---|---|
| Seismicity | Regional Seismicity and Tectonics of Eastern Nebraska, 'nnual Report June 1, 1978 to May 30, 1979. | NUREG/CR-1328 |
| Seismotectonic | New Madrid Seismotectonic Study, Activities During Fiscal Year 1979. | NUREG/CR-0977 |
| Semi-Markov | Markov and Semi-Markov Modelling of Small Engagements. | NUREG/CR-1309 |
| Sensing | Remote Sensing for Detection and Monitoring of Salt Stress on Vegetation: Evaluation and Guidelines. | NUREG/CR-1231 |
| Sensitivity | FRAP-T4 Best Estimate Sensitivity Study. | NUREG/CR-1267 |
| Sensitivity Theory | Generalized Sensitivity Theory for Systems of Coupled Nonlinear Equations. | NUREG/CR-1003 |
| Separate-Effects | Quarterly Process Report on Blowdown Heat Transfer Separate-Effects Program for July-September 1979. | NUREG/CR-1121 |
| Sequences | Analysis of the Three Mile Island Accident and Alternative Sequences. | NUREG/CR-1219 |
| Sequoyah | Supplement 1 to Safety Evaluation Report for Sequoyah Nuclear Plant, Units 1 and 2. | NUREG-0011 |
| Sequoyah | Sequoyah - Unit 1 Technical Specifications. | NUREG-0658 |
| Shell Code | FLX: A Shell Code for Coupled Fluid-Structure Analysis of Core Barrel Dynamics. | NUREG/CR-0957 |
| Side Grooves | The Effect of Crack Length and Side Grooves on the Ductile Fracture Toughness Properties of ASTM A533 Steel. | NUREG/CR-1171 |

| Keyword Listing S | Report Title | Report No. |
|---|---|---|
| Signal Processing | Evaluation of Selected Signal Processing Methods for the Characterization of Steam Generator Eddy Current Signals. | NUREG/CR-1007 |
| Signature | Best-Estimate LOCA Radiation Signature. | NUREG/CR-1237 |
| Simmer Models | Disassembly Phase Energetics: An Examination of the Impact of Simmer Models and Assumptions. | NUREG/CR-1027 |
| Simulating | Human Effects Aspects in Simulating Hostile Attacks Against Nuclear Facilities. | NUREG/CR-1310 |
| Simulation | Process Notebook for Aquatic Ecosystem Simulation. | NUREG/CR-1182 |
| Simulation | Rancho Seco Wake Effects on Atmospheric Diffusion: Simulation in a Meteorological Wind Tunnel. | NUREG/CR-1236 |
| Simulation Model | SABRES II: An Individual Resolution Small Arms Combat Simulation Model. | NUREG/CR-0929 |
| Simulator | PINSIM-MOD1: A Nuclear Fuel Pin/Electric Fuel Pin Simulator Transient Analysis Code. | NUREG/CR-0575 |
| SKIRMISH | SKIRMISH and AMBUSH - Tactical Board Games for Development and Evaluation of Road Transit Physical Protection Systems. | NUREG/CR-1255 |
| Small Arms | Sabres II: An Individual Resolution Small Arms Combat Simulation Model. | NUREG/CR-0929 |
| Small Break | Experiment Data Report for LOFT Nuclear Small Break Experiment L3-1. | NUREG/CR-1145 |
| Small Openings | Depleted Uranium Dioxide Power Flow Through Very Small Openings. | NUREG/CR-1099 |

| Keyword Listing S | Report Title | Report No. |
|---|---|---|
| Smooth | An Improvement in the Calculation of Turbulent Friction in Smooth Concentric Annuli. | NUREG/CR-0933 |
| Social | The Social and Economic Effects of the Accident at Three Mile Island - Findings to Date. | NUREG/CR-1215 |
| Source Material | Evaluation of Docket Files for Te. .nated Source Material Licenses. | NUREG/CR-1010 |
| Specifications | Sequoyah - Unit 1 Technical Specifications. | NUREG-0658 |
| Spectral | The Correlation of Response Spectral Amplitudes with Seismic Intensity. | NUREG/CR-1259 |
| Spectral Distributions | Delayed Beta- and Gamma-Ray Production due to Thermal-Neutron Fission of 239Pu: Tabular Graphical Spectral Distributions for Times After Fission Between 2 and 1400 Sec. | NUREG/CR-1172 |
| Spent Fuel | Methods for Evaluating the Leak Tightness of Spent Fuel Container Closures. | NUREG/CR-1312 |
| Split Rock | Final Environmental Statement Related to the Operation of Split Rock Uranium Mill - Western Nuclear, Inc., Docket No. 40-1162. | NUREG-0639 |
| Sprays | Microbial Aerosols from Cooling Towers and Cooling Sprays: A Pilot Study. | NUREG/CR-1207 |
| Stably | Stably Stratified Building Wakes. | NUREG/CR-1247 |
| Stack Release | Radioisotopic Composition of Yellowcake: An Estimation of Stack Release Rate. | NUREG/CR-1216 |
| Staff Analysis | NRC Staff Analysis of Public Comments on Advance Notice of Proposed Rulemaking on Emergency Planning. | NUREG-0628 |

| Keyword Listing S | Report Title | Report No. |
|---|---|---|
| Staff Panel | Report to the Nuclear Regulatory Commission from the Staff Panel on the Commission's Determination of an Extraordinary Nuclear Occurrence (ENO). | NUREG-0637 |
| Staffing | Power Plant Staffing. | NUREG/CR-1280 |
| State Laws | Compilation of State Laws and Regulations on the Transportation of Radioactive Materials. | NUREG/CR-1263 |
| State-Federal | Federal-State Cooperation in Nuclear Power Plant Licensing. | NUREG-0398 |
| State-Level | Econometric Model for the Disaggregation of State-Level Electricity Demand Forecast to the Service Area. | NUREG/CR-1147 |
| Steady-State | Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2. | NUREG/CR-1011 |
| Steam | Preliminary Analysis of the Containment Failure Probability by Steam Explosions Following a Hypothetical Core Meltdown in an LWR. | NUREG/CR-1104 |
| Steam Generator | Evaluation of Steam Generator Tube Rupture Events. | NUREG-0651 |
| Steam Generator | Evaluation of Selected Signal Processing Methods for the Characterization of Steam Generator Eddy Current Signals. | NUREG/CR-1007 |
| Steam Generators | Summary of Tube Integrity Operating Experience with Once-Through Steam Generators. | NUREG-0571 |
| Steel | Fractographic and Microstructural Analysis of Stress Corrosion Cracking of A533 Grade B Class 1 Plate and A508 Class 2 Forging in Pressurized Reactor-Grade Water at 93°C. | NUREG/CR-1127 |
| Steel | The Effect of Crack Length and Side Grooves on the Ductile Fracture Toughness Properties of ASTM A533 Steel. | NUREG/CR-1171 |

| Keyword Listing S | Report Title | Report No. |
|---|---|---|
| Steel | Application of the Key Curve Method to Determining J-R Curves for A533B Steel. | NUREG/CR-1290 |
| Steel Plate | Tensile Properties of Irradiated and Unirradiated Welds of A533 Steel Plate and A508 Forgings. | NUREG/CR-1158 |
| Steels | Notch Ductility Degradation of Low Alloy Steels with Low-to-Intermediate Neutron Fluence Exposures. | NUREG/CR-1053 |
| Step-Power | Procedure for the Qualitative Interpretation of Fuel Centerline Thermocouple Response to Step-Power Decreases. | NUREG/CR-1012 |
| Stratified | Stably Stratified Building Wakes. | NUREG/CR-1247 |
| Stress | Stress Corrosion Cracking of Inconel 600 Tubing in Deaerated High Temperature Water. | NUREG/CR-0858 |
| Stress | Stress and Duress Monitoring at NRC-Licensed Facilities. | NUREG/CR-1031 |
| Stress | Stress and Duress Monitoring at NRC-Licensed Facilities. | NUREG/CR-1032 |
| Stress Corrosion | Fractographic and Microstructural Analysis of Stress Corrosion Cracking of A533 Grade B Class 1 Plate and A508 Class 2 Forging in Pressurized Reactor-Grade Water at 93°C. | NUREG/CR-1127 |
| Strontium 85 | Distribution Coefficients for Radionuclides in Aquatic Environments. III. Adsorption and Desorption Studies of 106Ru, 137Cs, 241Am, 85Sr and 237Ru in Marine and Freshwater Systems. | NUREG/CR-0803 |
| Structural | Structural Integrity of Water Reactor Pressure Boundary Components. Annual Report - Fiscal Year 1979. | NUREG/CR-1128 |
| Structure | Central Virginia Regional Seismic Network: Crustal Velocity Structure in Central and Southwestern Virginia. | NUREG/CR-1217 |

| Keyword Listing S | Report Title | Report No. |
|---|---|---|
| Study | Study of the Nuclear Regulatory Commission's Appellate System. | NUREG-0648 |
| Subcompartment | Subcompartment Analysis Procedures. | NUREG/CR-1199 |
| Subroutine | ANS 5.4. A Computer Subroutine for Predicting Fission Gas Release. | NUREG/CR-1213 |
| Suppression | Rationale for a Perturbation Method for Analyzing Fluid-Structure Interactions in BWR Pressure-Suppression Containment Systems. | NUREG/CR-1313 |
| Surveillance | Pressure Vessel Surveillance Dosimetry Improvement Program -- 1979 Annual Report. | NUREG/CR-1291 |
| SVAP | Safeguard Vulnerability Analysis Program (SVAP) Data-Gathering Handbook. Volume II. | NUREG/CR-1169, Vol.2 |
| Systems Model | Aggregated Systems Model of Nuclear Safeguards, Vol. II. | NUREG/CR-1140, Vol.2 |
| Systems Model | Aggregated Systems Model of Nuclear Safeguards Executive Summary. | NUURE/CR-1140 |

| Keyword Listing T | Report Title | Report No. |
|---|---|---|
| Tactical | SKIRMISH and AMBUSH - Tactical Board Games for Development and Evaluation of Road Transit Physical Protection Systems. | NUREG/CR-1255 |
| Tailings | Characterization of Uranium Tailings Cover Materials for Radon Flux Reduction. | NUREG/CR-1081 |
| Tailings | Diffusion and Exhalation of Radon from Uranium Tailings. | NUREG/CR-1138 |
| Task Force | Report of the Bulletins & Orders Task Force of the Office of Nuclear Reactor Regulation. | NUREG-0645, Vol.1 |
| Task Force | Report of the Bulletins & Orders Task Force of the Office of Nuclear Reactor Regulation. Appendices. | NUREG-0645, Vol.2 |
| Tearing Instability | CSNI Specialists' Meeting on Plastic Tearing Instability. | NUREG/CP-0010 |
| Tearing Instability | Further Results on the Subject of Tearing Instability - Vol. 1. | NUREG/CR-1220, Vol.1 |
| Tearing Instability | Further Results on the Subject of Tearing Instability - Vol. 2. | NUREG/CR-1220, Vol.2 |
| Tearing Instability | Tearing Instability Analysis Handbook (Formulas and Curves). | NUREG/CR-1221 |
| Technical | Sequoyah - Unit 1 Technical Specifications. | NUREG-0658 |
| Techniques | Techniques of Analysis of Load-Displacement Records by J-Integral Methods. | NUREG/CR-1222 |
| Tectonics | Regional Seismicity and Tectonics of Eastern Nebraska, Annual Report June 1, 1978 to May 30, 1979. | NUREG/CR-1328 |

| Keyword Listing T | Report Title | Report No. |
|---|---|---|
| Temperature | Stress Corrosion Cracking of Inconel 600 Tubing in Deaerated High Temperature Water. | NUREG/CR-0858 |
| Tensile Properties | Tensile Properties of Irradiated and Unirradiated Welds of A533 Steel Plate and A508 Forgings. | NUREG/CR-1158 |
| Test | Test of 6-inch-thick Pressure Vessels, Series 3: Intermediate Test Vessel V-8. | NUREG/CR-0675 |
| Test PCM-5 | Behavior of a Nine-Rod Fuel Assembly During Power-Cooling-Mismatch Conditions - Results of Test PCM-5. | NUREG/CR-1103 |
| Testing | Qualification Testing Evaluation Program Light Water Reactor Safety Research Quarterly Report January - March 1979. | NUREG/CR-0970 |
| Testing | Performance Testing of Personnel Dosimetry Services: Procedures Manual. | NUREG/CR-1063 |
| Testing | Performance Testing of Personnel Dosimetry Services: Final Report of a Two-Year Pilot Study. | NUREG/CR-1064 |
| Testing | An Evaluation of the Unloading Compliance Procedure for J-Integral Testing in the Hot Cell. | NUREG/CR-1070 |
| Thermal | The Thermal and Mechanical Behavior of Xenon-Filled Fuel Rod as a Function of Burnup. | NUREG/CR-0749 |
| Thermal-Neutron | Delayed Beta- and Gamma-Ray Production due to Thermal-Neutron Fission of 239Pu: Tabular Graphical Spectral Distributions for Times After Fission Between 2 and 1400 Sec. | NUREG/CR-1172 |
| Thermocouple | Procedure for the Qualitative Interpretation of Fuel Centerline Thermocouple Response to Step-Power Decreases. | NUREG/CR-1012 |
| Thoriated | Estimated Radiation Doses from Thorium and Daughters Contained in Thoriated Welding Electrodes. | NUREG/CR-1039 |

| Keyword Listing T | Report Title | Report No. |
|---|---|---|
| Thorium | Estimated Radiation Doses from Thorium and Daughters Contained in Thoriated Welding Electrodes. | NUREG/CR-1039 |
| Threat | The Insider Threat to Secure Facilities--A Synopsis of Nine Interviews. | NUREG/CR-1279 |
| Three Mile Island | The Social and Economic Effects of the Accident at Three Mile Island - Findings to Date. | NUREG/CR-1215 |
| Three Mile Island | Analysis of the Three Mile Island Accident and Alternative Sequences. | NUREG/CR-1219 |
| Three Mile Island | Three Mile Island: A Report to the Commission and to the Public. | NUREG/CR-1250, Vol.1 |
| Three Mile Island | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. | NUREG/CR-1270, Vol.1 |
| Three Mile Island | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2. Appendices, Pt. 1. | NUREG/CR-1270, Vol.2 |
| Three Mile Island | Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island. Appendices, Pt. 3. | NUREG/CR-1270, Vol.3 |
| Three-Dimensional | A Three-Dimensional Fluid Finite Element. | NUREG/CR-1232 |
| Through-put | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.1 |
| Through-put | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.2 |
| Times After Fission | Delayed Beta- and Gamma-Ray Production due to Thermal-Neutron Fission of 239Pu: Tabular and Graphical Spectral Distributions for Times After Fission Between 2 and 1400. | NUREG/CR-1172 |

| Keyword Listing T | Report Title | Report No. |
|---|---|---|
| Toughness | The Effect of Crack Length and Side Grooves on the Ductile Fracture Toughness Properties of ASTM A533 Steel. | NUREG/CR-1171 |
| Toxicity | Calculation of Factors Affecting the Toxicity of Chlorine to Aquatic Organisms. | NUREG/CR-1350 |
| TRAC Code | A Radiative Heat Transfer Model for the TRAC Code. | NUREG/CR-0994 |
| Transient | PINSIM-MOD1: A Nuclear Fuel Pin/Electric Fuel Pin Simulator Transient Analysis Code. | NUREG/CR-0575 |
| Transient | Computational Results for a 7-pin Hexagonal Fuel Assembly During a Flow Rundown Transient Using the COMMIX-1A Computer Code. | NUREG/CR-1285 |
| Transients | Anticipated Transients Without Scram for Light Water Reactors. | NUREG-0460, Vol.4 |
| Transients | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications. | NUREG-0626 |
| Transients | Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants. | NUREG-0635 |
| Transients | Fuel Swelling due to Retained Fission Gas in Molten Fuel During High Temperature Transients. | NUREG/CR-1236 |
| Transport | LMFBR Aerosol Release and Transport Program Quarterly Progress Report for April-June 1979. | NUREG/CR-1062 |
| Transportation | Emergency Response Scenarios for Transportation Accidents Involving Radioactive Materials. | NUREG/CR-1149 |
| Transportation | Transportation of Radioactive Material in Illinois. | NUREG/CR-1193 |

| Keyword Listing T | Report Title | Report No. |
|---|---|---|
| Transportation | Compilation of State Laws and Regulations on the Transportation of Radioactive Materials. | NUREG/CR-1263 |
| Tube | Evaluation of Steam Generator Tube Rupture Events. | NUREG-0651 |
| Tube Integrity | Summary of Tube Integrity Operating Experience with Once-Through Steam Generators. | NUREG-0571 |
| Tubing | Stress Corrosion Cracking of Inconel 600 Tubing in Deaerated High Temperature Water. | NUREG/CR-0858 |
| Turbine | Simplified Damage Assessment of Nuclear Power Plants Objected to Turbine Fragments. | NUREG/CR-0966 |
| Turbulent | An Improvement in the Calculation of Turbulent Friction in Smooth Concentric Annuli. | NUREG/CR-0933 |
| Turbulent Flows | Measurement of Aerosol Deposition Rates in Turbulent Flows. | NUREG/CR-1264 |
| Two Phase | Gravity Dominated Two Phase Flows in Vertical Rod Bundles. | NUREG/CR-1218 |
| Two-Phase | Local Drag Laws in Dispersed Two-Phase Flow. | NUREG/CR-1230 |

| Keyword Listing U | Report Title | Report No. |
|---|---|---|
| U.S. | Data Summaries of Licensee Event Reports of Control Rods and Drive Mechanisms at U.S. Commercial Nuclear Power Plants from January 1, 1972, to April 30, 1978. | NUREG/CR-1331 |
| U.S. | Data Summaries of Licensee Event Reports of Diesel Generators of U.S. Commercial Nuclear Power Plants from January 1, 1976, to December 31, 1978. | NUREG/CR-1362 |
| Udall, Chairman | A Safeguards Case Study of the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 ES |
| Udall, Chairman | A Safeguards Case Study on the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 MR |
| Underground | Investigation of Radon-222 Emissions from Underground Uranium Mines. Progress Report 2. | NUREG/CR-1273 |
| Underwater | A Microprocessor Based Underwater Data Acquisition System. | NUREG/CR-1354 |
| Unirradiated | Tensile Properties of Irradiated and Unirradiated Welds of A533 Steel Plate and A508 Forgings. | NUREG/CR-1158 |
| Unit Approach | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.1 |
| Unit Approach | The Controllable Unit Approach to Material Control: Application to a High Through-put Mixed Oxide Process. | NUREG/CR-1214, Vol.2 |
| Unloading | An Evaluation of the Unloading Compliance Procedure for J-Integral Testing in the Hot Cell. | NUREG/CR-1070 |
| Unresolved | Task Action Plan for Unresolved Safety Issues Related to Nuclear Power Plants. | NUREG-0649 |
| Upper Plenum | An Investigation of the Distribution and Entrainment of ECC Water Injected into the Upper Plenum. | NUREG/CR-1078 |

| Keyword Listing U | Report Title | Report No. |
|---|---|---|
| Uranium | Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR Part 190). | NUREG-0543 |
| Uranium | A Safeguards Case Study of the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 ES |
| Uranium | A Safeguards Case Study of the Nuclear Materials and Equipment Company Uranium Processing Plant, Apollo, Pennsylvania. | NUREG-0627 MR |
| Uranium | Characterization of Uranium Tailings Cover Materials for Radon Flux Reduction. | NUREG/CR-1081 |
| Uranium | Depleted Uranium Dioxide Power Flow Through Very Small Openings. | NUREG/CR-1099 |
| Uranium | Diffusion and Exhalation of Radon from Uranium Tailings. | NUREG/CR-1138 |
| Uranium Mill | Final Environmental Statement Related to the Operation of Split Rock Uranium Mill - Western Nuclear, Inc., Docket No. 40-1162. | NUREG-0639 |
| Uranium Mill | Measured Concentrations of Radioactive Particles in Air in the Vicinity of the Anaconda Uranium Mill. | NUREG/CR-1320 |
| Uranium Mines | Investigation of Radon-222 Emissions from Underground Uranium Mines. Progress Report. | NUREG/CR-1273 |
| Users Manual | Sabres II: Code Descrip ion and Users Manual. | NUREG/CR-1178 |
| User's Manual | Fixed Site Neutralization Model User's Manual. | NUREG/CR-1307 |

| Keyword Listing V | Report Title | Report No. |
|---|---|---|
| Valve | Response of a 4-Inch Nuclear Power Plant Valve to Dynamic Excitation. | NUREG/CR-1317 |
| Vegetation | Remote Sensing for Detection and Monitoring of Salt Stress on Vegetation: Evaluation and Guidelines. | NUREG/CR-1231 |
| Velocity | Central Virginia Regional Seismic Network: Crustal Velocity Structure in Central and Southwestern Virginia. | NUREG/CR-1217 |
| Vent-Filtered | Program Plan for the Investigation of Vent-Filtered Containment Conceptual Design for Light Water Reactors. | NUREG/CR-1029 |
| Vertical | Gravity Dominated Two Phase Flows in Vertical Rod Bundles. | NUREG/CR-1218 |
| Vessel V-8 | Test of 6-inch-thick Pressure Vessels, Series 3: Intermediate Test Vessel V-8. | NUREG/CR-0675 |
| Vicinity | Measured Concentrations of Radioactive Particles in Air in the Vicinity of the Anaconda Uranium Mill. | NUREG/CR-1320 |
| Virginia | Central Virginia Regional Seismic Network: Crustal Velocity Structure in Central and Southwestern Virginia. | NUREG/CR-1217 |
| Vulnerability | Safeguard Vulnerability Analysis Program (SVAP) Data-Gathering Handbook. Volume II. | NUREG/CR-1169, Vol.2 |

| Keyword Listing W | Report Title | Report No. |
|---|---|---|
| Wake Effects | Rancho Seco Wake Effects on Atmospheric Diffusion: Simulation in a Meteorological Wind Tunnel. | NUREG/CR-1286 |
| Wakes | Stably Stratified Building Wakes. | NUREG/CR-1247 |
| Waste | Radioactive Waste Processing and Disposal. | NUREG-0643 |
| Waste | Radioactive Waste Processing and Disposal. | NUREG-0644 |
| Waste Containers | Properties of Radioactive Wastes and Waste Containers. | NUREG/CR-0619 |
| Waste Containers | Properties of Radioactive Wastes and Waste Containers Progress Report No. 10, July-September 1978. | NUREG/CR-0857 |
| Waste Containers | Properties of Radioactive Wastes and Waste Containers. | NUREG/CR-1126 |
| Waste Containers | Properties of Radioactive Wastes and Waste Containers Quarterly Progress Report Oct. - Dec. 1979. | NUREG/CR-1326 |
| Wastes | Study of Alternative Methods for the Management of Liquid Scintillation Counting Wastes. | NUREG-0656 |
| Water | Stress Corrosion Cracking of Inconel 600 Tubing in Deaerated High Temperature Water. | NUREG/CR-0858 |
| Water | An Investigation of the Distribution and Entrainment of ECC Water Injected into the Upper Plenum. | NUREG/CR-1078 |
| Water | Fractographic and Microstructural Analysis of Stress Corrosion Cracking of A533 Grade B Class 1 Plate and A508 Class 2 Forging in Pressurized Reactor-Grade Water at 93°C. | NUREG/CR-1127 |

| Keyword Listing W | Report Title | Report No. |
|---|---|---|
| Water | Structural Integrity of Water Reactor Pressure Boundary Components. Annual Report - Fiscal Year 1979. | NUREG/CR-1128 |
| Water Chemistry | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites. Progress Report No. 10, July-September 1978. | NUREG/CR-1037 |
| Water Chemistry | Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites - Quarterly Progress Report Oct.-Dec. 1979. | NUREG/CR-1325 |
| Water Reactor | Water Reactor Safety Research Division Quarterly Progress Report. April-June 1979. | NUREG/CR-1035 |
| Water Reactor | Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research. October - December 1979. | NUREG/CR-1203 |
| Welding | Estimated Radiation Doses from Thorium and Daughters Contained in Thoriated Welding Electrodes. | NUREG/CR-1039 |
| Weldments | Fatigue of Weldments in Nuclear Pressure Vessels and Piping. | NUREG/CR-1351 |
| Welds | Tensile Properties of Irradiated and Unirradiated Welds of A533 Steel Plate and A508 Forgings. | NUREG/CR-1158 |
| Western Nuclear | Final Environmental Statement Related to the Operation of Split Rock Uranium Mill - Western Nuclear, Inc., Docket No. 40-1162. | NUREG-0639 |
| Wind Tunnel | Rancho Seco Wake Effects on Atmospheric Diffusion: Simulation in a Meteorological Wind Tunnel. | NUREG/CR-1286 |

| Keyword Listing X | Report Title | Report No. |
|---|---|---|
| Xenon-Filled | The Thermal and Mechanical Behavior of Xenon-Filled Fuel Rod as a Function of Burnup. | NUREG/CR-0749 |

| Keyword Listing Y | Report Title | Report No. |
|---|---|---|
| Yellowcake | Radioisotopic Composition of Yellowcake: An Estimation of Stack Release Rate. | NUREG/CR-1216 |

| Numbered Keywords | Report Title | Report No. |
|---|---|---|
| 1978 | LMFBR Safety 7. Review of Current Issues and Bibliography of Literature (1978). | NUREG/CR-1160 |
| 1979 | New Madrid Seismotectonic Study, Activities During Fiscal Year 1979. | NUREG/CR-0977 |
| 1981 | Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program for Fiscal Year 1981. | NUREG 0657 |
| 40 CFR | Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR Part 190). | NUREG-0543 |

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|---|---|---|---|
| ANL-CT-80-10 | NUREG/CR-1285 | FBDU-218-2 | NUREG/CR-1081 |
| ANL-79-105 | NUREG/CR-1230 | HEDL/TME-79-51 | NUREG/CR-1158 |
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| BNL/NUREG-51025 | NUREG/CR-0933 | NRL Rpt 4122 | NUREG/CR-1128 |
| BNL/NUREG-51026 | NUREG/CR-0857 | NRL Rpt 8357 | NUREG/CR-1053 |
| BNL/NUREG-51027 | NUREG/CR-0858 | ORNL/NUREG-58 | NUREG/CR-0675 |
| BNL/NUREG-51044 | NUREG/CR-1007 | ORNL/NUREG-64 | NUREG/CR-1351 |
| BNL/NUREG-51054 | NUREG/CR-1006 | ORNL/NUREG-66 | NUREG/CR-1172 |
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| BNL/NUREG-51083 | NUREG/CR-1037 | ORNL/NUREG/TM-291 | NUREG/CR-0575 |
| BNL/NUREG-51089 | NUREG/CR-1031 | ORNL/NUREG/TM-342 | NUREG/CR-1010 |
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| DE-AC14-76FF02170 | NUREG/CR-1291 | ORNL/NUREG/TM-366 | NUREG/CR-1136 |
| EGG-EA-5044 | NUREG/CR-1205 | ORNL/NUREG/TM-367 | NUREG/CR-1160 |
| EGG-EA-5079 | NUREG/CR-1331 | ORNL/NUREG/TM-383 | NUREG/CR-1352 |
| EGG-EA-5092 | NUREG/CR-1362 | ORNL/Sub-7394/1 | NUREG/CR-1070 |
| EGG-2002 | NUREG/CR-1103 | ORNL/SUB-79/50917/3 | NUREG/CR-1171 |
| EGG-2007 | NUREG/CR-1145 | PNL-2787 | NUREG/CR-1132 |
| EGG-2012 | NUREG/CR-1203 | PNL-2948 | NUREG/CR-0749 |
| EGG-2014 | NUREG/CR-1236 | PNL-2975 | NUREG/CR-0797 |

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| PNL-3040-2 | NUREG/CR-0962 |
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| UCRL-52729 | NUREG/CR-1279 |
| UCRL-52731, Vol.2 | NUREG/CR-1169, Vol.2 |
| UCRL-52738 | NUREG/CR-1312 |
| UFL/COEL/TR/038 | NUREG/CR-1354 |
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U.S. NUCLEAR REGULATORY COMMISSION

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16. ABSTRACT (200 words or less)

This compilation lists all NRC Regulatory and Technical reports published under the NUREG series during the first quarter of 1980.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

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