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Vol. 7, No. 2

Regulatory and Technical Reports

Compilation for Second Quarter 1982 April - June

U.S. Nuclear Regulatory Commission

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Division of Technical Information and Document Control
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PREFACE

This compilation consists of bibliographic data and abstracts for the formal regulatory and technical reports issued by the U.S. Nuclear Regulatory Commission (NRC) Staff and its contractors. It is NRC's intention to publish this compilation quarterly and to cumulate it annually. Your comments will be appreciated. Please send them to:

Division of Technical Information
and Document Control
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Washington, D.C. 20555

The main citations and abstracts in this compilation are listed in NUREG number order: NUREG-XXXX, NUREG/CP-XXXX, and NUREG/CR-XXXX. These precede the following indexes:

Contractor Report Number Index
Personal Author Index
Subject Index
NRC Originating Organization Index (Staff Reports)
NRC Contract Sponsor Index (Contractor Reports)
Contractor Index
Licensed Facility Index

A detailed explanation of the entries precedes each index.

The bibliographic elements of the main citations are the following:

Staff Report

NUREG-0508: MARK II CONTAINMENT PROGRAM EVALUATION AND ACCEPTANCE CRITERIA. ANDERSON, C.J. Division of Safety Technology. August 1981. 90 pp. 8109140048. 09570:200.

Where the entries are (1) report number, (2) report title, (3) report author, (4) organizational unit of author, (5) date report was published, (6) number of pages in the report, (7) the NRC Document Control System accession number, (8) the microfiche address (for internal NRC use).

Conference Report

NUREG/CP-0017: EXECUTIVE SEMINAR ON THE FUTURE ROLE OF RISK ASSESSMENT AND RELIABILITY ENGINEERING IN NUCLEAR REGULATION. JANERP, J.S. Argonne National Laboratory. May 1981. 141 pp. 8105280299. ANL-81-3. 08632:070.

Where the entries are (1) report number, (2) report title, (3) report author, (4) organization that compiled the proceedings, (5) date report was published, (6) number of pages in the report, (7) the NRC Document Control System accession number, (8) the report number of the originating organization, (9) the microfiche address (for NRC internal use).

Contractor Report

NUREG/CR-1556: STUDY OF ALTERNATE DECAY HEAT REMOVAL CONCEPTS FOR LIGHT WATER REACTORS-CURRENT SYSTEMS AND PROPOSED OPTIONS. BERRY, D.L.; BENNETT, P.R. Sandia Laboratories. May 1981. 100 pp. 8107010449. SAND80-0929. 08912:242.

Where the entries are (1) report number, (2) report title, (3) report authors, (4) organizational unit of authors or publisher, (5) date report was published, (6) number of pages in the report, (7) the NRC Document Control System accession number, (8) the report number of the originating organization (if given), and (9) the microfiche address (for NRC internal use).

The following abbreviations are used to identify the document status of a report:

ADD - addendum
APP - appendix
DRFT - draft
ERR - errata
N - number
R - revision
S - supplement
V - volume

Availability of NRC Publications

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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 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.

Main Citations and Abstracts

The report listings in this compilation are arranged by report number, where NUREG-XXXX is an NRC staff originated report, NUREG/CP-XXXX is an NRC sponsored conference report, and NUREG/CR-XXXX is an NRC contractor-prepared report. The bibliographic information (see Preface for details) is followed by a brief abstract of the report.

NUREG-0020 V06 N02: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of January 31, 1982. (Grey Book) * Office of Management and Program Analysis. April 1982. 260pp. 8205120469. 13060:001.

The OPERATING UNITS STATUS REPORT - LICENSED OPERATING REACTORS provides data on the operation of nuclear units as timely and accurately as possible. This information is collected by the Office of Management and Program Analysis from the Headquarters staff of NRC's Office of Inspection and Enforcement, from NRC's Regional Offices, and from utilities. The three sections of the report are: monthly highlights and statistics for commercial operating units, and errata from previously reported data; a compilation of detailed information on each unit, provided by NRC's Regional Offices, IE Headquarters and the utilities; and an appendix for miscellaneous information such as spent fuel storage capability, reactor-years of experience and non-power reactors in the U.S. It is hoped the report is helpful to all agencies and individuals interested in maintaining an awareness of the U.S. energy situation as a whole.

NUREG-0020 V06 N03: LICENSED OPERATING REACTORS. Status Summary Report. February 1982. (Beige Book) * Management Information Branch. June 1982. 259pp. 8207210026. 13994:001.

The OPERATING UNITS STATUS REPORT - LICENSED OPERATING REACTORS provides data on the operation of nuclear units as timely and accurately as possible. This information is collected by the Office of Management and Program Analysis from the Headquarters staff of NRC's Office of Inspection and Enforcement, from NRC's Regional Offices, and from utilities. The three sections of the report are: monthly highlights and statistics for commercial operating units, and errata from previously reported data; a compilation of detailed information on each unit, provided by NRC's Regional Offices, IE Headquarters and the utilities; and an appendix for miscellaneous information such as spent fuel storage capability, reactor-years of experience and non-power reactors in the U.S. It is hoped the report is helpful to all agencies and individuals interested in maintaining an awareness of the U.S. energy situation as a whole.

NUREG-0030 V05 N04: NUCLEAR POWER PLANTS-CONSTRUCTION STATUS REPORT. Data As Of December 31, 1981. (Yellow Book) * Office of

Management and Program Analysis. April 1982. 400pp. 8204220533. 12827:005.

The "Construction Status Report," also referred to as the "Yellow Book," is a quarterly publication containing nuclear power plant construction data and actual progress. The information contained in this report is supplied to the NRC by applicants with Construction Permits.

NUREG-0030 V06 N01: NUCLEAR POWER PLANTS-CONSTRUCTION STATUS REPORT. Data as of March 31, 1981. (Yellow Book) * Management Information Branch. June 1982. 166pp. 8207220657. 14023:212.

The "Construction Status Report," also referred to as the "Yellow Book," is a quarterly publication containing nuclear power plant construction data and actual progress. The information contained in this report is supplied to the NRC by applicants with Construction Permits.

NUREG-0040 V06 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, January 1982-March 1982. (White Book) * Region 4, Office of Director. April 1982. 194pp. 8205060032. 13003:014.

This periodical covers the results of inspections performed by the NRC's Vendor Program Branch that have been distributed to the inspected organizations during the period from January 1982 through March 1982. Also included in this issue are the results of certain inspections performed prior to January 1982 that were not included in previous issues of NUREG-0040.

NUREG-0090 V04 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. Quarterly Report, October-December 1981. * Director's Office. May 1982. 30pp. 8206290543. 13658:321.

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 covers the period October 1 to December 31, 1981.

During the report period, there were two abnormal occurrences at the nuclear power plants licensed to operate. One involved a generic concern pertaining to blockage of coolant flow to safety-related systems. The other involved seismic design errors at Diablo Canyon Nuclear Power Plant with subsequent suspension of the fuel load and low power operating license during the report period; the Agreement States reported no abnormal occurrences to the NRC.

The report also contains information updating a previously reported abnormal occurrence.

NUREG-0304 V03 S01: REGULATORY AND TECHNICAL REPORTS. Compilation For 1975-1978. * Division of Technical Information & Document Control. April 1982. 26pp. 8205110137. 13037:340.

This compilation lists formal staff and contractor reports issued by the U.S. Nuclear Regulatory Commission that were not listed in "Regulatory and Technical Reports for 1975 - 1978," NUREG-0304, Vol. 3. This compilation contains a listing of reports and their abstracts and a keyword index.

NUREG-0519 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF LASALLE COUNTY STATION, UNITS 1 AND 2. Docket Nos. 50-373 And 50-374. (Commonwealth Edison Company, et al.) * Office of Nuclear Reactor Regulation, Director. April 1982. 24pp. 8205040024. 12971:098.

Supplement No. 3 to the Safety Evaluation Report of Commonwealth Edison Company's application for licenses to operate its La Salle County Station, Units 1 and 2, located in Brookfield Township, La Salle County, Illinois has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement addresses several items that have come to light since the previous supplement was issued.

NUREG-0540 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. Documents From October Through December 1974 For Dockets 50-334 Through STN 50-597. * Division of Technical Information & Document Control. June 1982. 175pp. 8207070146. 13782:001.

This document contains a description of information received and generated by the U.S. NRC. This special edition contains Docket 50 material from 1978 that has not appeared in previous editions of the Title List. The documents in this supplement are indexed by personal author, corporate source, and report number.

NUREG-0540 V03 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. December 1-31, 1981. * Division of Technical Information & Document Control. April 1982. 575pp. 8204160034. 12712:063.

This document is a monthly publication containing descriptions of information received and generated by the US NRC. This information includes (1) docketed material associated with civilian nuclear power plants and other uses of radioactive materials and (2) nondocketed material received and generated by NRC pertinent to its role as a regulatory agency. The docketed information includes information formerly issued through the U.S. Department of Energy's Technical Information Center under the title Power Reactor Docket Information (PRDI). This document replaces PRDI, which will no longer be prepared. This document contains the following indexes: Personal Author Index, Corporate Source Index, and Report Number Index.

NUREG-0540 V04 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31, 1982. * Division of Technical Information & Document Control. May 1982. 450pp. 8205200296. 13201:001.

This document is a monthly publication containing descriptions of information received and generated by the US NRC. This information includes (1) docketed material associated with civilian nuclear power plants and other uses of radioactive materials and (2) nondocketed material received and generated by NRC pertinent to its role as a regulatory agency. The docketed information includes information formerly issued through the U.S. Department of Energy's Technical Information Center under the title Power Reactor Docket Information (PRDI). This document replaces PRDI, which will no longer be prepared. This document contains the following indexes: Personal Author Index, Corporate Source Index, and Report Number Index.

NUREG-0540 V04 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. February 1-28, 1982. * Division of Technical Information & Document Control. May 1982. 402pp. 8205270478. 13291:237.

This document is a monthly publication containing descriptions of

NUREG-0304 V06 N04: REGULATORY AND TECHNICAL REPORTS. Compilation For 1981. SAVOLAINEN, A. Division of Technical Information & Document Control. May 1982. 499pp. 8205210507. 13216:278.
This compilation lists all NRC regulatory and technical reports published under the NUREG series during 1981.

NUREG-0304 V07 N01: REGULATORY AND TECHNICAL REPORTS. Compilation For First Quarter 1982. SAVOLAINEN, A. Division of Technical Information & Document Control. May 1982. 143pp. 8206230080. 13593:001.
This compilation lists all NRC regulatory and technical reports published under the NUREG series during the first quarter of 1982.

NUREG-0435 V04 N01: RESEARCH PROJECT CONTROL SYSTEM (RPCS) STATUS SUMMARY REPORT. Research Results Utilization. Data From July 1981-March 1982. (Buff Book) * Office of Nuclear Regulatory Research, Director. April 1982. 226pp. 8205270474. 13290:296.
This report on "Research Results Utilization" provides status and control information concerning the utilization of research results in the regulatory policies and practices of the NRC. Research Information Letters (RILs) are prepared by RES to transmit research results to NRC user offices upon completion of substantial, coherent and reasonably complete bodies of experimental and/or analytical research work. Section 3.0 of this report lists the RILs issued to date, together with an identification of the research program manager and the research program element which generated the RIL. The potential applicability of each RIL to the regulatory process is also identified here, and comments from the cognizant RES and user office staff are summarized which relate to the expected impact of the reported RILs on the regulatory process.

NUREG-0485 V04 N03: SYSTEMATIC EVALUATION PROGRAM STATUS SUMMARY REPORT. Data As Of March 31, 1982. (Buff Book) * Management Information Branch. April 1982. 98pp. 8204220529. 12825:235.
The Systematic Evaluation Program is intended to examine many safety-related aspects of eleven of the older light water reactors. This document provides the existing status of the review process including individual topic and overall completion status.

NUREG-0485 V04 N04: SYSTEMATIC EVALUATION PROGRAM STATUS SUMMARY REPORT. Data As Of April 30, 1982. (Buff Book) * Management Information Branch. May 1982. 97pp. 8206100013. 13475:136.
The Systematic Evaluation Program is intended to examine many safety-related aspects of eleven of the older light water reactors. This document provides the existing status of the review process including individual topic and overall completion status.

NUREG-0485 V04 N05: SYSTEMATIC EVALUATION PROGRAM STATUS SUMMARY REPORT. Data As Of May 31, 1982. (Buff Book) * Management Information Branch. June 1982. 99pp. 8206240030. 13612:166.
The Systematic Evaluation Program is intended to examine many safety-related aspects of 11 of the older light water reactors. This document provides the existing status of the review process including individual topic and overall completion status.

information received and generated by the US NRC. This information includes (1) docketed material associated with civilian nuclear power plants and other uses of radioactive materials and (2) nondocketed material received and generated by NRC pertinent to its role as a regulatory agency. The docketed information includes information formerly issued through the U.S. Department of Energy's Technical Information Center under the title Power Reactor Docket Information (PRDI). This document replaces PRDI, which will no longer be prepared. This document contains the following indexes: Personal Author Index, Corporate Source Index, and Report Number Index.

NUREG-0540 V04 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.March 1-31, 1982.(FOIA Supplement) * Division of Technical Information & Document Control. June 1982. 124pp. 8207060341. 13746:004.

This document is a monthly publication containing descriptions of information received and generated by the U.S. NRC. This information includes (1) docketed material associated with civilian nuclear power plants and other uses of radioactive materials and (2) nondocketed material received and generated by NRC pertinent to its role as a regulatory agency. The docketed information includes information formerly issued through the U.S. Department of Energy's Technical Information Center under the title, Power Reactor Docket Information (PRDI). This document replaces PRDI, which will no longer be prepared. This document contains the following indexes: Personal Author Index, Corporate Source Index, Report Number Index, and Cross Reference to Principal Documents Index.

NUREG-0540 V04 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.March 1-31, 1982. * Division of Technical Information & Document Control. June 1982. 683pp. 8207060342. 13742:001.

This document is a monthly publication containing descriptions of information received and generated by the U.S. NRC. This information includes (1) docketed material associated with civilian nuclear power plants and other uses of radioactive materials and (2) nondocketed material received and generated by NRC pertinent to its role as a regulatory agency. The docketed information includes information formerly issued through the U.S. Department of Energy's Technical Information Center under the title, Power Reactor Docket Information (PRDI). This document replaces PRDI, which will no longer be prepared. This document contains the following indexes: Personal Author Index, Corporate Source Index, Report Number Index, and Cross Reference to Principal Documents Index.

NUREG-0566 V02 N02: STANDARDS DEVELOPMENT STATUS SUMMARY REPORT.Data As Of February 28, 1982. (Green Book) * Internal Information Systems Branch. April 1982. 224pp. 8205040046. 12969:064.

The Standards Development Status Summary Report is designed for scheduling, monitoring, and controlling the process by which Regulatory Standards, Guides, Reports, Petitions, and Environmental Statements are written. It is a summary of the current schedule plans for development of the above products.

NUREG-0580 V11 N01-4: DRAFT REGULATORY LICENSING STATUS SUMMARY REPORT.Data as of January 1 - April 19,1982.(Blue Book) * Management Information Branch. May 1982. 75pp. 8206110008. 13492:196.

Provide a review of the status of the progress of the licensing reviews for all construction permits, operating licenses, special project and non-power reactor renewals under review, as reported to Congress.

NUREG-0580 V11 N05: REGULATORY LICENSING STATUS SUMMARY REPORT. Data As Of May 15, 1982. (Blue Book) * Management Information Branch. June 1982. 53pp. 8206230346. 13595:292.

Provides eview of the status of the progress of the licensing reviews for all construction permits, operating licenses, special projects and non-power reactor renewals under review, as reported to Congress.

NUREG-0606 V04 N02: UNRESOLVED SAFETY ISSUES SUMMARY. Data As Of May 21, 1982. (Aqua Book) * Office of Resource Management, Director. June 1982. 51pp. 8206250018. 11361:363.

Provides an overview of the status of the progress and plans for resolution of the generic tasks addressing "Unresolved Safety Issues" as reported to Congress.

NUREG-0712 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 1 & 2. Docket Nos. 50-361 And 50-362. (Southern California Edison Company) * Office of Nuclear Reactor Regulation, Director. June 1982. 69pp. 8207210141. 13993:075.

Supplement No. 6 to the Safety Evaluation Report for the application filed by Southern California Edison Company, et al for licenses to operate the San Onofre Nuclear Generating Station, Units 2 and 3 (Docket Nos. 50-361 and 50-362) located in San Diego County, California has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement updates the status of review with regard to certain items that were left unresolved in previous supplements and it evaluates several new review items.

NUREG-0725 R02: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL. * Office of Nuclear Material Safety & Safeguards, Director. June 1982. 88pp. 8207020050. 13696:169.

This circular has been prepared in response to numerous requests for information regarding routes used for the shipment of irradiated reactor (spent) fuel subject to regulation by the Nuclear Regulatory Commission (NRC), and to meet the requirements of Public Law 96-295. The NRC staff must approve such routes prior to their first use, in accordance with the regulatory provisions of Section 73.37 of 10 CFR Part 73. The information included reflects NRC staff knowledge as of May 1, 1982. Spent fuel shipment routes, primarily for road transportation, but also including one rail route, are indicated on reproductions of DOT road maps. Also included are the amounts of material shipped during the approximate three year period that safeguards regulations for spent fuel shipments have been effective. In addition, the Commission has chosen to provide information in this document regarding the NRC's safety and safeguards regulations for spent fuel shipments as well as safeguards incidents regarding spent fuel shipments (of which none have been reported to date). This additional information is furnished by the Commission in order to convey to the public a more complete picture of NRC regulatory practices concerning the shipment of spent fuel than could be obtained by the publication of the shipment routes and quantities alone.

NUREG-0748 V02 N03: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of March 31, 1982. (Orange Book) * Management Information Branch. April 1982. 400pp. 8205130235. 13076:341.

The Operating Reactors Licensing Actions Summary is designed to provide the management of the Nuclear Regulatory Commission (NRC) with an overview of licensing actions dealing with operating power and nonpower reactors.

NUREG-0748 V02 N04: OPERATING REACTOR LICENSING ACTIONS SUMMARY. Data As Of April 30, 1982. (Orange Book) * Management Information Branch. May 1982. 239pp. 8206100042. 13477:045.

The Operating Reactors Licensing Actions Summary is designed to provide the management of the Nuclear Regulatory Commission (NRC) with an overview of licensing actions dealing with operating power and nonpower reactors.

NUREG-0748 V02 N05: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of May 31, 1982. (Orange Book) * Management Information Branch. June 1982. 300pp. 8207060336. 13745:005.

The Operating Reactors Licensing Actions Summary is designed to provide the management of the Nuclear Regulatory Commission (NRC) with an overview of licensing actions dealing with operating power and nonpower reactors.

NUREG-0750 V13 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-June 1981. * Division of Technical Information & Document Control. April 1982. 98pp. 8204150571. 12686:172.

Indexes to legal issuances of the Atomic Safety and Licensing Board and Appeal Panels, the Commission, the Administrative Law Judge, and NRC Program Offices.

NUREG-0750 V14 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-September 1981. * Division of Technical Information & Document Control. June 28, 1982. 70pp. 8206290546. 13658:159.

Indexes to legal issuances of the Atomic Safety and Licensing Board and Appeal Panels, the Commission, the Administrative Law Judge, and NRC Program Offices.

NUREG-0786 R01: SITE-SUITABILITY REPORT IN THE MATTER OF THE CLINCH RIVER BREEDER REACTOR PLANT. Docket No. 50-537. * Clinch River Breeder Reactor Program Office. June 1982. 70pp. 8206250025. 13628:017.

The Office of Nuclear Reactor Regulation issued a Site Suitability Report (SSR) for the proposed Clinch River Breeder Reactor Plant (CRBRP) in March 1977. That report documented the results of the staff's evaluation of the suitability of the proposed CRBRP site for a facility of the general size and type as the CRBRP from the standpoint of radiological health and safety considerations. The staff concluded in that report that the proposed CRBRP site was suitable for such a facility.

This report supersedes the March 1977 report. Although a number of changes have occurred since the March 1977 Site Suitability Report was issued, the staff's conclusion in this report remains unchanged. The proposed CRBRP sit is suitable for a facility of the general size and type as the CRBRP fr the standpoint of radiological health and safety considerations.

NUREG-0787 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATERFORD STEAM ELECTRIC STATION, UNIT NO. 3. Docket No. 50-382. (Louisiana Power & Light Company) * Office of Nuclear Reactor Regulation, Director. April 1982. 200pp. 8205190042. 13169:260.

Supplement No. 3 to the Safety Evaluation Report for the application filed by Louisiana Power & Light Company for a license to operate the Waterford Steam Electric Station, Unit 3 (Docket No. 50-382), located in St. Charles Parish, Louisiana has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to provide the staff's evaluation of information submitted by the applicant since the Safety Evaluation Report and Supplement Nos. 1 and 2 were issued. This supplement also includes a copy of the supplemental report by the Advisory Committee on Reactor Safeguards dated March 9, 1982.

NUREG-0793: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF MIDLAND PLANT, UNITS 1 AND 2. Docket Nos. 50-329 And 50-330. (Consumers Power Company) * Office of Nuclear Reactor Regulation, Director. May 1982. 480pp. 8205190027. 13168:001.

The Safety Evaluation Report for the application filed by the Consumers Power Company, as applicant and owner, for a license to operate the Midland Plant, Units 1 and 2 (Docket Nos. 50-329 and 50-330), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located near the city of Midland in Midland County, Michigan. Subject to favorable resolution of the items discusses in this report, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

NUREG-0793 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF MIDLAND PLANT, UNITS 1 AND 2. Docket Nos 50-329 And 50-330. (Consumers Power Company) * Office of Nuclear Reactor Regulation, Director. June 1982. 50pp. 8207150635. 13860:093.

This report supplements the Safety Evalution Report, NUREG-0793, issued May 1982 by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by Consumers Power Company, as applicant and owner, for licenses to operate the Midland Plant, Units 1 and 2 (Docket Nos. 50-329 and 50-330). The facility is located in the city of Midland in Midland County, Michigan. This supplement provides recent information regarding resolution of some of the open items identified in the Safety Evaluation Report and discusses recommendations of the Advisory Committee on Reactor Safeguards in its interim report dated June 8, 1982.

NUREG-0820 DRFT: INTEGRATED PLANT SAFETY ASSESSMENT SYSTEMATIC EVALUATION PROGRAM FOR PALISADES PLANT. Docket No. 50-255. (Consumers Power Company) * Division of Licensing. April 1982. 475pp. 8204160032. 12714:082.

The Integrated Plant Safety Assessment Report for the Consumers Power Company's Palisades Plant (Docket No. 50-255) located in Covert Township, Van Buren County, Michigan, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission (NRC). The report documents the review completed under the Systematic Evaluation Program (SEP). The SEP was initiated by the NRC to review designs of older operating nuclear reactor plants to reconfirm and document their safety. The review has provided for (1) an assessment of the significance of the difference between current technical positions on safety issues and those that existed when the Palisades Plant was licensed, (2) a basis for deciding on how these differences should be resolved in an integrated plant review, and (3) a documentated evaluation of plant safety. Equipment and procedural changes have been identified as a result of the review. It is expected that this report will be one of the bases in considering the conversion of Palisades' provisional operating license to a full-term operating license.

NUREG-0821 DRFT: INTEGRATED PLANT SAFETY ASSESSMENT SYSTEMATIC EVALUATION PROGRAM FOR R. E. GINNA NUCLEAR POWER PLANT. Docket No. 50-244. (Rochester Gas & Electric Corporation) * Division of Licensing. May 1982. 400pp. 8206110322. 13481:302.

The Systematic Evaluation Program was initiated in February 1978 by the U. S. Nuclear Regulatory Commission to review the designs of older operating nuclear reactor plants to reconfirm and document their safety. The review provides (1) an assessment of how these plants compare with current licensing safety requirements relating to selected issues, (2) a basis for deciding on how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety.

This report documents the review of the R. E. Ginna Nuclear Power Plant, owned and operated by Rochester Gas and Electric Corporation (located in Wayne County near Rochester, NY), one of ten plants reviewed under Phase II of this program, and indicates how 137 topics selected for review under Phase I of the program were addressed. Equipment and procedural changes have been identified as a result of the review. It is expected that this report will be one of the bases in considering the issuance of a full-term operating license in place of the existing provisional operating license.

NUREG-0831 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF GRAND GULF NUCLEAR STATION, UNITS 1 & 2. Docket Nos. 50-416 & 50-417. (Mississippi Power And Light Company) * Office of Nuclear Reactor Regulation, Director. June 1982. 100pp. 8207070141. 13781:018.

Supplement No. 2 to the Safety Evaluation Report for Mississippi Power and Light Company, et al, joint application for licenses to operate the Grand Gulf Nuclear Station, Units 1 and 2, located on the east bank of the Mississippi River near Port Gibson, in Claiborne County, Mississippi, 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 resolved at the time of publication of the Safety Evaluation Report.

NUREG-0837 V01 N01-2: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report, January-June 1981. COHEN, L. K.; SLOBODIEN, M. J. Region 1, Office of Director. April 1982. 85pp.

8205130269. 13088:103.
This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of 55 NRC-licensed facility sites throughout the country for the first half of 1981. The program objectives, scope, and methodology are given. The TLD system, dosimeter location, data processing scheme, and quality assurance program are outlined.

NUREG-0837 V01 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report, July-December 1981. COSTELLO, F.; THOMPSON, T.; COHEN, L.K. Region 1, Office of Director. May 1982. 200pp. 8206090230. 13457:343.
This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC-licensed facility sites throughout the country for the second half of 1981.

NUREG-0837 V01 N03-4: Errata To NUREG-0837 Volume 1, Numbers 3-4, Changing Number 3 & Dates Covered To July-September, 1981, to NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. * Region 1, Office of Director. June 23, 1982. 1p. 8207140002. 13847:331.
This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC-licensed facility sites throughout the country for the second half of 1981.

NUREG-0837 V01 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report, October-December 1981. COSTELLO, F.; THOMPSON, T.; COHEN, L.K. Region 1, Office of Director. June 1982. 170pp. 8207140214. 13844:209.
This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facility sites throughout the country for the fourth quarter of 1981.

NUREG-0842: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF ST. LUCIE PLANT, UNIT NO. 2. Docket No. 50-389. (Florida Power & Light Company) * Office of Nuclear Reactor Regulation, Director. April 1982. 245pp. 8205110059. 13039:001.
The Final Environmental Statement related to the operation of the St. Lucie Plant Unit No. 2 by Florida Power and Light Company and Orlando Utilities Commission of the City of Orlando, Florida (Docket No. 50-389), located in St. Lucie County, Florida has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The statement reports on the staff's review of the impact of operation of the plant. Also included are comments of state and federal government agencies and members of the public on the Draft Environmental Statement for this project and staff responses to these comments. The NRC staff has concluded, based on a weighing of environmental, technical and other factors, that an operating license could be granted.

NUREG-0847: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF THE WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority) * Office of Nuclear Reactor Regulation, Director. June 1982. 595pp. 8207210019. 13995:001.

The Safety Evaluation Report for the application filed by the Tennessee Valley Authority, as applicant and owner, for a license to operate the Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Rhea County, Tennessee, near the Watts Bar Dam of the Tennessee River. Subject to favorable resolution of the items discussed in this report, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

NUREG-0848: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF BYRON STATION UNITS 1 AND 2. Docket Nos. STN 50-454 And STN 50-455. (Commonwealth Edison Company) * Office of Nuclear Reactor Regulation, Director. April 1982. 400pp. 8204210626. 12797:150.

The information in this Final Environmental Statement is the second assessment of the environmental impact associated with the construction and operation of the Byron Station, Units 1 and 2, located in Rockvale Township, Ogle County, Illinois, approximately seventeen miles southwest of Rockford, Illinois. The first assessment was the Final Environmental Statement related to construction issued in July 1974 prior to issuance of the Byron Construction Permits. The present assessment is the result of the NRC staff review of the activities associated with the proposed operation of the plant, and includes the staff response to comments on the Draft Environmental Statement.

NUREG-0849: STANDARD REVIEW PLAN FOR THE REVIEW AND EVALUATION OF EMERGENCY PLANS FOR RESEARCH AND TEST REACTORS. BATES, E.F.; GRIMES, B.K.; RAMOS, S.L. Director's Office, Office of Inspection and Enforcement. May 1982. 37pp. 8206020104. 13332:315.

This document provides a Standard Review Plan for the guidance of the NRC staff to assure that complete and uniform reviews are made of research and test reactor emergency plans.

The report is organized under ten planning standards which corresponds to the guidance criteria in Draft II of ANSI/ANS 15.16 as endorsed by Revision 1 to Regulatory Guide 2.6. The applicability of the items under each planning standard is indicated by subdivisions of the steady state thermal power levels at which the reactors are licensed to operate.

Standard emergency classes and example action levels for research and test reactors which should initiate these classes are given in an Appendix.

NUREG-0854: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF CLINTON POWER STATION, UNIT NO. 1. Docket No. 50-461. (Illinois Power Company, et al.) * Office of Nuclear Reactor Regulation, Director. May 1982. 400pp. 8206090137. 13457:051.

This Final Environmental Statement contains the second assessment of the environmental impact associated with operation of the Clinton Power Station, Unit 1, pursuant to the National Environmental Policy Act of 1969 (NEPA) and 10 CFR Part 51, as amended, of the NRC's regulations. This statement examines: the purpose and need for the Clinton project; the affected environment, environmental consequences and mitigating actions, and environmental and economic benefits and costs. The action called for is the issuance of an operating license for Unit 1 of the Clinton Power Station.

NUREG-0857 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PALO VERDE NUCLEAR GENERATING STATION, UNITS 1,2 & 3. Docket Nos. STN 50-528, STN 50-529 & STN 50-530. (Arizona Public Service Company) * Office of Nuclear Reactor Regulation, Director. May 1982. 31pp. 8206090227. 13457:311.

Supplement No. 2 to the Safety Evaluation Report for the application filed by Arizona Public Service Company, et al, for licenses to operate the Palo Verde Nuclear Generating Station, Units 1,2 and 3 (Docket Nos. 50-528/529/530), located in Maricopa County, Arizona has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation Report by providing (1) the evaluation of additional information submitted by the applicant since Supplement No. 1 to the Safety Evaluation was issued and (2) the evaluation of the matters that the staff had under review when Supplement No. 1 was issued.

NUREG-0861: TECHNICAL SPECIFICATIONS FOR LA SALLE COUNTY STATION, UNIT NO. 1. Docket No. 50-373 (Commonwealth Edison Company). * Office of Nuclear Reactor Regulation, Director. April 1982. 484pp. 8205060007. 13008:004.

The La Salle County Station, Unit 1 Technical Specifications were prepared by the U.S. Nuclear Regulatory Commission to 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.

NUREG-0863: SURVEY OF FOREIGN REACTOR OPERATOR QUALIFICATIONS, TRAINING, AND STAFFING REQUIREMENTS. AU, M.L.; DISALVO, R.; MERSCHOFF, E. Division of Facility Operations. May 1982. 500pp. 8205200297. 13198:001.

This report is a compilation of the data obtained from a survey of foreign nuclear power plant operator requirements. Included among the considerations are shifting staffing, operator eligibility, operator training programs, operator licensing or certification, and operator retraining. The data obtained from this survey are presented in matrix form and contrasted with U.S. requirements.

NUREG-0868: A COLLECTION OF MATHEMATICAL MODELS FOR DISPERSION IN SURFACE WATER AND GROUNDWATER. CODELL, R.B. Division of Engineering. KEY, K.J.; WHELAN, G. Battelle Memorial Institute, Pacific Northwest Laboratory. June 1982. 180pp. 8206180340. 13569:044.

This report represents a collection of some of the manual procedures and simple computer programs used by the Hydrological Engineering Section of the Division of Engineering, Office of Nuclear Reactor Regulation, for computing the fate of routinely or accidentally released radionuclides in surface water and groundwater. All models are straightforward simulations of dispersion with constant coefficients in simple geometries.

NUREG-0871 V01 N02: SUMMARY INFORMATION REPORT. January 1, 1982-March 31, 1982. (Brown Book) * Office of Management and Program Analysis. April 1982. 56pp. 8205190021. 13188:077.

Provides summary data concerning NRC and its licensees for general use by the Chairman, other Commissioners and Commission staff offices, the Executive Director for Operations, and the Office Directors.

NUREG-0872: A FEASIBILITY STUDY OF USING LICENSEE EVENT REPORTS FOR A STATISTICAL ASSESSMENT OF THE EFFECT OF OVERTIME AND SHIFT WORK ON OPERATOR ERROR. DISALVO, R.; GERY, A.; PITTMAN, J. Division of Facility Operations. June 1982. 94pp. 8207060004. 13741:121.

A study was made based upon the reported licensed operator errors from January 1981 to determine if a valid statistical determination could be made of the effects of shift work and overtime on operator error. The study concludes that the data reported in the Licensee Event Reports are inadequate to draw conclusions on the influence of overtime and shift work on operator error. The analysis did show that the errors are not uniform over the hours of the day or the days of the week; the causes of the non-uniformity could not be determined.

NUREG-0878: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF WOLF CREEK GENERATING STATION, UNIT 1. Docket No. STN 50-482. (Kansas Gas And Electric Company, et al.) * Office of Nuclear Reactor Regulation, Director. June 1982. 155pp. 8206290526. 13659:104.

The information in this Final Environmental Statement is the second assessment of the environmental impact associated with the construction and operation of the Wolf Creek Generating Station, Unit No. 1, located in Coffey County, Kansas. The Draft Environmental Statement was issued in January 1982. The first assessment was the Final Environmental Statement related to construction issued in October 1975 prior to issuance of the Wolf Creek Construction Permit. The present assessment is the result of the NRC staff's review of the activities associated with the proposed operation of the plant, and includes the staff response to comments on the Draft Environmental Statement.

NUREG-0881: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WOLF CREEK GENERATING STATION, UNIT NO. 1. Docket No. STN 50-482. (Kansas Gas And Electric Company, et al.) * Office of Nuclear Reactor Regulation, Director. April 1982. 800pp. 8204220539. 12826:001.

The Safety Evaluation Report for the application filed by the Kansas Gas and Electric Company, as applicant and agent for the owners, for a license to operate the Wolf Creek Generating Station, Unit 1 (Docket No. STN 50-482), has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The Facility is located in Coffey County, Kansas. Subject to favorable resolution of the items discussed in this report, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

NUREG-0887: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PERRY NUCLEAR POWER PLANTS, UNITS 1 & 2. Docket Nos. 50-440 And 50-441. (Cleveland Electric Illuminating Company) * Office of Nuclear Reactor Regulation, Director. May 1982. 370pp. 8206170055. 13556:001.

This Safety Evaluation Report has been prepared by the Office of

Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission in response to an application filed by the Duquesne Light Company, the Ohio Edison Company, the Pennsylvania Power Company and the Toledo Edison Company (the Central Area Power Coordination Group, CAPCO), as applicants and owners, for a license to operate the Perry Nuclear Power Plant Units 1 and 2 (Docket Nos. 50-440 and 50-441). The facility is located near Lake Erie in Lake County, Ohio. Subject to favorable resolution of the items discussed in this report, the NRC staff concludes that the facility can be operated by the Cleveland Electric Iluminating Company without endangering the health and safety of the public.

NUREG-0891: NUCLEAR PROPERTY INSURANCE: STATUS AND OUTLOOK. LONG, J. D. Office of State Programs, Director. May 1982. 115pp. 8206170048. 13543:233.

The report addresses the problem of the unavailability of adequate levels of property insurance for commercial power reactors to pay for decontamination and cleanup costs arising from accidents. The report is designed to answer six questions, as follows:

1. What has been the development of each principal source of nuclear property insurance used as of early 1982 by nuclear utilities in the United States?
2. What are some of the distinguishing features of nuclear property insurance as offered by the principal sources?
3. How much nuclear property insurance was offered by each of these sources as of January 1, 1982?
4. Assuming that present plans came to fruition, how much property insurance is likely to be offered by each of these sources as of January 1, 1983?
5. What, if any, principal sources of nuclear property insurance are likely to emerge in the private sector by January 1, 1983?
6. What problems serious enough to warrant action of the NRC exist with respect to nuclear property insurance and what actions should NRC take in reponse to each problem?

NUREG-0894: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE CONSTRUCTION OF SKAGIT/HANFORD NUCLEAR PROJECT, UNITS 1 AND 2. Docket Nos. STN 50-522 And STN 50-523. (Puget Sound Power And Light Company, Pacific Power And Light Company, et al.) * Office of Nuclear Reactor Regulation, Director. April 1982. 350pp. 8205120106. 13054:001.

This draft environmental statement contains an assessment of the environmental impact associated with the construction of Skagit/Hanford Nuclear Project, Units 1 and 2 (S/HNP) pursuant to the National Environmental Policy Act of 1969 (NEPA) and 10 CFR 51, as amended, of the NRC's regulations. This statement examines: the purpose and need for the S/HNP project, alternatives to the project, the affected environment, environmental consequences and mitigating actions, and environmental and economic benefits and costs. No water-use impacts are expected from cooling-tower makeup withdrawn from, or blowdown discharged into, the Columbia River. Land-use and terrestrial- and aquatic-ecological impacts will be small. Impacts to historic and prehistoric sites will be negligible with the development and implementation of the applicant's cultural-resources management plan. The risk associated with accidental radiation exposure is very low. The net socio-economic effects of the project will be beneficial. The action called for is the issuance of a construction permit for Skagit/Hanford Nuclear Project, Units 1 and 2.

NUREG-0895: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF SEABROOK STATION, UNITS 1 AND 2. Docket Nos. 50-443 & 50-444. (Public Service Company Of New Hampshire, et al.) * Office of Nuclear Reactor Regulation, Director. May 1982. 400pp. 8205200290. 13200:020.

The information in this statement is the second assessment of the environmental impact associated with the construction and operation of the Seabrook Station, Units 1 and 2, located in the town of Seabrook, New Hampshire. The first assessment was the Final Environmental Statement related to construction, issued in December 1974, prior to issuance of the Seabrook construction permits. The construction of Unit 1 is now 62% complete and commercial operation is scheduled for February 1984. The present assessment is the result of the NRC staff review of the activities associated with the proposed operation of the plant.

NUREG-0902: SITE SUITABILITY, SELECTION AND CHARACTERIZATION BRANCH TECHNICAL POSITION - Low Level Waste Licensing Branch. SIEFKEN, D.; PANGBURN, G.; PENNIFILL, R.; et al. Division of Waste Management. April 1982. 29pp. 8205060014. 13003:208.

The staff provides an expanded interpretation of the site suitability requirements in the proposed rule 10 CFR Part 61, a description of the anticipated site selection process, and a detailed discussion of the site characterization program needed to support a license application and environmental report. The paper provides early-on guidance to prospective applicants in these three subject areas.

NUREG-0903: SURVEY OF INDUSTRY AND GOVERNMENT PROGRAMS TO COMBAT DRUG AND ALCOHOL ABUSE. ALTMAN, W.; BROWN, W.; BUSH, C.; et al. Director's Office, Office of Inspection and Enforcement. June 1982. 76pp. 8206290552. 13658:357.

Report of an NRC survey of the drug and alcohol programs of ten licensed nuclear utilities, two federal agencies, and two large corporations not in the nuclear industry. Report contains management views on the extent of the drug and alcohol problem, policies on work-related use or possession of alcohol or drugs, and views on applicable proposed NRC regulatory initiatives. Report describes practice and perceptions on: use of background investigations, psychological tests, supervisory training and behavioral observation, employee awareness programs, employee assistance and rehabilitation programs, and use of chemical tests and other detection measures.

Report describes a recommended generic Baseline Program for combatting drug and alcohol problems in the nuclear industry, which includes: written drug and alcohol policy, company resolve to exercise the policy, employee awareness program, employee assistance or rehabilitation program, employee screening program, and drug and alcohol detection program.

NUREG-0904: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE DECOMMISSIONING OF THE RARE EARTHS FACILITY, WEST CHICAGO, ILLINOIS. Docket No. 40-2061. (Kerr-McGee Chemical Corporation) * Office of Nuclear Material Safety & Safeguards, Director. May 1982. 400pp. 8205200288. 13197:005.

This Draft Environmental Impact Statement is issued by the U.S. Nuclear Regulatory Commission in response to the plan proposed by Kerr-McGee Chemical Corporation for the decommissioning of their Rare Earths Facility located in West Chicago, Illinois. The statement considers the Kerr-McGee preferred plan and various alternatives to that plan. The action proposed by the Commission is the renewal of the Kerr-McGee license to allow safe storage of the radioactive waste onsite for a period of 5 years. At the end of this period, the following alternatives will be evaluated:

1. Renewal of the license for an additional period of 5 years and the possible imposition of additional conditions or remedial actions.

2. Removal of the material to a licensed low-level waste disposal site.

3. Termination of the license and transfer of the property to federal or state ownership.

NUREG-0909: NRC REPORT ON THE JANUARY 25, 1982 STEAM GENERATOR TUBE RUPTURE AT R. E. GINNA NUCLEAR POWER PLANT. MARTIN, T. T. Division of Engineering & Technical Programs. April 1982. 335pp. 8204210731. 12798:065.

This NRC Task Force report documents the circumstances surrounding the January 25, 1982, steam generator tube rupture event at the R. E. Ginna Nuclear Power Plant. It focuses on the period from 9:25 a.m. on January 25, when the tube rupture occurred, to 10:45 a.m. on January 25, when the plant entered the recovery phase. Information outside this period is recounted as necessary to place the event in perspective. The report is intended to describe factual information and significant findings associated with the event and, thereby, provide the required data base for appropriate detailed analysis and recommendations by various NRC offices.

NUREG-0911: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE WASHINGTON STATE UNIVERSITY TRIGA REACTOR. Docket No. 50-27. * Division of Licensing. May 1982. 64pp. 8206240025. 13610:003.

This Safety Evaluation Report for the application filed by the Washington State University (WSU) for a renewal of operating license number R-76 to continue to operate a research reactor has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is owned and operated by the Washington State University and is located on the WSU campus in Pullman, Whitman County, Washington. The staff concludes that the TRIGA reactor facility can continue to be operated by WSU without endangering the health and safety of the public.

NUREG-0913: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE RESEARCH REACTOR AT THE UNIVERSITY OF FLORIDA. Docket No. 50-83. * Office of Nuclear Reactor Regulation, Director. May 1982. 90pp. 8206240021. 13612:080.

This Safety Evaluation Report for the application filed by the University of Florida (UF) for a renewal of operating license number R-56 to continue to operate their Argonaut-type research reactor has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is owned and operated by the University of Florida and is located on the UF campus in Gainsville, Alachua County, Florida. The staff concludes that the reactor facility can continue to be operated by UF without endangering the health and safety of the public.

NUREG-0916: SAFETY EVALUATION REPORT RELATED TO RESTART OF R.E. GINNA NUCLEAR POWER PLANT. Docket No. 50-244. (Rochester Gas And Electric Corporation) * Office of Nuclear Reactor Regulation, Director. May 1982. 250pp. 8206100045. 13478:034.

This report documents NRC's evaluation of the tube rupture which occurred at the R. E. Ginna Nuclear Power Plant on January 25, 1982. This plant, which is located in Wayne County, New York, is owned and operated by Rochester Gas and Electric Corporation. In NUREG-0916, the staff has determined, based on conclusions reached in Section 10.0, that operation of the Ginna plant would be acceptable subject to the commitments contained in Section 9.0 of that report which have been incorporated into the license as conditions.

NUREG-0916 ERR: SAFETY EVALUATION REPORT RELATED TO THE RESTART OF R.E. GINNA NUCLEAR POWER PLANT. Docket No 50-244. (Rochester Gas And Electric Corporation) * Office of Nuclear Reactor Regulation, Director. May 26, 1982. 2pp. 8206110182. 13493:268.

This report documents NRC's evaluation of the tube rupture which occurred at the R. E. Ginna Nuclear Power Plant on January 25, 1982. This plant, which is located in Wayne County, New York, is owned and operated by Rochester Gas and Electric Corporation. In NUREG-0916, the staff has determined, based on conclusions reached in Section 10.0, that operation of the Ginna plant would be acceptable subject to the commitments contained in Section 9.0 of that report which have been incorporated into the license as conditions.

NUREG-0923: ADVANCE NOTIFICATION OF SHIPMENTS OF NUCLEAR WASTE AND SPENT FUEL: Guidance. * Office of Nuclear Material Safety & Safeguards, Director. June 1982. 22pp. 8206240082. 13607:281.

U.S. Nuclear Regulatory Commission regulations in 10 CFR 70.5b and 73.37(f) require NRC licensees to notify the governor of a state prior to making a shipment of nuclear waste or spent fuel within or through the state. This guidance document was prepared to assist licensees in carrying out those requirements.

NUREG-0925: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF THE TETON PROJECT. Docket No. 40-8781. (Teton Exploration Drilling Company, Incorporated) * Office of Nuclear Material Safety & Safeguards, Director. June 1982. 200pp. 8207140207. 13845:018.

This Draft Environmental Impact Statement is issued by the U.S. Nuclear Regulatory Commission in response to the request by Teton Exploration Drilling, Inc. for the issuance of an NRC Source and Byproduct Material License authorizing operation of the proposed Teton Project to mine uranium in situ by injecting a carbonate/bicarbonate lixiviant into the ore body. The statement considers: (1) alternative of no licensing action, (2) alternative energy sources, and (3) alternatives if uranium ore is mined and refined on the site. The proposed action is to grant a Source and Byproduct Material License to the applicant subject to the stipulated license condition.

NUREG-0926: TECHNICAL SPECIFICATIONS FOR GRAND GULF NUCLEAR STATION, UNIT NO. 1. Docket No. 50-416. (Mississippi Power and Light Company) * Division of Safety Technology. June 1982. 210pp. 8207020058. 13699:331.

The Grand Gulf Nuclear Station, Unit 1 Technical Specifications were prepared by the U.S. Nuclear Regulatory Commission to 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.

NUREG/CP-0022: PROCEEDINGS OF THE SYMPOSIUM ON UNCERTAINTIES ASSOCIATED WITH THE REGULATION OF THE GEOLOGIC DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTE. KOCHER, D.C. Oak Ridge National Laboratory. April 1982. 600pp. 8205110116. CONF-810372. 13049:103.

The primary purpose of this symposium was to provide a forum for wide-ranging discussions on (1) technical aspects related to the development of standards for regulating geologic disposal of high-level radioactive waste, with particular emphasis on the sources and magnitudes of uncertainties associated with current methods for predicting post-closure repository performance and potential health risks to future generations, (2) important licensing and regulatory issues involved in geologic waste disposal, and (3) the current social and political climate in which issues of high-level waste management are being debated. Significant contributions to these discussions were provided by representatives from the U.S. Nuclear Regulatory Commission (NRC), U.S. Department of Energy, U.S. Environmental Protection Agency (EPA), various contractors of these three agencies, and other interested parties not affiliated with the Federal Government or its contractors. The symposium was timed to coincide with the development and publication by the NRC of the proposed technical criteria for regulating the disposal of high-level radioactive wastes in geologic repositories. An additional subject of considerable interest at the symposium was the development of environmental radiation protection standards for high-level radioactive waste by the EPA and the relationship of these standards to the NRC's proposed technical criteria.

NUREG/CP-0026: WORKSHOP ON PSYCHOLOGICAL STRESS ASSOCIATED WITH THE PROPOSED RESTART OF THREE MILE ISLAND, UNIT 1. WALKER, P.; FRAIZE, W.E.; GORDON, J.J.; et al. Mitre Corp. April 1982. 152pp. 8204210661. MTR-82W26. 12799:032.

On 4 and 5 February 1982, eleven experts in the field of psychological stress and related fields met for a two-day Workshop at the MITRE Corporation, McLean, Virginia. The general purpose of the Workshop, sponsored by the Nuclear Regulatory Commission, was to assess the state-of-knowledge relevant to assessing psychological stress which may be associated with the restart of the nuclear power reactor Unit 1 at the Three Mile Island site of the Metropolitan Edison Company (TMI-1). Of particular interest was the extent to which existing concepts and studies might be used to extrapolate or infer the range of stress responses likely to result from the proposed restart of TMI-1. This report summarizes the discussions of the Workshop participants.

NUREG/CP-0031 V01: PROCEEDINGS OF THE CSNI SPECIALIST MEETING ON OPERATOR TRAINING AND QUALIFICATIONS. * Division of Technical Information & Document Control. June 1982. 371pp. 8207210132. CSNI REPT NO. 63. 13992:053.

The events during the accident at TMI-2, along with others identified in retrospect at other nuclear plants, re-emphasized the critical role of the reactor operator. Many countries are focusing greater attention on the capabilities of control room operating staff and on the problems they face. In view of the importance to safety on the subject, the CSNI Subcommittee on Licensing decided that a specialist meeting should be held on the broad aspects of operator selection and training and the functions and organization of operating staff. The meeting focused on the functions, role and organization of control room personnel as a crew and as individuals; selection and qualifications of personnel; operator training and requalification; evaluation of crew and individual performance; professional and career alternatives for control room personnel; and "concepts for the future" (e.g., implementation and impact of computer technology, advanced simulator concepts, off-site monitoring and support). Fourteen countries and three international organizations were represented. This report consists of two volumes.

NUREG/CP-0031 V02: PROCEEDINGS OF THE CSNI SPECIALIST MEETING ON OPERATOR TRAINING AND QUALIFICATIONS. * Division of Technical Information & Document Control. June 1982. 364pp. 8207210130. CSNI REPT NO. 63. 13991:076.

The events during the accident at TMI-2, along with others identified in retrospect at other nuclear plants, re-emphasized the critical role of the reactor operator. Many countries are focusing greater attention on the capabilities of control room operating staff and on the problems they face. In view of the importance to safety on the subject, the CSNI Subcommittee on Licensing decided that a specialist meeting should be held on the broad aspects of operator selection and training and the functions and organization of operating staff. The meeting focused on the functions, role and organization of control room personnel as a crew and as individuals; selection and qualifications of personnel; operator training and requalification; evaluation of crew and individual performance; professional and career alternatives for control room personnel; and "concepts for the future" (e.g., implementation and impact of computer technology, advanced simulator concepts, off-site monitoring and support). Fourteen countries and three international organizations were represented. This report consists of two volumes.

NUREG/CR-0169 V13: LOFT EXPERIMENTAL MEASUREMENTS UNCERTAINTY ANALYSES. Volume XIII.Temperature Measurements. LASSAHN,G.D. EG&G, Inc. April 1982. 44pp. 8205110101. EGG-2037. 13037:142.

Estimates of measurements for thermocouples and resistance thermometer used to measure temperatures in the Loss-of-Fluid Test (LOFT) system during experiments are provided. The estimated uncertainties were obtained by evaluating the temperature measurements to determine possible errors and then combining the errors for each measurement. The evaluation showed that different uncertainity components are important for different temperature measurements and that no one error source is a major source of uncertainty for all the LOFT temperature measurements.

NUREG/CR-0200 ERR: SCALE:MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. * Oak Ridge National Laboratory. April 22, 1982. 1p. 8204230011. 12834:335.

This manual provides documentation for a new, multi-faceted computational system called SCALE (Standardized Computer Analyses for Licensing Evaluation) that has been developed to provide a standard analysis tool for use by the NRC staff and licensees in evaluating nuclear fuel facility and package designs. The SCALE system consists of several automated analytical sequences (control modules) which perform criticality, shielding, and/or heat transfer calculations with a minimum of user-required input. The computer codes (functional modules) used within each analytical sequence can also be run in a stand-alone fashion or coupled together in a sequence determined by the user.

NUREG/CR-1030 V02: SEDIMENT AND RADIONUCLIDE TRANSPORT IN RIVERS. Phase 2-Field Sampling Program For Cattaraugus And Buttermilk Creeks, New York. WALTERS, W. H.; ECKERT, R. M.; ONISHI, Y. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1982. 180pp. 8205060002. PNL-3117. 13002:001.

A field sampling program was conducted on Cattaraugus and Buttermilk Creeks, New York during September 1978 to investigate the transport of radionuclides in surface waters as part of a continuing program to provide data for application and verification of Pacific Northwest Laboratory's (PNL) sediment and radionuclide transport model, SERATRA. Suspended sediment, bed sediment, and water samples were collected during low flow conditions over a 45 mile reach of stream channel. Radiological analysis of these samples included primarily gamma ray emitters; however, six alpha and beta-emitting radionuclides were analyzed using radiochemical methods. The Nuclear Fuel Services facilities are a possible source of two gamma-emitting radionuclides: 1) Cesium-134, and 2) Cesium-137. The principal beta-emitter found was Strontium-90. Elevated levels of both Cesium-137 and Strontium-90 were found at the sampling stations immediately downstream of the facilities. Based on downstream trends of activity levels of other radionuclides, the Nuclear Fuel Field Services facilities may also be a possible source of Plutonium-238 and 239, 240, Americium-241, Curium-244, and Tritium. This field sampling effort is the second of a three phase program to collect hydrologic and radiologic data at three different flow conditions.

NUREG/CR-1233 V04: THE STRUCTURED ASSESSMENT APPROACH. VERSION 1. COMPUTATIONAL ANALYSIS PACKAGE. PARZIALE, A. A.; PATENAUDE, C. J.; RENARD, P. A.; et al. Lawrence Livermore Laboratory. April 1982. 145pp. 8206100031. UCID-18146. 13474:172.

A methodology, called the Structured Assessment Approach (SAA), has been developed to assess the effectiveness of material control and accounting safeguards systems at nuclear fuel cycle facilities. The methodology has been refined into a computational tool, the Version 1 analysis package, that has been used first to assess a hypothetical nuclear fuel cycle facility and more recently to assess operational nuclear plants.

The Version 1 analysis package is designed to analyze safeguards systems that prevent the diversion of Special Nuclear Material from nuclear fuel cycle facilities and to provide assurance that diversion has not occurred.

NUREG/CR-1245 R01: CORRECTIONS AND ADDITIONS TO USER'S GUIDE FOR SNAP. (NUREG/CR-1245, SAND80-0315). POLITO, J. Sandia Laboratories. May 1982. 95pp. 8205190005. SAND82-7017. 13186:259.

This document contains corrections and additions to the "User's Guide for SNAP" (NUREG/CR-1245, SAND82-0315). These update the SNAP report so that it documents the most current version of SNAP. An additional program, BATLE Statistics (BSTAT), is described here. It provides a post-processing capability to analyze engagement data from SNAP simulations.

NUREG/CR-1594 V04: ADVANCED REACTOR SAFETY RESEARCH QUARTERLY REPORT OCTOBER-DECEMBER 1980. * Sandia Laboratories. April 1982. 362pp. 8205040028. SAND80-1646. 12969:288.

The Advanced Reactor Safety Research Program, initiated in FY 1975, is a comprehensive research activity to assure that the necessary safety data and theoretical understanding exists to license and regulate the Liquid Metal Fast Breeder Reactor (LMFBR) or other advanced converters, breeders or advanced light water reactors which may be commercialized in the United States. Recently the emphasis has shifted toward applying advanced reactor safety technology to LWR Class 9 accident concerns which have been of considerable interest following the accident at TMI-2. For FY 1981 the program is organized in the following Tasks, progress on which is reported herein.

Task 1 Advanced Reactor Core Phenomenology,
Task 2 Light Water Reactor (LWR) Severe Core Damage Phenomenology,
Task 3 Core Debris Behavior -- Inherent Retention,
Task 4 Containment Analysis,
Task 5 Elevated Temperature Design Assessment,
Task 6 LMFBR Accident Delineation, and
Task 7 Test and Facility Technology.

NUREG/CR-1622: FLOW MEASUREMENT BY PULSED-NEUTRON ACTIVATION TECHNIQUES AT THE PKL FACILITY AT ERLANGEN (GERMANY). KEHLER, P. Argonne National Laboratory. April 1982. 140pp. 8205130241. ANL-CT-81-35. 13087:001.

Flow velocities in the downcomer at the PKL facility (in Erlangen, Germany) were measured by the Pulsed-Neutron activation (PNA) technique. This was the first time that a fully automated PNA system, incorporating a dedicated computer for on-line data reduction, was used for flow measurements. A prototype of a portable, pulsed, high-output neutron source, developed by the Sandia National Laboratories for the U.S. Nuclear Regulatory Commission, was also successfully demonstrated during this test. The PNA system was the primary flow-measuring device used at the PKL, covering the whole range of velocities of interest. In this series, the PKL simulated small-break accidents similar to the one that occurred at TMI. The flow velocities in the downcomer were, therefore, very low, ranging between 0.03 and 0.35 m/sec.

Two additional flow-measuring methods were used over a smaller range of velocities. Wherever comparison was possible, the PNA-derived velocity values agreed well with the measurements performed by the two more conventional methods.

NUREG/CR-1636 V04: RISK METHODOLOGY FOR GEOLOGIC DISPOSAL OF RADIOACTIVE WASTE: EFFECTS OF VARIABLE HYDROLOGIC PATTERNS ON THE ENVIRONMENTAL TRANSPORT MODEL. BROWN, J.B.; HELTON, J.C. Sandia Laboratories. April 1982. 113pp. 8205030642. SAND79-1909. 12949:033.

The Environmental Transport Model is a compartment model which represents radionuclide movement through a surface hydrologic system. Some of the parameters in the model are based on water and solid flow rates between various compartments in the system. Mean yearly flow rates have been used in the calculation of these parameters, whereas the flow rates are (at best) periodic functions of time or (more realistically) periodic stochastic processes. This report presents the results of an investigation into the effects that these variable hydrologic patterns have on the Environmental Transport Model.

NUREG/CR-1659 V03: REACTOR SAFETY STUDY METHODOLOGY APPLICATIONS PROGRAM: Calvert Cliffs No. 2 PWR Power Plant. HATCH, S. W. ; KOLB, G. O. Sandia Laboratories. CYBULSKIS, P. Battelle Memorial Institute, Columbus Laboratories. June 1982. 240pp. 8206240048. SAND80-1897 V03. 13618:016.

This volume represents the results of the analysis of the Calvert Cliffs Unit 2 Nuclear Power Plant which was performed as part of the Reactor Safety Study Methodology Applications Program (RSSMAP). The RSSMAP was conducted to apply methodology developed in the Reactor Safety Study (RSS) to an additional group of plants with the following objectives: (1) identification of the risk dominating accident sequences for a broader group of reactor designs; (2) comparison of these accident sequences with those identified in the RSS; and (3) based on this comparison, identification of design differences which have a significant impact on risk. Significant use of RSS insights and results was made for the Calvert Cliffs analysis. Loss of coolant accidents (LOCAs) and transients were used as initiating events. The release categories, human error, and component failure data bases were the same as those used in the RSS. The transient and LOCA event trees for Calvert Cliffs differ somewhat from the RSS event trees due to different systems and interactions among systems at Calvert Cliffs. In addition, the RSSMAP transient and LOCA trees are interrelated in recognition that transient initiating events may ultimately lead to LOCA conditions. A "Survey and Analysis" technique was used to identify the most likely failure modes of a system. The determination of which accident sequences result in core melt and the subsequent containment response and release was made by the MARCH and CORRAL codes.

NUREG/CR-1672 V03: RISK ASSESSMENT METHODOLOGY DEVELOPMENT FOR WASTE ISOLATION IN GEOLOGIC MEDIA:Technical Review of NUREG/CR-1636, Vols 1, 2 and 3, December 1, 1981-March 31, 1982. STEVENS, C. A. ; FULLWOOD, R. R. ; AMIRIJAFARI, B. ; et al. Science Applications, Inc. June 1982. 98pp. 8207140108. SAI-288-82-PA. 13846:125.

This project is an ongoing independent technical review of products from an NRC research program to develop a risk-based methodology for assessing the long-term risk of a nuclear waste repository in a geologic medium. This report presents a review of three technical reports on environmental transport modeling of the risk methodology.

NUREG/CR-1681: WRAP-PWR VERIFICATION STUDIES. GREGORY, M. V. ; BERANEK, F. ; AMES, P. L. ; et al. Savannah River Laboratory. May 1982. 110pp. 8206090132. DPST-80-4. 13442:199.

A modular computational system known as the Water Reactor Analysis Package - Evaluation Model (WRAP-EM) was developed for the Nuclear Regulatory Commission (NRC) to interpret and evaluate reactor vendor EM methods and computed results. A subset of the system (WRAP-EM) provides the computational tools to perform a complete analysis of loss-of-coolant accidents (LOCA's) in pressurized water reactors (PWR's). A set of calculations modeling experimental tests in the Semiscale and LOFT facilities, and calculations of a large break in a typical four-loop Westinghouse PWR plant have verified that the WRAP-PWR-EM system is functioning as intended.

NUREG/CR-1820: STATUS REPORT ON THE FISSION-PRODUCT RESEARCH PROGRAM. CUMMINGS, J. C. ; SALLACK, R. A. ; ELRICK, R. M. Sandia Laboratories. April

1982. 59pp. 8204290629. SAND80-2662. 12898:342.

This preliminary report discusses the status of fission-product research conducted through September, 1980 as a part of a program entitled "Separate Effects Tests for TRAP Code Development." We have used transpiration and microbalance techniques to measure vapor pressures, study vapor-vapor and vapor-wall reactions, and measure surface absorption/desorption rates of fission-product species. We are currently constructing a Fission-Product Reaction Facility (FPRF) to study the chemistry of fission-product species in a high-temperature steam environment. A Raman spectroscopy diagnostic setup, for use with the FPRF, is being tested and calibrated with an interim Raman cell.

NUREG/CR-1826 V01: RELAP5/MOD1 CODE MANUAL. Volume 1: System Models And Numerical Methods. RANSOM, V. H.; WAGNER, R. J.; TRAPP, J. A.; et al. EG&G, Inc. April 1982. 129pp. 8205060137. EGG-2070. 13001:107.

The RELAP5/MOD1 code is described in three volumes: Volume 1, System Models and Numerical Methods; Volume 2, Users Guide and Input Requirements; and Volume 3, Checkout Problems Summary. Volume 1 contains technical developments of the basic thermal-hydraulic model, constitutive relations, and solution scheme. The adaptations of the basic model for system components such as pumps, valves, accumulators, and branches are discussed with development of the core neutronics and control system models. Volume 2 gives recommendations on code application and detailed input requirements. Volume 3 summarizes the descriptions and results of example checkout problems to which the RELAP5/MOD1 code was applied. The problems range from simple, separate-effects tests to integral LOFT experiment simulations. Existing data are compared to code results.

NUREG/CR-1826 V02: RELAP5/MOD1 CODE MANUAL. Volume 2: User's Guide And Input Requirements. WAGNER, R. J.; CARLSON, K. E.; TRAPP, J. A.; et al. EG&G, Inc. April 1982. 179pp. 8205060192. EGG-2070. 13000:001.

The purpose of Volume 2 of the RELAP5 documentation is to provide sufficient information to allow application of RELAP5 to thermal-hydraulic systems. This volume assumes that the user has some familiarity with the RELAP5 models described in Volume 1. This volume has two principal parts. The first describes the RELAP5 program from the user's viewpoint. Each model or feature is discussed with emphasis on how the user uses the feature to represent a physical system. Input data requirements, user options, and descriptions of available output are included. A description of the programming features of RELAP5 has not been prepared, so this volume includes some information regarding required files and use of the program transmittal tape. The second part is a detailed description of the input data requirements and format. This information is maintained as a file of 72 character records and a copy of this file is included on the transmittal tape. This information is formatted by TEXTJAB to a report form that can be printed on a Cyber printer with an upper/lower case print train. The detailed input description is presented in Appendix A.

NUREG/CR-1851: REACTOR PHYSICS DESIGN CALCULATIONS FOR THE ACPR UPGRADE. PICKARD, P. S.; ODOM, J. P. Sandia Laboratories. June 1982. 171pp. 8206250033. SAND80-0764. 13620:137.

This report describes the reactor physics calculations performed for the upgrade of the Annular Core Pulse Reactor (ACPR). The ACPR has been in operation since 1967 and has been utilized for a variety of simulation and reactor safety experiments involving both transient and steady-state operations. The limitation in peforming such experiments in the ACPR has been the degree to which realistic reactor safety and nuclear effects simulation conditions could be created. The motivation for the ACPR Upgrade was to increase pulse and steady-state performance with a sufficiently harder neutron energy spectrum to allow a wider range of tests to be performed.

NUREG/CR-1890: ABS, SRSS AND CDF RESPONSE COMBINATION EVALUATION FOR MARK III CONTAINMENT AND DRYWELL STRUCTURES. PHILIPPACOPOULO. Brookhaven National Laboratory. June 1982. 200pp. 8206240054. BNL-NUREG-51328. 13611:001.

The behavior of a representative Mark III containment and its drywell is investigated with respect to their structural capacity when subjected to various load combinations that may be expected during their lifetime. Mathematical models based on finite element idealization procedures are developed and verified. These include three-dimensional finite element models and the so-called stick models of the Mark III containment system. The latter are employed for soil-structure interaction analysis. Various BNL computer codes are utilized to evaluate structural responses. A set of dynamic loads originating from LOCA, SRV and EARTHQUAKE are compiled from reviewing the current literature. The combinations are performed by employing both the Absolute Sum (ABS) and Square-Root-of-the-Sum-of-the-Squares (SRSS) methods. In addition a probabilistic evaluation of the combination outcome is carried out by using a Monte-Carlo technique. This is done by generating cumulative distribution functions (CDF's) expressing the nonexceedance probability (NEP) level of the maxima of the combinations. The results from a large number of combination cases are demonstrated.

NUREG/CR-2000 V01 N2: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of February 1982. * Oak Ridge National Laboratory. April 1982. 61pp. 8205210512. ORNL/NSIC-200. 13215:331.

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of this document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting are described in detail in NRC Regulatory Guide 1.16 and NUREG-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, and keyword indexes follow the summaries. The components and systems are those identified by the utility when the LER form is initiated; the keywords are assigned by the NSIC staff when the summaries are prepared for computer entry.

NUREG/CR-2000 V01 N3: LICENSEE EVENT REPORT (LER) COMPILATION: For Month of March 1982. * Oak Ridge National Laboratory. May 1982. 61pp. 8206110016. ORNL/NSIC-200. 13492:101.

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of this document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory

Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting are described in detail in NRC Regulatory Guide 1.16 and NUREG-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, and keyword indexes follow the summaries. The components and systems are those identified by the utility when the LER form is initiated; the keywords are assigned by the NSIC staff when the summaries are prepared for computer entry.

NUREG/CR-2000 V01 N4: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of April 1982. * Oak Ridge National Laboratory. June 1982. 120pp. 8206220025. ORNL/NSIC-200. 13584:001.

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of this document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting are described in detail in NRC Regulatory Guide 1.16 and NUREG-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, and keyword indexes follow the summaries. The components and systems are those identified by the utility when the LER form is initiated; the keywords are assigned by the NSIC staff when the summaries are prepared for computer entry.

NUREG/CR-2000 V01 N5: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of May 1982. * Oak Ridge National Laboratory. June 1982. 102pp. 8207060291. ORNL/NSIC-200. 13744:001.

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of this document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting are described in detail in NRC Regulatory Guide 1.16 and NUREG-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, and keyword indexes follow the summaries. The components and systems are those identified by the utility when the LER form is initiated; the keywords are assigned by the NSIC staff when the summaries are prepared for computer entry.

NUREG/CR-2015 V04: SEISMIC SAFETY MARGINS RESEARCH PROGRAM PHASE I FINAL REPORT - SOIL STRUCTURE INTERACTION (PROJECT III). JOHNSON, J. J.; MASENIKOV, O. R.; CHEN, J. C.; et al. Lawrence Livermore Laboratory. June 1982. 146pp. 8207060334. UCRL-53021 V04. 13746:116.

There were three objectives of the soil-structures interaction (SSI) project of the Seismic Safety Margins Research Program (SSMRP). They were 1) to model SSI for system analysis, using state-of-the-art analysis techniques; 2) to identify important parameters in the SSI phenomena through sensitivity studies; and 3) to compare analysis techniques. SSI was modeled in the systems analysis by the substructure approach, as implemented in the CLASSI family of computer programs. The CLASSI formulation clearly separates the roles of earthquake, soil, and structures--a basic requirement of the system analysis. The calculative process is extremely efficient, as it must be to perform repeated deterministic analyses simulating earthquake occurrences. The SSI input to the system analysis is detailed.

SSI analysis of the Zion Nuclear Power Plant was examined in relation to modeling decisions concerning the free-field ground motion and idealizing the soil-structure system. Specifying the free-field motion includes location of the control point, frequency characteristics of the control motion, and the spatial variation of motion. Idealizing the soil-structure system entails modeling the soil configuration, dynamic soil behavior, foundations, and structures. A comparison of linear approaches to SSI analysis was performed.

NUREG/CR-2019: THIRD PHASE OF POCKET-SIZED ELECTRONIC DOSIMETER TESTING. FOX, R.A.; HOOKER, C.D.; HOGAN, B.T.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1982. 25pp. 8206090211. PNL-3762. 13457:285.

The experiences of industrial radiographers have indicated that electronic radiation-warning devices become inoperative when they are used under some types of ambient conditions. This report, as a followup to NUREG/CR-0554 and NUREG/CR-1452, documents the nature of tests performed on several additional commercially available models. None of the four models tested passed the test for ruggedness and severe environmental conditions. However, all models passed most of the requirements of a Health Physics Society draft standard of performance specifications for those devices. The test procedures used in the project and the results obtained are discussed. Conclusions from the tests and recommendations concerning potentially useful modifications to existing devices are presented.

NUREG/CR-2022: TECHNICAL REVIEW OF THE DISPERSION AND DOSE MODELS USED IN THE MILDOS COMPUTER PROGRAM. HORST, T.W.; SOLDAT, J.K.; BANDER, T.J. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1982. 30pp. 8205260118. PNL-3772. 13243:042.

This report reviews the technical basis of the models used in the MILDOS computer code. Two major areas are addressed: the models used for atmospheric dispersion, and the models used in the food chain and in human dosimetry.

The atmospheric dispersion review investigates relevant topics, such as diffusion meteorology, plume rise, deposition, and resuspension. The environmental analysis review investigates the food chain model involving retention and translocation assumptions as applicable to the ingestion pathway in humans. In addition, the human dosimetry model used in MILDOS is discussed in terms of all the appropriate potential pathways for human exposure.

Suggested modifications are presented for possible revision of the MILDOS computer program.

NUREG/CR-2039: DYNAMIC COMBINATIONS FOR MARK II CONTAINMENT STRUCTURES. PHILIPPACOPOULO; REICH, M. Brookhaven National Laboratory. June

1982. 170pp. 8206250015. BNL-NUREG-51366. 13627:203.

The behavior of a representative Mark II containment is investigated with respect to its structural capacity when subjected to various load combinations that may be expected during its lifetime. Mathematical models based on finite element idealization procedures are developed and verified. These include three-dimensional finite models and the so-called stick models of the Mark II containment system. The latter are employed for soil-structure interaction analysis. Various BNL computer codes are utilized to evaluate structural responses. A set of loads are compiled from reviewing the current literature. The combinations are performed by employing both the Absolute Summ (ABS) and Sqaure-Root-of-the-Sum-of-the-Sqaures (SRSS) methods. In addition, a probabilistic evaluation of the combination outcome is carried out by using a Monte-Carlo technique. This is done by generating cumulative distribution functions (CDF's) expressing the nonexceedance probability (NEP) level of the maxima of the combinations. The results from a set of 800 combination cases are demonstrated.

NUREG/CR-2053: HEAT TRANSFER ANALYSIS OF THE LWR PRESSURE VESSEL STEEL IRRADIATION CAPSULES IN THE OAK RIDGE RESEARCH REACTOR-PRESSURE VESSEL BENCHMARK FACILITY. SIMAN-TOV, I. I. Oak Ridge National Laboratory. April 1982. 135pp. 8205180123. ORNL/NUREG/TM-4. 13136:317.

The purpose of this study was to determine a design for irradiation capsules for the Light Water Reactor (LWR) Pressure Vessel Wall Simulation (PVWS) experiment in the Poolside Facility of the Oak Ridge Research Reactor. The experiment's structural configuration is based on the actual configuration of an LWR PV wall, the surveillance specimen capsule, and the thermal shield. The design temperature at which the metallurgical test specimens are to be maintained in the experiment is based on an LWR PV operating temperature of 288 degrees C.

A detailed investigation of the thermal behavior of the proposed design configuration was performed to arrive at an optimum flexibility design that will ensure a uniform temperature distribution of 288 degrees C plus or minus 10 degrees C for all the test specimens, while allowing for uncertainties in thermal behavior, component dependability, and nuclear heating rates in the iron.

The conclusions of these studies determine the final design parameters for these irradiation capsules.

NUREG/CR-2059: COMPILATION OF DATA CONCERNING KNOWN AND SUSPECTED WATER HAMMER EVENTS IN NUCLEAR POWER PLANTS (CY 1969-MAY 1981). CHAPMAN, R. L.; CHRISTENSEN, D.; DAFOE, R. E.; et al. EG&G, Inc. May 1982. 100pp. 8206100064. EGG-CAAD-5629 13481:006.

This report compiles data concerning known and suspected water hammer events reported by BWR and PWR nuclear power plants in the United States from January 1969 to May 1981. This information is summarized for each event and is tabulated for all events by plant, plant type, year of occurrence, type of water hammer, system affected, basis/cause for the event, and damage incurred. Information is also included from other events not specifically identified as water hammer related. The events involve vibration and/or system components similar to those involved in the water hammer events. These other events are included to ensure completeness of the report, but are not used to point out particular facts or trends. Also, this report does not evaluate findings which can be abstracted from the data.

This report shows a total of 81 BWR and 67 PWR occurrences having been reported as water hammer induced over a 12-year period. Of these, approximately half occurred during preoperational testing, or the first year of commercial operation. The remainder occurred during normal plant operation, operational surveillance testing and/or maintenance. The report provides event summaries and corrective action taken to prevent reoccurrence.

NUREG/CR-2099: COMMON CAUSE FAULT RATES FOR DIESEL GENERATORS:ESTIMATES BASED ON LICENSEE EVENT REPORTS AT U.S. COMMERCIAL NUCLEAR POWER PLANTS,1976-1978. ATWOOD,C.L.; STEVENSON,J.A. EG&G, Inc. June 1982. 87pp. 8207060337. 13744:106.

This report presents estimates of common cause fault rates and related quantities, based on Licensee Event Reports for diesel generators in nuclear reactors. The Licensee Event Report data base is described. For estimating rates, the binomial failure rate model is used, extending to allow for the substantial observed plant-to-plant variability, and for shocks that by their nature make all the diesel generators in a plant inoperable. Every quantity is estimated by both a point estimate and a 90% interval. All rates are expressed per calendar hour.

NUREG/CR-2133: BWR REFILL-RELOAD PROGRAM TASK 4.4 - 30 SSTF DESCRIPTION DOCUMENT. BARTON,J.E.; SCHUMACHER,D.G.; FINDLAY,J.A.; et al. General Electric Co. May 1982. 115pp. 8206140333. EPRI NP-1584. 13508:065.

The 30 degrees Steam Sector Test Facility (SSTF), located at General Electric's Lynn, Massachusetts plant, is a mockup of a 30 degrees sector of a GE boiling water reactor (BWR). Its purpose is to provide a data base for assessment of best estimate models and identification and evaluation of controlling phenomena during the refill phase of a hypothesized BWR loss-of-coolant accident. This report describes the design, construction, and operation of the SSTF.

NUREG/CR-2141 V04: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Quarterly Progress Report For October-December 1981. WHITMAN,G.D.; BRYAN,R.H. Oak Ridge National Laboratory. May 1982. 145pp. 8206100030. ORNL/TM-8252. 13472:155.

The Heavy-Section Steel Technology Program is an engineering research activity conducted by the Oak Ridge National Laboratory for the Nuclear Regulatory Commission. The program comprises studies related to all areas of the technology of materials fabricated into thick-section primary-coolant containment systems of light-water-cooled nuclear power reactors. The investigation focuses on the behavior and structural integrity of steel pressure vessels containing crack-like flaws. Current work is organized into six tasks: (1) program administration and procurement, (2) fracture mechanics analyses and investigations, (3) investigations of irradiated materials, (4) thermal-shock investigations, (5) pressure vessel investigations, and (6) steel cladding investigations.

Thermal strain modification to two- and three-dimensional fracture mechanics were checked. Subcontractors investigated fracture initiation and arrest toughness and the transition from cleavage to fibrous fracture. Investigation of properties of irradiated steel included statistical analysis of Charpy data and continued irradiation of specimens. Thermal-shock experiment TSE-6 was conducted. Welds in intermediate test vessel V-8A and testing of material characterization specimens were completed. Further analyses of pressurized thermal-shock test concepts were made. Work commenced on a study of the effects of weld overlay cladding on fracture behavior.

NUREG/CR-2172: SUMMARY AND BIBLIOGRAPHY OF SAFETY-RELATED EVENTS AT BOILING-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1980. MCCORMACK, K. E.; GALLAHER, R. B. Oak Ridge National Laboratory. May 1982. 199pp. 8206110011. ORNL/NSIC-195. 13490:352.

This document presents a bibliography that contains 100-word abstracts of event reports submitted to the U.S. Nuclear Regulatory Commission concerning operational events that occurred at boiling-water-reactor nuclear power plants in 1980. The 1547 abstracts included on microfiche in this bibliography describe incidents, failures, and design or construction deficiencies that were experienced at the facilities. These abstracts are arranged alphabetically by reactor name and then chronologically for each reactor. Full-size keyword and permuted-title indexes to facilitate location of individual abstracts are provided following the text. Tables that summarize the information contained in the bibliography are also provided. The information in the tables includes a listing of the equipment items involved in the reported events and the associated number of reports for each item. Similar information is given for the various kinds of instrumentation and systems, causes of failures, deficiencies noted, and the time of occurrence (i.e., during refueling, operation, testing, or construction). Some of the more interesting events that occurred during the year are reviewed in detail.

NUREG/CR-2172 ERR: SUMMARY AND BIBIOLOGRAPHY OF SAFETY-RELATED EVENTS AT BOILING-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1980. * Oak Ridge National Laboratory. May 10, 1982. 1p. 8206140327. 13509:048.

This document presents a bibliography that contains 100-word abstracts of event reports submitted to the U.S. Nuclear Regulatory Commission concerning operational events that occurred at boiling-water-reactor nuclear power plants in 1980. The 1547 abstracts included on microfiche in this bibliography describe incidents, failures, and design or construction deficiencies that were experienced at the facilities. These abstracts are arranged alphabetically by reactor name and then chronologically for each reactor. Full-size keyword and permuted-title indexes to facilitate location of individual abstracts are provided following the text. Tables that summarize the information contained in the bibliography are also provided. The information in the tables includes a listing of the equipment items involved in the reported events and the associated number of reports for each item. Similar information is given for the various kinds of instrumentation and systems, causes of failures, deficiencies noted, and the time of occurrence (i.e., during refueling, operation, testing, or construction). Some of the more interesting events that occurred during the year are reviewed in detail.

NUREG/CR-2173: SUMMARY AND BIBLIOGRAPHY OF SAFETY-RELATED EVENTS AT PRESSURIZED-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1980. MCCORMACK, K. E.; GALLAHER, R. B. Oak Ridge National Laboratory. May 1982. 270pp. 8206110001. ORNL/NSIC-196. 13491:189.

This report summarizes the data contained in reports submitted by licensees to the U.S. Nuclear Regulatory Commission concerning operational events that occurred at pressurized-water-reactor power plants in 1980. A bibliography containing 100-word abstracts of the event reports is included. The 21666 abstracts describe the incidents, failures, and design or construction deficiencies experienced at the facilities. They are arranged alphabetically by reactor name and then chronologically for each reactor. Keyword and permuted-title indexes are provided to facilitate location of the abstracts of interest. Tables summarizing the information contained in the bibliography are also presented and discussed. Information listed in the tables includes instrument failures, equipment failures, system failures, causes of failures, deficiencies noted, and time of occurrence (i.e., during refueling, operation, testing, or construction). Some of the more interesting events that occurred during the year are reviewed in detail.

NUREG/CR-2173 ERR: SUMMARY AND BIBLIOGRAPHY OF SAFETY-RELATED EVENTS AT PRESSURIZED-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1980. MCCORMACK, K.E.; GALLAHER, R.B. Oak Ridge National Laboratory. May 10, 1982. 1p. 8206110002. ORNL/NSIC-196. 13493:284.

This report summarizes the data contained in reports submitted by licensees to the U.S. Nuclear Regulatory Commission concerning operational events that occurred at pressurized-water-reactor power plants in 1980. A bibliography containing 100-word abstracts of the event reports is included. The 21666 abstracts describe the incidents, failures, and design or construction deficiencies experienced at the facilities. They are arranged alphabetically by reactor name and then chronologically for each reactor. Keyword and permuted-title indexes are provided to facilitate location of the abstracts of interest. Tables summarizing the information contained in the bibliography are also presented and discussed. Information listed in the tables includes instrument failures, equipment failures, system failures, causes of failures, deficiencies noted, and time of occurrence (i.e., during refueling, operation, testing, or construction). Some of the more interesting events that occurred during the year are reviewed in detail.

NUREG/CR-2181 V04: PHYSICS OF REACTOR SAFETY. Quarterly Report, October-December 1981. * Argonne National Laboratory. April 1982. 28pp. 8205030657. ANL-81-29. 12928:215.

This quarterly progress report summarizes work done during the months of October - December 1981. The work in the Applied Physics Division includes reports on reactor safety modeling and assessment by members of the Reactor Safety Appraisals Section. Work on reactor core thermal-hydraulics is performed in ANL's Components Technology Division, emphasizing 3-dimensional code development for LMFBR accident under natural convection conditions. An executive summary is provided including a statement of the findings and recommendations of the report.

NUREG/CR-2184: COMPARISON OF THE RADIOLOGICAL IMPACTS OF THORIUM AND URANIUM NUCLEAR FUEL CYCLES. MEYER, H.R.; WITHERSPOON, J.; MCBRIDE, J.P.; et al. Oak Ridge National Laboratory. April 1982. 33pp. 8205060091. ORNL/TM-7868. 13002:238.

A study is being performed for the Nuclear Regulatory Commission (NRC) to determine whether the existing regulations for the uranium fuel cycles require modification and/or additions in order to regulate thorium fuel cycles. This report was prepared during Phase 2 of the study and compares the radiological impacts of a fuel cycle in which only uranium is recycled, as presented in the Final Generic Environmental Statement on the "Use of Recycled Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors (GESMO)," with those of the light-water breeder reactor (LWBR) thorium/uranium fuel cycle in the "Final Environmental Statement, Light Water Breeder Reactor Program." The significant offsite radiological impacts from routine operation of the fuel cycles result from the mining and milling of thorium and uranium ores, reprocessing spent fuel, and reactor operations. The major difference between the impacts from the two fuel cycles is the larger dose commitments associated with current uranium mining and milling operations as compared to thorium mining and milling. Estimated dose commitments from the reprocessing of either fuel type are small and show only moderate variations for specific doses. No significant differences in environmental radiological impact are anticipated for reactors using either of the fuel cycles. Radiological impacts associated with routine releases from the operation of either the thorium or uranium fuel cycles can be held to acceptably low levels by existing regulations.

NUREG/CR-2192 V01 N2: EVALUATION OF ISOTOPE MIGRATION-LAND BURIAL. Quarterly Progress Report, April-June 1981. CZYSCINSKI, K. S.; PIETRZAK, R. F.; WEISS, A. J. Brookhaven National Laboratory. May 1982. 35pp. 8206100089. 13480:259.

Results are reported for radionuclide sorption experiments performed under anaerobic conditions and as a function of solution/solid ratio for trench shale and waters collected at the Maxey Flats disposal site in Kentucky. The observed degree of sorption (equilibrium K(d)) varied unpredictably as a function of solution to solid ratio. Measurements of pH and Eh were performed before and after the determinations to determine if redox conditions were altered significantly during the experiments. The experimental procedure appears capable of maintaining anaerobic conditions during most of the determinations. Changes in solution/solid ratio appear to affect the observed equilibrium sorption more than any variations in redox state during the determinations. However, our final evaluation of the proposed test procedure for measuring sorption of radionuclides from anoxic groundwater is that the test is not completely reliable. Since further improvements in the experimental procedure are not planned, this type of batch sorption test for anoxic waters will be terminated. Organo-radionuclide complex stability experiments in controlled environment chambers were completed. Controlled oxidation experiments using disposal site trench waters were initiated. Preliminary results suggested that high contents of dissolved ferrous iron in trench waters can act as redox buffers to preserve low redox conditions during subsurface migration. Data on coprecipitation of radionuclides on ferric oxyhydroxide will be reported when analyses are completed.

NUREG/CR-2193 V01 N2: PROPERTIES OF RADIOACTIVE WASTES AND WASTE CONTAINERS. Quarterly Progress Report, April-June 1981. MORCOS, N.; WEISS, A. J. Brookhaven National Laboratory. May 1982. 77pp. 8206100060. BNL-NUREG-51410. 13480:292.

An empirical relationship has been developed to estimate the cumulative fractional releases of (137)Cs from simulated waste forms as a function of leaching time and the geometric surface-to-volume ratios. Data from an ongoing leaching study were used. The simulated waste forms consisted of organic cation exchange resins solidified in Portland I cement at a waste-to-cement ratio of 0.6 and water-to-cement ratio of 0.4. The nominal specimen dimensions were: 1-inch diameter x

1-inch high, 2-inch diameter x 2-inch high, 2-inch diameter x 4-inch high, 3-inch diameter x 3-inch high, 6-inch diameter x 6-inch high, 6-inch diameter x 12-inch high, and 12-inch diameter x 12-inch high. The waste forms were leached in deionized water using a modified IAEA leaching procedure.

A study designed to evaluate the leachability of (137)Cs, (85)Sr, and (60)Co from simulated boric acid waste solidified in Portland III cement and to measure the compressive strength of the ensuing waste forms before and after leaching was concluded. Leaching data extending over 229 days are presented. The simulated waste forms were leached in deionized water using a modified IAEA leaching procedure. The compressive strength of the specimens was measured initially and after their exposure to a leaching environment for 352 days.

NUREG/CR-2194: CONTAINMENT RESEARCH PRIORITIES. SCICCA,F.W. Sandia Laboratories. April 1982. 200pp. 8204150560. SAND81-1370. 12692:001.

This report presents the results of efforts to establish priorities among key areas of LMFBR containment research. The research areas are concerned primarily with those phenomena and events that follow from whole-core accidents that can result in a challenge to the primary and/or secondary containment of an LMFBR. They are concerned with the accident progression and key factors or aspects which may alter or mitigate the accident progression. The evaluation was divided into two categories as follows:

A. Primary containment areas:
   1. Fuel debris location and configuration following core damage.
   2. In-vessel fuel debris coolability and characteristics.
   3. In-vessel core retention structure effectiveness.
   4. Post-Accident Heat Removal effectiveness: global heat removal from damaged fuel.
   5. Energetic recriticality outside of core region.
   6. Primary containment failure modes and failure characteristics.

B. Secondary containment research areas:
   1. Debris bed coolability, location, and characteristics.
   2. Fuel-steel-concrete interactions.
   3. Sodium-concrete interactions.
   4. Core retention structure/sacrificial material effectiveness.
   5. Post-Accident Heat Removal (PAHR) effectiveness.
   6. Gas, vapor and aerosol conditions and behavior.

Three separate efforts were employed in the overall evaluation of containment research priorities. This was done to provide at least a partial check on the consistency and validity of the results obtained. The most extensive evaluation process employed a weighted-score ranking scheme. In this process, each of the candidate research areas was compared against three evaluation criteria.

NUREG/CR-2201: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1978. PELOQUIN,R.A.; SCHWAB,J.D.; BAKER,D.A. Battelle Memorial Institute, Pacific Northwest Laboratory. June 1982. 123pp. 8207080063. PNL-4039. 13795:001.

Population radiation dose commitments have been estimated from reported radionuclide releases from commercial power reactors operating during 1978. Fifty-year dose commitments from a one-year exposure were calculated from both liquid and atmospheric releases for four population groups (infants, child, teen-ager and adult) residing between 2 and 80 km from each site. This report tabulates the results of these calculations, showing the dose commitments for both liquid and airborne pathways for each age group and organ. Also included for each site is a histogram showing the fraction of the total population within 2 to 80 km around each site receiving various average dose commitments from the airborne pathways. The total dose commitment from both liquid and airborne pathways ranged from a high of 200 person-rem to a low of 0.0004 person-rem with an arithmetic mean of 14 person-rem. The total population dose for all sites was estimated at 660 person-rem for the 93 million people considered at risk. The average individual dose commitment from all pathways on a site basis ranged from a low of 3 x 10-6 mrem to a high of 0.08 mrem. No attempt was made in this study to determine the maximum dose commitment received by any one individual from the radionuclides released at any of the sites.

NUREG/CR-2204 V04: ADVANCED TWO-PHASE FLOW INSTRUMENTATION PROGRAM. Quarterly Progress Report, October-December 1981. HARDY, J. E.; ROGERS, S. C.; MILLER, G. N.; et al. Oak Ridge National Laboratory. May 1982. 26pp. 8206100054. ORNL/TM-8231. 13471:341.

The performance of the Westinghouse Rector Vessel Level Indicating System (RVLIS) during tests S-UT-6 and S-UT-7 (5% cold-leg breaks in the Semiscale Test Facility) was analyzed. The RVLIS, a system employing differential pressure (dP) cells, gave estimates of vessel level similar to those of Semiscale level instrumentation when measuring over equal spans. These RVLIS measurements are conservative to vessel coolant levels for both S-UT-6 and S-UT-7. At times, the RVLIS indications are greater than the vessel collapsed liquid level measured by Semiscale instrumentation. During S-UT-6, level estimate differences between RVLIS and Semiscale dPs of up to 215 cm (85 in.) were observed. These discrepancies may be explained by differences in Semiscale and Westinghouse pressurized-water reactor internal designs. Excellent agreement was noted between Semiscale and Westinghouse vessel levels for S-UT-7, an upper-head injection test.

NUREG/CR-2220 V02: THE IMPACT OF ENTRAINMENT AND IMPINGEMENT ON FISH POPULATIONS IN THE HUDSON RIVER ESTUARY. BARNTHOUSE, L. W.; VAN WINKLE, W.; GOLUMBEK, J.; et al. Oak Ridge National Laboratory. May 1982. 165pp. 8206100043. ORNL/NUREG/TM-3. 13476:023.

The purpose of this three-volume report is to publish the individual pieces of testimony involving ORNL staff in a three-year adjudicatory hearing on the effects of electric power generation on the Hudson River. Volume II contains four exhibits relating to impingement impacts and three critiques of certain aspects of the utilities' case. The first exhibit is a quantitative evaluation of four sources of bias (collection efficiency, reimpingement, impingement on inoperative screens, and impingement survival) affecting estimates of the number of fish killed at Hudson River power plants. The following two contain, respectively, a detailed assessment of the impact of impingement on the Hudson River white perch population and estimates of conditional impingement mortality rates for seven Hudson River fish populations. The fourth exhibit is an evaluation of the engineering feasibility and potential biological effectiveness of several types of modified intake structures proposed as alternatives to cooling towers for reducing impingement impacts. This volume also consists of critical evaluations of the utilities' empirical evidence for the existence of density-dependent growth in young-of-the-year striped bass and white perch, the estimate of age-composition of striped bass spawning stock in the Hudson River, and their use of the Lawler, Matusky, and Skelly

(LMS) Real-Time Life Cycle model to estimate the impact of entrainment and impingement on the Hudson River striped bass population.

NUREG/CR-2220 V03: THE IMPACT OF ENTRAINMENT AND IMPINGEMENT ON FISH POPULATIONS IN THE HUDSON RIVER ESTUARY. GOODYEAR, C. P.; KIRK, B. L.; CHRISTENSEN, S. Oak Ridge National Laboratory. April 1982. 400pp. 8204290485. ORNL/NUREG/TM-3. 12893:146.

The purpose of this three-volume report is to publish the individual pieces of testimony involving ORNL staff in a three-year adjudicatory hearing on the effects of electric power generation on the Hudson River.

Volume III addresses the validity of the utilities' use of the Ricker stock-recruitment model to extrapolate the combined entrainment-impingement losses of young fish to reductions in the equilibrium population size of adult fish. In our testimony, a methodology was developed and applied to address a single fundamental question: if the Ricker model really did apply to the Hudson River striped bass population, could the utilities' estimates, based on curve-fitting, of the parameter alpha (which controls the impact) be considered reliable? The present Volume III includes, in addition, an analysis of the efficacy of an alternative means of estimating alpha, termed the technique of prior estimation of beta (used by the utilities in a report prepared for regulatory hearings on the Cornwall Pumped Storage Project). Our validation methodology should also be useful in evaluating inferences drawn in the literature from fits of stock-recruitment models to data obtained from other fish stocks.

NUREG/CR-2221 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH. Quarterly Progress Report, October 1-December 31, 1981. BALL, S. J.; CLEVELAND, J. C.; HARRINGTON, R. M.; et al. Oak Ridge National Laboratory. June 1982. 23pp. 8206220101. ORNL/TM-8260. 13585:023.

Report covers progress during Oct. - Dec. 1981 under the High-Temperature Gas-Cooled Reactor (HTGR) Systems and Safety Analysis Program. Work continued on code development and verification activities and included improvements in the ORTAP code steam line model and the ORECA code capabilities for long-term transients. A preliminary severe accident sequence analysis exercise is presented that includes reactor building release source term, atmospheric dispersion, and radiation exposure calculations.

NUREG/CR-2223: AN EVALUATION OF THE SOLID ANGLE METHOD USED IN NUCLEAR CRITICALITY SAFETY. THOMAS, J. T. Oak Ridge National Laboratory. June 1982. 107pp. 8206230319. ORNL/CSD/TM-158. 13603:232.

The solid angle method has long been used to establish safe spacings for subcritical units of fissile materials, especially uranium with a low (235)U content. Analytic representation of criticality in terms of the total solid angle subtended by the unit nearest the center of an array has permitted an evaluation of the margin of subcriticality implicit in an allowable total solid angle, omega(A). It is shown that the method cannot have general applicability but is dependent upon the type of fissile material, the number and specific arrangement of the units in array, and the array reflector conditions. The method is principally one of comparison. The relative difference between the allowed total solid angle and the total solid angle corresponding to criticality is a measure of the safety. This study demonstrates that the arbitrary application of an omega(A) to an array of fissile material without having established the magnitude of the margin of subcriticality is questionable. The method is usable provided the area of applicability is defined by a validated method.

NUREG/CR-2229 V01: BWR LARGE BREAK SIMULATION TESTS--BWR BLOWDOWN/EMERGENCY CORE COOLING PROGRAM. LEE, L. S. ; SOZZI, G. L. ; ALLISON, S. A. General Electric Co. April 1982. 185pp. 8205120138. EPRI NP-1783. 13055:222.

The BD/ECC Program is an experimentally based program jointly sponsored by the Nuclear Regulatory Commission, The Electric Power Research Institute, and The General Electric Company. The BD/ECC 1A Test Phase of this program involves investigating the integral systems effects of emergency core coolant injection during a hypothetical LOCA. Tests were conducted in a BWR system simulator, the Two-Loop Test Apparatus (TLTA), which features a full-sized electrically heated bundle. Fluid delivery systems were included to simulate emergency coolant injections.

Tests conducted under this program include large break (design basis accident), small break, and core uncovery under slow loss-of-coolant (boil-off) transients. Three separate topical reports are issued, one for each type of test. This topical covers the large break results.

NUREG/CR-2231: BWR LOW FLOW BUNDLE UNCOVERY TEST AND ANALYSIS. SEELY, D. S. ; MURALIDHARAN, R. General Electric Co. April 1982. 200pp. 8205200302. EPRI NP-1781. 13196:004.

A series of separate effects tests was performed to evaluate bundle heat transfer and thermohydraulic flow conditions in a simulated BWR/6 core during a boil-off scenario. The tests were conducted in the Two-Loop Test Apparatus. The tests were run using constant bundle powers (near decay heat levels) and at constant pressures to determine the effects of power and pressures on bundle response. The resultant measured and derived thermohydraulic quantities (such as axial void distributions, two-phase levels, and heat transfer coefficients) are compared to the predictions by current thermal hydraulic analysis methods. In general, the predicted quantities agree closely with the test measurements.

NUREG/CR-2238 V01: ADVANCED REACTOR SAFETY RESEARCH. Quarterly Report, January-March 1981. * Sandia Laboratories. June 1982. 142pp. 8206100021. SAND81-1529 V01. 13473:024.

Sandia Laboratories' Advanced Reactor Safety Research Program, initiated in FY 1975, is a comprehensive research activity conducted as part of the NRC's confirmatory research effort to assure that the necessary safety data and theoretical understanding exist to license and regulate the Liquid Metal Fast Breeder Reactor (LMFBR) or other advanced converters, breeders or advanced light water reactors which may be commercialized in the United States. A portion of the early effort in the program was directed toward obtaining data to support the licensing review of the Clinch River Breeder Reactor (CRBR) and the Fast Flux Test Facility (FFTF). Recently the emphasis has shifted toward applying advanced reactor safety technology to LWR Class 9 accident concerns which have been of considerable interest following the accident at TMI-2. For FY 1981, the program is organized in the followng subtasks, progress on which is reported herein. Task 1, Core

Debris Behavior - Inherent Retention, Task 2, Containment Analysis, Task 3, Elevated Temperature Design Assessment, Task 4, LMFBR Accident Delineation, Task 5, Advanced Reactor Core Phenomenology, Task 6, Light Water Reactor (LWR) Severe Core Damage Phenomenology, and Task 7, Test and Facility Technology.

NUREG/CR-2279: WATER RELEASE FROM HEATED CONCRETES. KENT, L. A. Sandia Laboratories. May 1982. 30pp. 8206090115. SAND81-1732. 13442:309.

Water release from three concretes as a function of temperature has been determined experimentally. Limestone concrete releases more water at a moderate temperature than do magnetite or basalt concretes. The amount of water in the concrete is 6.2%, 6.3%, and 5% by weight for limestone, basalt and magnetite concretes respectively. All of the concretes show three distinct weight losses as a function of temperature. By 450K, 52 to 75 percent of the water is lost--all of the water is lost by 750-800K.

NUREG/CR-2281 V02: NUCLEAR REACTOR SAFETY. April 1-June 30, 1981. STEVENSON, M. G. Los Alamos Scientific Laboratory. April 1982. 28pp. 8205110125. LA-9209-PR. 13037:309.

The work that is highlighted here represents accomplishments for the period April 1 - June 30, 1981 by the groups at Los Alamos involved in reactor safety research for the Division of Accident Evaluation, Office of Nuclear Regulatory Research of the US Nuclear Regulatory Commission. Presented are brief overviews compiled by project, along with a bibliography of Technical Notes and publications written during this quarter.

NUREG/CR-2281 V03: NUCLEAR REACTOR SAFETY. July 1-September 30, 1981. STEVENSON, M. G. Los Alamos Scientific Laboratory. April 1982. 36pp. 8205060162. LA-9229-PR. 12999:033.

This report represents accomplishments for the period July 1 - September 30, 1981 in the areas of Trac Code Development (TRAC), Thermal-Hydraulic Research for Reactor Safety Analysis, Full-Length Emergency Core Heat Transfer--Systems Effects and Separate Effects (FLECHT-SEASET) tests, TRAC Application to 2D/3D, SIMMER Model Development and Qualification Testing, Methods for Safety Analysis of Core Disruptive Accidents, Advanced Converter Safety Research on (HTGR), TRAC Calculation Assistance and User Liaison, and the Severe Accident Sequence Analysis Program for the Division of Accident Evaluation, Office of Nuclear Regulatory Research. Presented are brief overviews compiled by project, along with a bibliography of Technical Notes and publications written during this quarter.

NUREG/CR-2283: DIRECT OBSERVATION OF MELT BEHAVIOR DURING HIGH TEMPERATURE MELT/CONCRETE INTERACTIONS. POWERS, D. A.; ARELLANO, F. E. Sandia Laboratories. April 1982. 119pp. 8204150561. SAND81-1754. 12692:212.

The feasibility of using a pulsed x-ray source and an x-ray image intensification system to provide continuous, real time data on high temperature melt behavior during interaction with concrete is demonstrated. A test of the system using a 1972g metallothermically generated melt interacting with limestone/common sand concrete is described. Analysis of the recorded x-ray image of the melt is used to determine the mode of melt contact with concrete, the time dependence of pool swelling due to entrained gas, and the nature of gas injection into the melt. Localized gas injection is found. Swelling of the pool increases with superficial gas velocity to an asymptotic limit. Results are shown to be consistent with the gas film model of melt-to-concrete heat transfer. The image data are used to assist interpretation of diagnostic data -- gas generation rate, gas composition and concrete temperatures -- gathered in the test.

NUREG/CR-2297: SECURITY MANAGEMENT TECHNIQUES AND EVALUATIVE CHECKLISTS FOR SECURITY FORCE EFFECTIVENESS. SCHURMAN, D. L.; DATESMAN, G. H.; TRUITT, J. D. Applied Science Associates, Inc. April 1982. 125pp. 8204280027. ASA NO. 635. 12877:111.

The report presents a system for evaluating and correcting deficiencies in security-force effectiveness in licensed nuclear facilities. There are four checklists which security managers can copy directly, or can use as guidelines for developing their own checklists. The checklists are keyed to corrective-action guides found in the body of the report. In addition to the corrective-action guides, the report gives background information on the nature of security systems and discussions of various special problems of the licensed nuclear industry.

NUREG/CR-2299 V04: AEROSOL RELEASE AND TRANSPORT PROGRAM. Quarterly Progress Report For October-December 1981. ADAMS, R. E.; TOBIAS, M. L. Oak Ridge National Laboratory. June 1982. 43pp. 8206090119. ORNL/TM-8307. 13442:309.

This report summarizes progress for the Aerosol Release and Transport Program sponsored by the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research, Division of Accident Evaluation, for the period October-December 1981. Topics discussed include (1) under-sodium tests in the Fuel Aerosol Simulant Test (FAST) Facility, (2) U_3O_3 and Fe_2O_3 in steam (light-water reactor accident) aerosol experiments in the Nuclear Safety Pilot Plant, (3) generation and characterization of cadmium and CdO aerosols in the basic aerosol experimental program, (4) core-melt tests of Zircaloy-clad fuel capsules, (5) initial results of a piston-model bubble oscillation code allowing liquid bypass, and (6) calculations with the UVABUBL code to compare with underwater and under-sodium period measurements in FAST experiments.

NUREG/CR-2300 V01 R1: DRAFT PRA PROCEDURE GUIDE. A Guide To The Performance Of Probabilistic Risk Assessments For Nuclear Power Plants. HICKMAN, J. W. American Nuclear Society. * Institute of Electrical & Electronic Engineers. April 1982. 200pp. 8204070131. 12593:001.

This procedures guide describes methods for performing probabilistic risk assessments (PRAs) for nuclear power plants at four levels of scope: (1) systems analysis; (2) systems and containment analysis; (3) systems, containment, and consequence analysis; and (4) full risk assessment, including external events. After reviewing its objectives and limitations, this document describes the organization and management of a PRA project and then presents procedures for accident-sequence definition and systems modeling, human-reliability analysis, the development of a data base, and the quantification of accident sequences. Procedures for evaluating the physical processes of core meltdown are presented next, followed by guidance on the evaluation of radionuclide releases from the containment as well as the analysis of environmental transport and offsite consequences. The analysis of external hazards is discussed next, including procedures for seismic, fire, and flood analyses. The guide concludes with suggestions for the development and interpretation of results and the performance of uncertainty analyses.

NUREG/CR-2301: FRACTURE MECHANICS MODELS DEVELOPED FOR PIPING RELIABILITY ASSESSMENT IN LIGHT WATER REACTORS. WOO,H.H. Los Alamos Scientific Laboratory. LIM,E.Y.; DEDHIA,D.D.; et al. Science Applications, Inc. June 1982. 237pp. 8206230340. UCRL-15490. 13595:001.

This report summarizes the work performed during fiscal year 1981 by Science Application, Inc. on the Piping Reliability Project for Lawrence Livermore National Laboratory. The efforts concentrated on modifications of the stratified Monte Carlo code called PRAISE (Piping Reliability Analysis Including Seismic Events) to make it more widely applicable to probabilistic fracture mechanics analysis of nuclear reactor piping. Pipe failures are considered to occur as the result of crack-like defects introduced during fabrication that escape detection during inspections. The code modifications allow the following factors in addition to those considered in earlier work to be treated: other materials, failure criteria and subcritical crack growth characteristic; welding residual and vibratory stresses; and longitudinal welds (the original version considered only circumferential welds). The fracture mechanics background for the code modification is included, and details of the modifications themselves provided. Additionally, an updated version of the PRAISE user's manual is included. The revised code, known as PRAISE-B was then applied to a variety of piping problems, including various size lines subject to stress corrosion cracking and vibratory stresses. Analyses including residual stresses and longitudinal welds were also performed. The results of these analyses indicate that lines subject to stress corrosion cracking (SCC) are more failure prone than ones subject to fatigue.

NUREG/CR-2306: CSRL-V: PROCESSED ENDF/B-V 227-NEUTRON-GROUP AND POINTWISE CROSS-SECTION LIBRARIES FOR CRITICALITY SAFETY, REACTOR AND SHIELDING STUDIES. FORD,W.E.; DIGGS,B.R.; PETRIE,L.M. Oak Ridge National Laboratory. June 1982. 95pp. 8207190039. ORNL/CSD/TM-160. 13919:066.

A P(3) 227-neutron-group cross-section library has been processed for the subsequent generation of problem-dependent fine- or broad-group cross sections for a broad range of applications, including shipping cask calculations, general criticality safety analyses, and reactor core and shielding analyses. The energy group structure covers the range 10(-5) eV - 20 MeV, including 79 thermal groups below 3 eV. The 129-material library includes processed data for all materials in the ENDF/B-V General Purpose File, several data sets prepared from LENDL data, hydrogen with water- and polyethelene-bound thermal kernels, deuterium with D(2)O-bound thermal kernels, carbon with a graphite thermal kernel, a special 1/V data set, and a dose factor data set. The library, which is in AMPX master format, is designated CSRL-V (Criticality Safety Reference Library based on ENDF/B-V data).

Also included in CSRL-V is a pointwise total, fission, elastic scattering, and (n,gamma) cross-section library containing data sets for all ENDF/B-V resonance materials. Data in the pointwise library were processed with the infinite dilute approximation at a temperature of 296 K.

NUREG/CR-2314: AGING WITH RESPECT TO FLAMMABILITY AND OTHER PROPERTIES IN FIRE-RETARDED ETHYLENE PROPYLENE RUBBER AND CHLOROSULFONATED POLYETHYLENE. SALAZAR,E.A.; BOUCHARD,D.A.; FURGAL,D.T. Sandia Laboratories. April 1982. 65pp. 8206100010. SAND81-1906. 13490:001.

The flammability characteristics of ethylene propylene and chlorosulfonated polyethylene rubbers containing fire-retardant additives, aged in different thermal and radiation environments have been studied. Flammability parameters for these materials (time to ignition, mass pyrolysis, burning rate and fuel consumption) when exposed to, and aged in thermal, radiation, and thermal/radiation environments are discussed. Two formulations of each type of rubber are compared. The results are a direct contradiction to expected results based on small-scale flammability tests. They show that the fire-retarding agents used in this investigation do not reduce, and in some cases, contribute to, rubber flammability when exposed to a full-scale fire environment. In addition, the results show that for full-scale fire conditions, the energy required for ignition of chlorosulfonated polyethylene is lower than that required for ethylene propylene rubber; a complete reversal of expected results. The effects of aging on the tensile-elongation properties have been determined. Radiation dose-rate effects are also discussed. Results show that the fire-retardant additives have a negligible influence on the tested materials' tensile-elongtion properties and on material aging, regardless of the aging environment. The data obtained, however, may be too limited to show significant dose-rate effects.

NUREG/CR-2317 V01 N3: CONTAINER ASSESSMENT-CORROSION STUDY OF HLW CONTAINER MATERIALS.Quarterly Progress Report,July-September 1981. AHN,T.M.; SOO,P. Brookhaven National Laboratory. May 1982. 21pp. 8206100034. BNL-NUREG-51449. 13474:320.

During this quarter work has been started on the corrosion and hydrogen embrittlement behavior of commercially pure titanium (ASTM Grade 2), TiCode-12 (ASTM Grade 12), and OFHC copper, which are primary candidate materials for high level waste containers. The test environment used is a simulated brine solution typical of bedded salt at 150 degrees C or room temperature. The immersion test results for these materials are in reasonable agreement with previous screening test results of Sandia National Laboratory; electron beam welded titanium and TiCode-12 samples show higher corrosion rates than the non-welded samples.

NUREG/CR-2317 V01 N4: CONTAINER ASSESSMENT-CORROSION STUDY OF HLW CONTAINER MATERIALS.Quarterly Progress Report,October-December 1981. AHN,T.M.; SOO,P. Brookhaven National Laboratory. June 1982. 45pp. 8207190050. BNL-NUREG-51449. 13919:309.

Efforts in this quarter have been concentrated on the uniform and crevice corrosion, and hydrogen embrittlement of TiCode-12, which are considered to be potential corrosion failure modes in high level waste container systems. The weight gain of TiCode-12 in WIPP Brine A is in good agreement with previous results from Sandia National Laboratory. The selective etching in weld heat-affected zones is considered to be responsible for the slower weight gain in the welded TiCode-12 and commercially pure (CP) titanium. The interaction of the oxide film with a salt compound precipitated from the solution makes it difficult to correlate the weight gain with the thickness of the oxide film. The crevice corrosion of TiCode-12 in neutral brine solutions at 150 degrees C has been identified by the observation of corrosion products and oxygen effects. The predominant oxide phase inside the crevice is TiO(2). In order to understand the mechanisms involved, crevice corrosion of CP titanium has also been studied. The effect of the oxidizer (produced by radiolysis) on the open circuit corrosion potential has been studied for TiCode-12 in WIPP Brine B. For a concentration of 33,000 ppm HClO(3), change in the potential has been observed, which is an indication of enhanced susceptibility to stress corrosion cracking in this material. Fractographic analysis of TiCode-12 and titanium in the study of internal hydrogen effects has been carried out using Scanning Electron Microscopy (SEM). A limited amount of theoretical work has been performed on the construction of potential-pH diagrams for copper and lead at 100 degrees C. Measurements have been made of the pressure build-up during gamma irrradiation of brine and the gases generated were analyzed.

NUREG/CR-2334: INTERPHASE TRANSPORT IN HORIZONTAL STRATIFIED CONCURRENT FLOW. JENSEN, R. J.; YUEN, M. C. Northwestern Univ. May 1982. 197pp. 8206220005. 13583:128.

The problem of interfacial transport in concurrent, horizontal stratified gas-liquid systems is considered. Local condensation heat transfer coefficients and interface shear stress were obtained from mass and force balances. Based on concurrent stratified air-water flow data, the noncondensing interface shear stress was found to be a function of the relative velocity between the phase and the liquid fraction. Incorporated into Linehan's relation for condensing flow shear stress, the correlation was found to estimate the shear velocity for the condensation data considered. Local condenstation heat transfer coefficients and gas absorption mass transfer coefficients were found to be directly proportional to the shear velocity. If the inner scales (u*) and (v/u*) are substituted into Lamont's models for the interface mass transfer coefficient, many features of the present correlation for scalar transfer are predicted. The correlations for interfacial shear stress and scalar transport can be combined to yield an iterative technique suitable for an engineering analysis of the interfacial shear, mass, and momentum transfer in a single driving force concurrent system.

NUREG/CR-2343: RISK METHODOLOGY FOR GEOLOGIC DISPOSAL OF RADIOACTIVE WASTE: THE DNET COMPUTER CODE USER'S MANUAL. CRANWELL, R. M.; CAMPBELL, J. E.; STUCKWISCH, S. E. Sandia Laboratories. April 1982. 163pp. 8205030650. SAND81-1663. 12928:246.

This report describes a network flow model (DNET) for use in simulating the process of salt dissolution in bedded salt formations. Included in the model are the capabilities for simulating processes such as salt creep, subsidence, and thermomechanical effects, all of which can effect the salt dissolution process. The model was developed for use by the Nuclear Regulatory Commission in the analysis of nuclear waste facilities in deep bedded salt formations. This document is a user's manual and is intended to facilitate the use of the DNET simulator. Mathematical equations, submodels and a description of the flow network are given. A complete input data guide is included as well as four sample problems with input deck descriptions and associated output.

NUREG/CR-2350: SENSITIVITY ANALYSIS TECHNIQUES: SELF-TEACHING CURRICULUM. IMAN, R. L. ; CONOVER, W. D. Sandia Laboratories. June 1982. 163pp. 8207080061. SAND81-1978. 13795:102.

This report contains discussions and exercises that illustrate the application of the sensitivity analysis techniques developed at Sandia National Laboratories for the Risk Methodology for Geologic Disposal of Radioactive Waste Project. With this report the user may familiarize himself with the application of the Latin Hypercube Sampling (LHS) program and the Stepwise Regression (STEP) program with the groundwater transport model NWFT/DVM to do sensitivity and uncertainty analyses. The user may require the User's Guides for LHS (SAND 79-1473), STEP (SAND 79-1472), and NWFT/DVM (NUREG/CR-2081) to make full use of this self-teaching curriculum. This report is one of a series of self-teaching curricula prepared under a technology transfer contract for the U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards.

NUREG/CR-2353 V02: SPECIFICATION AND VERIFICATION OF NUCLEAR POWER PLANT TRAINING SIMULATOR RESPONSE CHARACTERISTICS. Part II: Conclusions And Recommendations. HAAS, P. M. ; KERLIN, T. W. ; SELBY, D. L. ; et al. Oak Ridge National Laboratory. May 1982. 76pp. 8206090122. ORNL/TM-7985/P2. 13456:185.

This report is the second volume of a two-volume report summarizing the findings, conclusions, and recommendations of a survey study for the U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research: (1) to gather information on standards and practices of the nuclear industry and certain non-nuclear industries which might be used to specify and verify the response characteristics of nuclear power plant training simulators; (2) to compare findings from the nuclear and non-nuclear industries in order to identify which of the non-nuclear practices might be profitably adopted by the nuclear industry; and (3) to recommend actions that could be pursued, by NRC or by the nuclear industry, to improve standards and practices. The first volume of the report summarized the information gathered from the nuclear and non-nuclear industries. This report presents the conclusions and recommendations from the survey study. A general conclusion is: The nuclear industry should adopt and NRC regulatory and research actions should support the systems approach to training as a structured framework for development and validation of personnel training systems.

NUREG/CR-2356: UPDATED INPUT FOR THE WRAP-EM SYSTEM. REED, R. L. ; GREGORY, M. V. Savannah River Laboratory. April 1982. 180pp. 8204290497. DPST-80-6. 12895:350.

The Water Reactor Analysis Package (WRAP) provides the capability to analyze loss-of-coolant accidents (LOCAs) in both pressurized water reactors (PWRs) and boiling water reactors (BWRs) by using evaluation models (EMs). The specifications for modules in the WRAP-EM system have been presented in previous documents. This document presents revised and updated input specifictions for the WRAP-EM modules.

NUREG/CR-2359: ATMOSPHERIC STRUCTURE PRIOR TO TORNADOES AS DERIVED FROM PROXIMITY AND PRECEDENT UPPER AIR SOUNDINGS. TAYLOR, G. E. ; DARKOW, G. L. Missouri, Univ. of, Columbia. May 1982. 106pp. 8206110325. 13490:087.

The uniqueness of the thermodynamic and dynamic structure of the atmosphere in the area of imminent tornado-bearing storm development is analyzed by comparing 115 tornado proximity soundings with upper air soundings made at the same location 6 and 12 hours earlier (precedent soundings) and with soundings made simultaneously at neighboring upper air stations. The comparisons suggest that both the proximity station and the neighboring station upstream with respect to the mean flow in the low level moist air display very similar degrees of hydrostatic and potential-convective instability by late afternoon. The principal difference is in the wind profiles at the two locations. The tornado proximity station displays significantly stronger wind speeds above 1 km with the most striking difference being in the vertical shear of the wind in the layer from 1 to 3 km above ground level. In this layer the winds at the proximity station show an average increase of about 3 m/sec while the upstream, nontornadic station shows a slight decrease of wind speed with height.

NUREG/CR-2362: RELATIONSHIPS BETWEEN CHARPY V-NOTCH IMPACT ENERGY AND FRACTURE TOUGHNESS. DOUGAN, J. R. Oak Ridge National Laboratory. April 1982. 103pp. 8204290619. ORNL/TM-7921. 12899:138.

The Fracture and Irradiation Effects Program has been concerned with the development of a better technical basis for the preparation of regulatory guides regarding the prevention of fracture or excessive deformation under the expected environmental conditions in the pressure boundaries of light-water reactors. One program objective has been the development of toughness estimates for fracture analysis in the upper-shelf temperature range for beltline materials that have low upper-shelf Charpy energy values due to radiation damage. This report documents the investigation of correlations between Charpy-V-notch impact energy and fracture toughness.

NUREG/CR-2366 V02: MULTIROD BURST TEST PROGRAM PROGRESS REPORT FOR JULY-DECEMBER 1981. CROWLEY, J. L. Oak Ridge National Laboratory. April 1982. 75pp. 8205110109. ORNL/TM-8190. 13038:003.

The assembly of B-6, the final bundle of the Multirod Burst Test Program, was completed. The bundle is now being connected to power and instrumentation for the burst test in January 1982. The test condition heat rate and burst temperature have been revised slightly (5 K/s and 925 degrees C, respectively), but the purpose remains that of determining the large bundle effect on deformation in the a-B transition region.

Posttest examination of B-5, including a water flow test, epoxy casting, sectioning, and measurement of sections, has been completed. Examination of the data obtained continues, and some preliminary observations are included in this report.

A special series of five single-rod burst tests was conducted at essentially identical conditions for the purpose of investigating statistical variations in deformation. The variation of the sample standard deviation about the mean value for the burst-to-original circumference ratio was about 11%.

NUREG/CR-2377: TESTS & CRITERIA FOR FIRE PROTECTION OF CABLE PENETRATIONS. WILLIAMSON, R. B.; FISHER, F. L. Sandia Laboratories. April 1982. 107pp. 8205060133. SAND81-7160. 13001:001.

A series of experiments are described which evaluate the effects of test furnace pressure differential and excess pyrolyzates on the fire resistance of cable penetrations installed in fire resistive walls. It is shown that the measured fire resistance of pentrations can be strongly influenced by the pressure difference between the test furnace and the unexposed face of the penetration, and, to a lesser degree, by the presence or absence of excess pyrolyzates. Methods for the local introduction of excess pyrolyzates into fire test furnace are discussed.

NUREG/CR-2381: GEOLOGIC AND HYDROLOGIC RESEARCH AT THE WESTERN NEW YORK NUCLEAR SERVICE CENTER, WEST VALLEY, NEW YORK. Progress Report, August 1979-July 1981. ALBANESE, J. R.; DUNN, L. A.; ROGERS, W. B.; et al. New York, State of. May 1982. 113pp. 8205200277. 13197:238.

This is a report of the progress made during the first part of a proposed multi-year program of geologic and hydrologic investigations at the Western New York Nuclear Service Center. The New York State Geological Survey previously worked (1975-1979) on a small part of this area, specifically that of the New York State-licensed radioactive waste burial trenches. During the latest reporting period a large scale topographic map of the 140 hectare site immediately surrounding the nuclear fuel reprocessing plant has been produced, and three additional permanent stream stations have been installed to allow monitoring of most runoff from the site. Ten holes drilled in the North Plateau determined the geometry of the surfical gravel deposits there. A system of groundwater monitoring wells was established in these holes. The second phase of the geomorphic investigations of the Buttermilk Creek drainage basin and a study of the effect of submergence on the geotechnical properties of the burial till were completed.

NUREG/CR-2387: CREDIBLE ACCIDENT ANALYSES FOR TRIGA AND TRIGA-FUELED REACTORS. HAWLEY, S. C.; KATHREN, R. L. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1982. 63pp. 8205110094. PNL-4028. 13038:295.

Credible accidents were developed and analyzed for TRIGA and TRIGA-fueled reactors. The only potential for offsite exposure appears to be from a fuel-handling accident that, based on highly conservative assumptions, would result in dose equivalents of less than or equal to 1 mrem to the total body from noble gases and less than or equal to 1.2 rem to the thyroid from radioiodines. Credible accidents from excess reactivity insertions, metal-water reactions, lost, misplaced, or inadvertent experiments, core rearrangements, and changes in fuel morphology and ZrHx composition are also evaluated, and suggestions for further study provided.

NUREG/CR-2392: SUMMARY OF ORNL WORK ON NRC-SPONSORED HTGR SAFETY RESEARCH, JULY 1974-SEPTEMBER 1980. CLEVELAND, J. C.; CONKLIN, J. C.; HARRINGTON, R. M.; et al. Oak Ridge National Laboratory. April 1982. 58pp. 8205130231. ORNL/TM-8073. 13090:106.

A summary is presented of the major accomplishments of the research program on High-Temperature Gas-Cooled Reactor (HTGR) safety. This report is intended to help the Nuclear Regulatory Commission establish goals for future research by comparing the status of the work here (as well as at other laboratories) with the perceived safety needs of the large HTGR. The program includes extensive work on dynamics-related safety code development, use of codes for studying postulated accident sequences, and use of experimental data for code verification. Cooperative efforts with other programs are also described. Suggestions for near-term and long-term research are presented.

NUREG/CR-2393: FUEL AEROSOL SIMULANT TEST DATA RECORD REPORT: UNDERWATER TESTS. SMITH, A. M. ; WRIGHT, A. L. ; ROCHELLE, J. M. ; et al. Oak Ridge National Laboratory. April 1982. 77pp. 8205130249. ORNL/TM-8085. 13089:263.

This data record summarizes 34 uranium dioxide (UO(2)) vaporization experiments performed under water in the Fuel Aerosol Simulant Test (FAST) project. The FAST project is part of the Oak Ridge National Laboratory Aerosol Release and Transport Program sponsored by the Division of Accident Evaluation of the Nuclear Regulatory Commission. The underwater tests were performed as a prelude to under-sodium tests and were done to permit characterization of the behavior of UO(2) vapor bubbles for various test conditions. Included in the report are descriptions of test procedures along with tables and graphs summarizing the results.

NUREG/CR-2393 ERR: Errata, changing rept number to NUREG/CR-2593, to A USER'S MANUAL FOR COMPUTER CODE RIBD/IRT. THAYER, D. D. ; LURIE, N. A. Sandia Laboratories. April 22, 1982. 1p. 8205200254. SAND82-7013. 13202:355.

The computer code RIBD/IRT is a modified version of RIBD-II. It is a grid processor that calculates isotopic concentrations resulting from two fission sources with normal down-chain decay by beta emission and isomeric transfers and inter-chain coupling resulting from n-gamma reactions. Calculations can be made to follow an irradiation history through an unlimited number of step changes of unrestricted duration and variability including shutdown periods, restarts at different power levels and/or any other level changes. Output information includes time-dependent inventories, activities, decay powers, and energy releases for as many as 800 fission products. Modifications to RIBD-II were necessitated by Loss-of-Coolant Accident (LOCA) studies conducted by IRT Corporation regarding fission product source term definition. These modifications permit the user to track and modify the concentrations of individual elements as they decay with time following reactor shutdown. In essence, one can determine time-dependent fission product source terms resulting from any reactor operating history which then can be used as input into fission product transport codes. Other modifications to RIBD-II expanded the output information to assist the user in analyzing the source term. This manual describes the modifications to RIBD/II, input requirements and a sample problem. The appendicies give a listing of RIBD/IRT, sample output, and a listing of a code called ZIP which prepares the library tape for input to RIBD/IRT. The code is available in a UNIVAC 1100/81 version and a VAX 11/780 version.

NUREG/CR-2394 ERR: Errata, changing rept number to NUREG/CR-2594, to A USER'S MANUAL FOR THE GABAS SPECTRUM COMPUTER CODE. THAYER, D. D. ; LURIE, N. A. Sandia Laboratories. April 20, 1982. 1p. 8205200265. SAND82-7014. 13202:356.

The Gamma and Beta Spectrum computer code (GABAS) was developed at IRT Corporation for calculating time-dependent beta and/or gamma spectra from decaying fission products. GABAS calculates composite fission product spectra based on the technique used by England, et al., in conjunction with the CINDER family of fission product codes.

Multigroup beta and gamma spectra for individual nuclides are folded with their corresponding time-dependent activities (usually generated by a fission product inventory code) to produce a composite time-dependent fission product spectrum. This manual contains the methodology employed by GABAS, input requirements for proper execution, a sample problem and a FORTRAN listing compatible with a UNIVAC machine.

The code is available in a UNIVAC 1100/81 version and a VAX 11/780 version. The former may be obtained from the Radiation Shielding Information Center (RSIC); the latter may be obtained directly from IRT Corporation.

NUREG/CR-2403 S01: SURVEY OF INSULATION USED IN NUCLEAR POWER PLANTS AND THE POTENTIAL FOR DEBRIS GENERATION. KOLBE,R.; GAHAN,E. Burns & Roe Co. May 1982. 100pp. 8206110004. SAND82-0927. 13492:273.

In support of Unresolved Safety Issue, USI A-43, "Containment Emergency Sump Performance," 8 additional nuclear power plants (representative of different U.S. reactors' manufacturers and architect-engineers) were surveyed to identify and document the types and amounts of insulation used, location within containment, components insulated, material characteristics, and methods of installation and attachment. The plants were selected to obtain survey information on "older" plants and supplements survey information previously reported in NUREG/CR-2403. In addition, a preliminary assessment was made of the potential for migration of the insulation debris which might be generated as a result of the postulated loss-of-coolant accident (pipe break).

NUREG/CR-2412: HEAT REMOVAL FROM A STRATIFIED UO2-SODIUM PARTICLE BED. MITCHELL,G.W.; LIPINSKI,R.J.; SCHWARZ,M.L. Sandia Laboratories. May 1982. 120pp. 8205180110. SAND81-1622. 13136:045.

The D6 Debris Bed Experiment is one in a series of Post Accident Heat Removal (PAHR) Experiments being conducted to investigate the coolability of debris beds which might exist as a result of a severe nuclear reactor accident. The D6 experiment is the first in the series to investigate the effects of particle size stratification, which would likely exist for many accident scenarios, on debris bed coolability. The D6 debris bed contained 4.87 kg of UO(2) particulate, which formed a bed 114 mm high and 102 mm in diameter. At low power, heat removal could be described to the conduction equation, with effective bed conductivity in agreement with the Kampf-Karsten relation to within ten percent. Single phase convection was not observed in the bed. The power required to achieve dryout ranged from 0.28 to 0.45 W/g for overlying bulk sodium temperatures. These powers are significantly below that which would be predicted by current models. Based on evaluation of the data, it appears that stratification suppresses convection, reduces the power required to achieve dryout, and suppresses the formation of vapor channels which would result in increased coolability.

NUREG/CR-2413: SURVEY OF REMOTE AREA MONITORING SYSTEMS AT U.S. LIGHT-WATER-COOLED POWER REACTORS. KATHREN,R.L. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1982. 46pp. 8204280021. PNL-4106. 12876:285.

A study was made of the capabilities and operating practices, including calibration, of remote area monitoring (RAM) systems at light-water-cooled power reactors in the United States. The information was obtained by mail questionnaire. Specific design capabilities, including range, readout and alarm features are documented along with the numbers and location of detectors, calibration and operational procedures. Comments of respondents regarding RAB systems are also included.

NUREG/CR-2416: INITIAL QUANTIFICATION OF HUMAN ERROR ASSOCIATED WITH SPECIFIC INSTRUMENTATION AND CONTROL SYSTEM COMPONENTS IN LICENSED NUCLEAR POWER PLANTS. LUCKAS, W. J. ; LETTIERI, V. ; HALL, R. E. Brookhaven National Laboratory. May 1982. 20pp. 8206100038. BNL-NUREG-51480. 13475:030.

This report provides a methodology for the initial quantification of specific categories of human errors made in conjunction with several instrumentation and control system components operated, maintained, and tested in licensed nuclear power plants. The resultant human error rates (HER) provide the first real systems bases of comparison for the existing derived and/or best judgement equivalent set of such rates or probabilities. These calculated error rates also provide the first real indication of human performance as it related directly to specific tasks in nuclear plants. This work of developing specific HERs is both an extension of and an outgrowth of the generic HERs developed for safety system pumps and valves as reported in NUREG/CR-1880.

NUREG/CR-2417: IDENTIFICATION AND ANALYSIS OF HUMAN ERRORS UNDERLYING PUMP AND VALVE RELATED EVENTS REPORTED BY NUCLEAR POWER PLANT LICENSEES. SPEAKER, D. M. ; THOMPSON, S. R. ; LUCKAS, W. J. Brookhaven National Laboratory. May 1982. 31pp. 8206100063. BNL-NUREG-51481. 13475:355.

This report provides a useful and adaptable data base of human error associated with the operation, testing, and maintenance of reactor safety system pumps and valves in licensed nuclear power plants. To produce this data base, a practical and workable methodology was developed and implemented on more than 3,000 Licensee Event Reports (LERs) which resulted in a human error data base six times larger than indicated by the LERs themselves. This data base is intended to provide a realistic assessment of the appropriate human error populations required in NUREG/CR-1880.

NUREG/CR-2431: BURN MODE ANALYSIS OF HORIZONTAL CABLE TRAY FIRES. SCHMIDT, W. H. Sandia Laboratories. April 1982. 55pp. 8204290610. SAND81-0079. 12895:140.

Electrical cable fire tests have been conducted at the Sandia Fire Research Facility in Albuquerque, New Mexico, in order to evaluate cable tray fire safety criteria for the Nuclear Regulatory Commission. A burn mode concept was developed in order to describe and classify the thermodynamic phenomena which occur in the presence of smoke and to compare the fire growth and recession of different cable types under otherwise unchanged fire test conditions. The importance of deep seated fires in cable trays from the standpoint of propagation, detection, and suppression is emphasized. The cable tray fire tests demonstrate that fire recession and deep seated fires can result from a descending smoke layer and that reignition and secondary fire growth is possible by readmission of fresh air.

NUREG/CR-2432: A UNIQUE CONCEPT FOR LIQUID LEVEL AND VOID FRACTION DETECTION IN SEVERE FUEL DAMAGE TESTS. TOKARZ, R. D. ; CROWELL, S. L. ; PANISKO, F. E. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1982. 47pp. 8205200284. PNL-4070. 13200:263.

This report describes a direct-contacting liquid level and void fraction detection system that is being developed by Pacific Northwest Laboratory. The measurement technique could be used in the severe fuel damage tests that will be conducted at the Power Burst Facility, Idaho Falls, Idaho, and at the ESSOR reactor, Ispra, Italy. The detection system could also be retrofitted for commercial operating reactors to provide definitive thermal-hydraulic information. The technique can provide unambiguous, real-time data on liquid level and void fraction during normal reactor operation as well as during shutdown and accident conditions.

NUREG/CR-2434: FRAC (FAILURE RATE ANALYSIS CODE): A COMPUTER PROGRAM FOR ANALYSIS OF VARIANCE OF FAILURE RATES. An Application User's Guide. MARTZ, H. F. ; BECKMAN, R. J. ; MCINTEER, C. R. Los Alamos Scientific Laboratory. May 1982. 52pp. 8205180018. LA-9116-MS. 13135:238.

Probabilistic risk assessments (PRAs) require estimates of the failure rates of various components whose failure modes appear in the event and fault trees used to quantify accident sequences. Several reliability data bases have been designed for use in providing the necessary reliability data to be used in constructing these estimates. In the nuclear industry, the Nuclear Plant Reliability Data System (NPRDS) and the In-Plant Reliability Data System (IPRDS), among others, were designed for this purpose.

An important characteristic of such data bases is the selection and identification of numerous factors used to classify each component that is reported and the subsequent failures of each component. However, the presence of such factors often complicates the analysis of reliability data in the sense that it is inappropriate to group (that is, pool) data for those combinations of factors that yield significantly different failure rate values. These types of data can be analyzed by analysis of variance. Analysis of variance is a statistical data analysis methodology for use in addressing such questions as: How do the factors affect the failure rate? What are the estimated effects due to these factors? Which factor combinations yield the largest failure rate estimates? Are there factor interactions that significantly affect the failure rate? FRAC (Failure Rate Analysis Code) is a computer code that performs an analysis of variance of failure rates and provides information for estimates answering the above questions. In addition, FRAC provides failure rate estimates.

NUREG/CR-2435: DISPERSED FLOW FILM BOILING IN ROD BUNDLE GEOMETRY-STEADY STATE HEAT TRANSFER DATA AND CORRELATION COMPARISONS. YODER, G. L. ; OTT, L. J. ; MORRIS, D. G. ; et al. Oak Ridge National Laboratory. April 1982. 337pp. 8205130274. ORNL-5822. 13087:126.

Assessment of six film boiling correlations and one single-phase vapor correlation has been made using data from 22 steady state upflow rod bundle tests (series 3.07.9): 1. Dougall-Rosenow, 2. Dougall-Rosenow (wall Prandtl number), 3. Groeneveld 5.7, 4. Groenevend 5.9, 5. Condie-Bengston IV, 6. Groeneveld-Delorme, and 7. Dittus-Boelter. Bundle fluid conditions were calculated using energy and mass conservation considerations. Results of the steady state film boiling tests support the conclusions reached in the analysis of prior transient tests 3.03.6AR, 3.06AR, 3.06.6B, and 3.08.6C. Comparisons between experimentally determined and correlation-predicted heat transfer coefficients indicate that the Dougall-Rohsenow correlation often overpredicts the heat transfer coefficient, while the Groeneveld 5.7, Groeneveld 5.9, and Condie-Bengston IV correlations tend to be in better agreement with the data. The Groenevend-Delorme correlation underpredicts heat fluxes near dryout but improves as distance from dryout increases. The Dittus-Boelter correlation, which tends to overpredict the heat transfer coefficient, was evaluated only when equilibrium qualities were greater than 1.

related to the total core volumetric vapor generation rate. Assessment of commonly used local void-fraction models indicated that of the correlations examined, the Yeh void correlation was best suited for use under the subject test conditions.

NUREG/CR-2460: TECHNICAL SUPPORT FOR IMPROVING THE LICENSING REGULATORY BASE FOR SELECTED FACILITIES ASSOCIATED WITH THE FRONT END OF THE FUEL CYCLE. CLARK, R. G.; SCHRIEBER, R. E.; JAMISON, J. D.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1982. 112pp. 8204280011. PNL-4086. 12877:240.

Pacific Northwest Laboratory (PNL) has reviewed the health, safety and environmental regulatory base to assess its adequacy as a guide to applicants for licenses to operate UF(6) conversion facilities and fuel fabrication plants. The regulatory base was defined as the body of documented requirements and guidance to licensees, including laws passed by Congress, Federal Regulations developed by the NRC to implement the laws, license conditions added to each license to deal with special requirements for that specific license, and Regulatory Guides.

The study concentrated on the renewal licensing accomplished in the last few years at five typical facilities, and included analyses of licensing documents and interviews with individuals involved with different aspects of the licensing process. Those interviewed included the NMSS staff, Inspection and Enforcement (IE) officials, and selected licensees. From the results of the analyses and interviews, the PNL study team concludes that the regulatory base is adequate but should be codified for greater visibility. PNL recommends that NMSS clarify distinctions among legal requirements of the licensee, acceptance criteria employed by NMSS, and guidance used by all. In particular, a prelicensing conference among NMSS, IE and each licensee would be a practical means of setting license conditions acceptable to all parties.

NUREG/CR-2464: METHODS FOR CLASSIFYING MIXTURES OF EXPONENTIAL DISTRIBUTIONS BASED ON EITHER EXPONENTIAL OR POISSON DATA. BECKMAN, R. J.; MARTZ, H. F.; HARPER, M. D.; et al. Los Alamos Scientific Laboratory. April 1982. 40pp. 8205130261. LA-9133-MS. 13088:326.

In conducting probabilistic risk analyses of nuclear power plants a suitable data base must be developed for use in estimating component unavailabilities which are required in quantification of accident sequences. Often data exists on either the time to failure of certain components or the number of component failures in a total operating or test time. Frequently there is not a single underlying failure rate (lambda) for all of these data and the data represent a mixture of different populations. Techniques are developed in this manuscript which allow the analyst to classify data as coming from populations with failure rates that either do or do not differ by a specified amount such as an order of magnitude. It is assumed that the failure data either follow an exponential (time to failure observed) or a Poisson (number of failures observed) distribution and that the true failure rate is itself a random variable with a specified prior distribution. Several different prior distributions are considered in examining the performance of the methods. For both types of data, three classification schemes are presented. The first is a classical scheme which ignores the prior data. In the second scheme, data are classified according to their maximum posterior probability, and the last method involves the minimization of an expected loss function.

NUREG/CR-2442: RELIABILITY ANALYSIS OF STEEL CONTAINMENT STRENGTH. Technical Report, August 1980-September 1981. GREIMANN, L.; FANOUS, F.; WOLD-TINSAE, A.; et al. Iowa State Univ. June 1982. 201pp. 8207190033. 13921:019.

A best estimate and uncertainty assessment of the resistance of the St. Lucie, Cherokee, Perry, WPPSS and Browns Ferry containment vessels was performed. The Monte Carlo simulation technique and second moment approach were compared as a means of calculating the probability distribution of the containment resistance. A uniform static internal pressure was used and strain ductility was taken as the failure criterion. Approximate methods were developed and calibrated with finite element analysis. Both approximate and finite element analyses were performed on the axi-symmetric containment structure. An uncertainty assessment of the containment strength was then performed by the second moment reliability method. Based upon the approximate methods, the cumulative distribution for the resistance of five containments (shell modes only) is presented in the text.

NUREG/CR-2455: EXPERIMENTAL INVESTIGATIONS OF BUNDLE BOILOFF AND REFLOOD UNDER HIGH-PRESSURE LOW HEAT FLUX CONDITIONS. HYMAN, C. R.; ANKLAM, T. M.; WHITE, M. D. Oak Ridge National Laboratory. April 1982. 131pp. 8205110657. ORNL-5846. 13036:038.

Results are reported from high-pressure bundle boiloff and reflood tests run during the second series of pressurized-water reactor small-break loss-of-coolant accident (SBLOCA) heat transfer experiments. Tests were conducted at Oak Ridge National Laboratory in the Thermal Hydraulic Test Facility (THTF), a 64-rod full-length rod bundle heat transfer loop. Tests discussed include five bundle boiloff tests and five reflood tests. Tests were performed under conditions similar to those expected in an SBLOCA.

NUREG/CR-2456: EXPERIMENTAL INVESTIGATIONS OF UNCOVERED-BUNDLE HEAT TRANSFER AND TWO-PHASE MIXTURE-LEVEL SWELL UNDER HIGH-PRESSURE LOW HEAT-FLUX CONDITIONS. ANKLAM, T. M.; MILLER, R. J.; WHITE, M. D. Oak Ridge National Laboratory. April 1982. 311pp. 8205130229. ORNL-5848. 13076:045.

Results are reported from a series of uncovered-bundle heat transfer and mixture-level swell tests. Experimental testing was performed in the Thermal Hydraulic Test Facility (THTF). The THTF is an electrically heated bundle test loop configured to produce conditions similar to those in a small-break loss-of-coolant accident. The objective of heat transfer testing was to acquire heat transfer coefficients and fluid conditions in a partially uncovered bundle. Testing was performed in a quasi-steady-state mode with the heated core 30 to 40% uncovered. Linear heat rates varied from 0.32 to 2.22 kW/m.rod (0.1 to 0.68 kW/ft.rod). Under these conditions peak clad temperatures in excess of 1050 K (1430 degrees F) were observed, and total heat transfer coefficients ranged from 0.0045 to 0.037 W/cm(2).K (8 to 65 Btu/h.ft(2).degrees F). Spacer grids were observed to enhance heat transfer at, and downstream of the grid. Radiation heat transfer was calculated to account for as much as 65% of total transfer in low-flow tests. It is recommended that a reference temperature correlation, based on the modified wall Reynolds number, be used to predict convective heat transfer in the range 2000 less than or equal to Re(mw) less than or equal to 10,000. Results of mixture-level swell testing showed that the relative expansion of the boiling length caused by the presence of vapor voids (mixture-level swell) was linearly

NUREG/CR-2469: AN ANALYSIS OF TRANSIENT FILM BOILING OF HIGH-PRESSURE WATER IN A ROD BUNDLE. MORRIS, D. G.; MULLINS, C. B.; YODER, G. L. Oak Ridge National Laboratory. April 1982. 255pp. 8205120110. ORNL/NUREG-85. 13053:003.

The following six dispersed-flow film-boiling correlations were assessed using data from three ORNL transient film-boiling experiments conducted in the THTF: 1. Dougall-Rohsenow, 2. Dougal-Rohsenow (with Prandtl number evaluated at the wall temperature, as used in RELAP4-MOD7), 3. Groeneveld 5.9, 4. Groeneveld 5.7, 5. Groeneveld-Delorme, and 6. Condie-Bengston IV. The correlations were evaluated with the bundle fluid conditions calculated using a homogeneous two-phase flow and thermodynamic equilibrium thermal-hydraulics code. Comparisons made between experimentally determined heat transfer coefficients and the individual correlations' heat transfer coefficients indicate that (1) the Dougall-Rohsenow correlations often overpredict the heat transfer coefficients; (2) the Groeneveld 5.7, Groeneveld 5.9, and Condie-Bengston IV correlations tend to be in general agreement with the data; and (3) the Groeneveld-Delorme correlation underpredicts the data. Equilibrium bundle fluid conditions are reported along with fuel rod simulator surface temperature and heat flux. Calculated experimental heat transfer coefficients are also reported. Uncertainties are reported for calculated heat transfer parameters for one of the transient tests. Thermodynamic nonequilibrium in the three transient tests was examined with an advanced two-fluid thermal-hydraulics code.

NUREG/CR-2470: THERMOMETRY IN THE MULTIROD BURST TEST PROGRAM. ANDERSON, R. L.; CARR, K. R.; KOLLIE, T. G. Oak Ridge National Laboratory. June 1982. 91pp. 8206230031. ORNL/TM-8024. 13593:146.

An important objective of the MRBT program is to improve the understanding of the behavior of the Zircaloy cladding of nuclear fuel rods under conditions postulated for large-break, loss-of-coolant accidents. A temperature measurement error analysis was performed for the Type S (0.25-mm-diam, bare-wire) and Type K (0.71-mm-diam, sheathed) thermocouple circuits used to measure the temperature of the Zircaloy-clad, electrically heated fuel-rod simulators in the Multirod Burst Test program (MRBT). The analysis produced the following estimates for the total maximum errors in the range 300 to 1000 degrees C: Type K thermocouples (worst case of two test facilities) exclusive of thermal shunting error, which remains to be estimated by mathematical modeling: +12.50 degrees C in addition to -1.7 degrees C due to thermocouple cold work. Type S thermocouples: +10.6 degrees C in addition to -1.4 degrees C due to thermocouple cold work. Eight categories of error sources were studied both analytically and experimentally: thermal shunting; electrical shunting and leakage; thermocouple decalibration in service; thermocouple properties of thermocouple extension wear, plugs, and jacks; thermocouple reference junction; data acquisition system; and electrical noise.

NUREG/CR-2473: SIMMER ANALYSIS OF PROMPT BURST ENERGETICS EXPERIMENTS. HITCHCOCK, J. T. Sandia Laboratories. June 1982. 48pp. 8206250010. SAND81-0933. 13620:307.

The Prompt Burst Energetics experiments are designed to measure the pressure behavior of fuel and coolant as working fluids during a hypothetical prompt burst disassembly in an LMFBR. The work presented in this report consists of a parametric study of PBE-5S, a fresh oxide fuel experiment, using SIMMER-II. The various pressure sources in the experiment are examined, and the dominant source identified as incondensible contaminant gasses in the fuel. The important modeling uncertainties and limitations of SIMMER-II as applied to these experiments are discussed.

NUREG/CR-2481: LIGHT WATER REACTOR SAFETY RESEARCH PROGRAM Semiannual Report, April-September 1981. BERMAN, M. Sandia Laboratories. April 1982. 300pp. 8204300030. SAND82-0006. 12919:009.

This report covers progress in 5 main programs during April-September 1981.

1. The Molten Fuel-Concrete Interactions (MFCI) study presently consists of analytical investigations of the chemical and physical phenomena associated with interactions between molten core materials and concrete. Such interactions are possible during hypothetical fuel melt accidents in light water reactors (LWRs).

2. The two main purposes of the steam explosion phenomena program are: (a) to identify experimentally the magnitudes and time characteristics of pressure pulses and other initial conditions necessary to trigger and propagate explosive interactions between water and molten light water reactor (LWR) materials; and (b) to assess the probability and consequences of steam explosions during postulated meltdown accidents in LWRs.

3. The Containment Emergency Sump Performance (CESP) investigates the reliability of ECCS sumps and has two main purposes: (a) to provide a containment-sump data base to NRC, and (b) to provide ECCS sump design information to the nuclear industry.

4. The goals of the Hydrogen Program are to quantify the threat posed by hydrogen released during LWR accidents and to generate information and equipment concepts which will prevent or mitigate that threat.

5. The combustible Gas in Containment Program determines the quantity and rate of generation of hydrogen from the corrosion of zinc (in galvanized steel and in zinc-bearing paints) located within light water reactor containment buildings.

NUREG/CR-2493: AQUEOUS IODINE CHEMISTRY IN LWR ACCIDENTS: Review And Assessment. BELL, J. T.; PALMER, R. A.; CAMPBELL, D. A.; et al. Oak Ridge National Laboratory. May 1982. 77pp. 8206100007. ORNL-5824. 13474:029.

Radioactive iodine is among the most significant fission products with respect to potential environmental insult in the event of a serious nuclear reactor accident. The potential environmental insult will depend on the chemical forms and radioactivities of the iodine inside the reactor containment. Few publications report studies of aqueous iodine chemistry at conditions pertinent to a reactor accident. This report assesses that chemistry under accident conditions, but based on results from studies of systems under convenient experimental conditions. Several items of interest to iodine chemistry are summarized: (1) redox reactions of iodine species, (2) hydrolysis and disproportionation reactions, (3) formation and reactions of organic iodide, (4) radiation effects on aqueous iodine species, (5) liquid-gas phase partitioning, and (6) computer program (IQU) to calculate equilibrium concentrations of ten iodine species.

NUREG/CR-2494: OR-FLAW: A FINITE ELEMENT PROGRAM FOR DIRECT EVALUATION OF K-FACTORS FOR USER-DEFINED FLAWS IN PLATES, CYLINDERS AND PRESSURE-VESSEL NOZZLE CORNERS. ATLURI, S. N.; BRYSON, J. W.; BASS, B. R.; et al. Oak Ridge National Laboratory. May 1982. 82pp. 8206100022.

ORNL/CSD/TM-165. 13473:309.

This report describes the linear elastic finite element computer program OR-FLAW (Oak Ridge-FLAW). The program directly calculates the mixed-mode stress intensity factors (K(I),K(II), and K(III)) along user-defined flaws in plates, cylinders, and pressure-vessel nozzle corners. Special three-dimensional crack front elements are used to model the immediate vicinity of the flaw. These crack front elements have the proper square root and inverse square root variations for displacement and stresses, respectively. Regular isoparametric elements are used away from the flaw front. Interelement displacement compatibility between singular and regular elements is satisfied by assuming an independent boundary displacement field (hybrid-displacement procedures) and using a Lagrange multiplier technique to enforce the compatibility constraint. The stress intensity factors at various points on the crack front are solved directly along with the unknown nodal displacements. The program provides for automatic generation of a finite element model incorporating either a mathematical or user-defined flaw. Generation and analysis of the model are performed with program input consisting of 8 to 12 cards. Applications of the program to a surface flaw in a flat plate and to a symmetrical corner crack in a plate-hole configuration are described.

NUREG/CR-2495: CHARACTERIZATION OF SOIL TO PLANT TRANSFER COEFFICIENTS FOR STABLE CESIUM AND STRONTIUM. HOFFMAN, L. G.; KELLER, J. H. Exxon Nuclear Co., Inc. (subs. of Exxon Corp.). June 1982. 53pp. 8206230352. 13595:239.

Soil and vegetation samples were collected from seven counties in the United States in which commercial nuclear power reactors were sited. Samples were analyzed for stable cesium and strontium by atomic emission spectrometry. In addition, soils were analyzed for major elements content, organic content, pH and ion exchange capacity using standard soil analytical methods. Soils were classified using U.S. Department of Agriculture (USDA) soil survey maps. Soil to plant transfer coefficients were calculated for dry vegetation and dry soil and for fresh vegetation and dry soil. The observed transfer coefficient values are higher than those reported in the U.S. Nuclear Regulatory Commission's (USNRC) Regulatory Guide 1.109 for both cesium and strontium. The coefficients vary by a factor of 100 for cesium and by 1000 for strontium for corn. Low cesium concentrations in both the vegetation and soil resulted in some ambiguity in the transfer coefficients in some samples. The soil extraction method used, a mineral acid leach, may result in transfer coefficients higher than those which would have resulted if a total dissolution technique had been used. The limited number of samples collected at any site precluded any statistical treatment of the data.

NUREG/CR-2496: HUMAN ENGINEERING DESIGN CONSIDERATIONS FOR CATHODE RAY TUBE-GENERATED DISPLAYS. BANKS, W. W.; GERTMAN, D. I.; PETERSEN, R. J. EG&G, Inc. May 1982. 125pp. 8206100059. 13470:093.

The preliminary findings are that research is needed in the following areas of (CRT)-generated displays in order to anchor regulatory guidelines and regulations to firm empirical data: a. Image Distortion; b. Display Format; c. Work Surface Light Reflection; d. Cognitive Fidelity; e. Response Time; and f. Phosphor.

NUREG/CR-2497 V01: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1969-1979. A Status Report. Vol. 1. Main Report And App. A, C, D And E. MINARICK, J. W. ; KUKIELKA, C. A. Science Applications, Inc. June 1982. 350pp. 8207220676. ORNL/NSIC-182. 14019:227.

Descriptions of 169 operational events reported as Licensee Event Reports, which occurred at commercial light-water reactors during 1969-1979 and which are considered to be precursors to potential severe core damage, are presented, along with associated event trees and categorizations and subsequent analyses. This report summarizes work in (1) the development of methods used to screen approximately 19,400 LER abstracts for potential precursors, (2) the initial screening of those abstracts to determine which should be reviewed in detail, (3) the detailed review of those selected LERs that yielded the 169 events, (4) the categorization of the 169 events, (5) the calculation of function failure estimates based on precursor data, (6) the use of probability of severe core damage estimates to rank precursor events and estimate the frequency of severe core damage, (7) the identification of 52 events considered significant, (8) trends analyses of those significant events, and (9) the identification of the other events of interest that occurred within 1 month of significant events.

NUREG/CR-2497 V02: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENT: 1969-1979. A Status Report. Vol. 2 - Appendix B. MINARICK, J. W. ; KUKIELKA, C. A. Science Applications, Inc. June 1982. 725pp. 8207220678. ORNL/NSIC-182. 14024:062.

Descriptions of 169 operational events reported as Licensee Event Reports, which occurred at commercial light-water reactors during 1969-1979 and which are considered to be precursors to potential severe core damage, are presented, along with associated event trees and categorizations and subsequent analyses. This report summarizes work in (1) the development of methods used to screen approximately 19,400 LER abstracts for potential precursors, (2) the initial screening of those abstracts to determine which should be reviewed in detail, (3) the detailed review of those selected LERs that yielded the 169 events, (4) the categorization of the 169 events, (5) the calculation of function failure estimates based on precursor data, (6) the use of probability of severe core damage estimates to rank precursor events and estimate the frequency of severe core damage, (7) the identification of 52 events considered significant, (8) trends analyses of those significant events, and (9) the identification of the other events of interest that occurred within 1 month of significant events.

NUREG/CR-2505: ELECTRICAL IMPEDANCE STRING PROBES FOR TWO-PHASE VOIDS AND VELOCITY MEASUREMENTS. HARDY, J. E. ; HYLTON, J. O. Oak Ridge National Laboratory. June 1982. 89pp. 8207190054. ORNL/TM-8172. 13920:280.

Report covers an instrumentation scheme developed to measure two-phase flow velocity and void fraction during refill/reflood stages of a loss-of-coolant accident in experimental test facilities. The principle operation was based on measurement of the electrical impedance of two-phase mixtures. Two-phase velocity estimated by time-of-flight analysis of signals from two spatially separated sensors. Capacitive technique employed to measure void fraction. The impedance sensor dubbed "string" probe consists of a pair of stainles steel wires strung back and forth across a stainless steel frame and was designed to withstand temperatures of 350 degrees C, thermal transients of approximately 300 degrees C/s, and severe fluid- and condensation-induced shocks. Void measurements from developed string probes were compared with gamma attenuation densitometer values; velocity measurements by the string probe were compared with calculated phase velocities and turbine meter velocities. In large open-flow areas (such as an upper plenum or end box), good agreement was found between densitometer void values and string sensor voids. Flow velocities detemined by the string probe yielded reasonable agreement when compared with turbine and phase velocities. Generally, the string probe instrumentation (1) proved to be durable in air/water and steam/water flows and (2) demonstrated an ability to measure a wide range of flow velocities (1 to 15 m/s) and void fractions (0 to 0.99+).

NUREG/CR-2512: RADIATION DOSE ESTIMATES AND HAZARDS EVALUATIONS FOR INHALED AIRBORNE RADIONUCLIDES. Annual Progress Report July 1980-June 1981. MEWHINNEY, J. A. Lovelace Biomed & Environmental Research Institute. April 1982. 52pp. 8205060003. LMF-92. 13002:181.

The objective of this project is to conduct confirmatory research on aerosol characteristics and the resulting radiation dose distribution in animals following inhalation and to provide prediction of health consequences in humans due to airborne radioactivity which might be released in normal operations or under accident conditions during production of mixed oxide nuclear fuel. Four research reports summarize the results of research being conducted. The first presents results for several types of physical chemical characterizations of aerosol samples collected at an industrial facility during normal fabrication of mixed oxide fuel. The second paper reports on the methods development process used for measurement of the specific surface area of aerosols, an important determinant in the rate of dissolution of particulates deposited in the lung. The third paper provides updated information on the retention, distrbution and excreticn of Pu after inhalation by Beagles of aerosols of either 750 degrees C treated UO(2) plus PuO(2), 1750 degrees C treated (U,Pu)O(2) or 850 degrees C treated "pure" PuO(2) including the formulation of a biomathematical model useful in describing the results. The fourth paper describes the early results from two studies in which Fischer-344 rats received inhalation exposure to aerosols of (U,Pu)O(2) or "pure" PuO(2) to determine the relationship of radiation dose to biological response.

NUREG/CR-2516 V01 N1: CHARACTERIZATION OF TMI-TYPE WASTES AND SOLID PRODUCTS. Quarterly Progress Report, April-September 1981. SWYLER, K. J.; WEISS, A. J. Brookhaven National Laboratory. May 1982. 48pp. 8206090216. BNL-NUREG-51499. 13456:260.

Progress is reported on a research program to systematically characterize the type of radwastes which may be generated in cleanup procedures following off-normal reactor operations. Specifically, the program is presently investigating how the properties of wastes containing ion-exchange media may be modified by heavy doses of irradiation from sorbed radionuclides. Special effort is being devoted toward quantifying the effects of factors such as radiation dose rate, chemical loading on the ion exchangers, moisture content and composition of external media, etc., which may influence the relation between laboratory test results and field performance.

NUREG/CR-2518: THERMODYNAMIC PROPERTIES OF WATER FOR COMPUTER SIMULATION OF POWER PLANTS. KUCK, I. Z. Arizona, Univ. of. May 1982. 66pp. 8206090125. 13456:304.

Steam property evaluations may represent a significant portion of the computing time necessary for power system simulations. The iterative nature of the solutions for heat transfer and kinetic equations often requires thousands of steam property evaluations during the execution of a single program. Considerable savings may be realized by simplification of property evaluations.

Empirical equations have been obtained for the thermodynamic properties of water in the region of interest. To maintain thermodynamic consistency, the compressibility factor Z, in terms of pressure and temperature, was obtained by curve fitting, and the enthalpy, entropy, and internal energy were derived by standard relationships. Formulations for heat capacity, saturation temperature as a function of saturation pressure, and specific volume of saturated water as a function of the saturation temperature were determined by curve fitting of independent equations. Derivatives were obtained by differentiation of the appropriate formulations.

Evaporator and superheater components of a liquid metal fast breeder reactor power plant simulator were chosen as test cases for the empirical representations. Results obtained using the empirical equations were comparable to those obtained using tabular values and required 24% less computing time.

NUREG/CR-2521: METHOD FOR ESTIMATING WAKE FLOW AND EFFLUENT DISPERSION NEAR SIMPLE BLOCK-LIKE BUILDINGS. HOSKER,R.P. Commerce,Dept. of, National Oceanographic & Atmospheric Administration. June 1982. 158pp. 8207190046. ERL-ARL-108. 13919:155.

This report is intended as an interim guide for those who routinely face air quality problems associated with near-building exhaust stack placement and height, and the resulting concentration patterns. The report consolidates available data and methods for estimating wake flow and effluent dispersion near isolated block-like structures. The near-building and wake flows are described, and quantitative estimates for frontal eddy size, height and extent of roof and wake cavities, and far wake behavior are provided. Concentration calculation methods for upwind, near-building, and downwind pollutant sources are given. For an upwind source, it is possible to estimate the required stack height, and to place upper limits on the likely near-building concentration. The inflences of near-building source location and characteristics relative to the building geometry and orientation are considered. Methods to estimate effective stack height, upper limits for concentration due to flush roof vents, and the effect of changes in rooftop stack height are summarized. Current wake and wake cavity models are presented. Numerous graphs of important expressions have been prepared to facilitate computations and quick estimates of flow patterns and concentration levels for specific simple buildings.

NUREG/CR-2522: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS PROGRAM PLAN. MILLER,R.L.; PAASCH,R.A. United Nuclear Corp. April 1982. 32pp. 8205110257. 13038:077.

This Program Plan describes a multi-year program initiated by the Nuclear Regulatory Commission (NRC) to assess and evaluate the methods, radiation exposure and costs associated with decommissioning retired nuclear facilities. The objective of this program is to provide the NRC licensing staff with comparative data that will allow assessment of decommissioning alternatives for regulatory and ALARA implementation of future decommissioning proposals. The program is currently limited to nuclear reactors.

Licensees currently decommissioning a facility or licensees who are planning decommissioning projects will be solicited for inclusion in the program. An analysis will be performed for each project and will include a comparison of the methods, costs and exposure usage with data contained in generic decommissioning studies.

NUREG/CR-2525 V01: ORNL ROD BUNDLE HEAT TRANSFER TEST DATA. Volume 1-ORNL Small Break LOCA Test Series I: Experimental Data Report. ANKLAM, T. M.; HUNT, D. F.; THOMPSON, M. S.; et al. Oak Ridge National Laboratory. May 1982. 107pp. 8206090135. ORNL/NUREG/TM-4. 13454:254.

The report presents experimental data and calculated steady-state and transient instrument uncertainties from Oak Ridge National Laboratory Small Break Loss of Coolant Accident (LOCA) Heat Transfer Test Series I. The subject test series was composed of six high-pressure, low-flow, quasi-steady-state heat transfer tests and six high-pressure reflood tests. The test series was designed to obtain data under conditions similar to those expected in a small break LOCA. In addition to the experimental data, calculated inlet and outlet mass flows and rod powers are presented.

NUREG/CR-2525 V02: ORNL ROD BUNDLE HEAT TRANSFER TEST DATA. Volume 2 - Thermal -Hydraulic Test Facility Experimental Data Report for Test 3.03.6AR - Transient Film Boiling In Upflow. MULLINS, C. B.; FELDE, D. K.; SUTTON, A. G.; et al. Oak Ridge National Laboratory. May 1982. 249pp. 8206090129. ORNL/NUREG/TM-4. 13455:164.

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) Test 3.03.6AR. This test was conducted on May 21, 1980. The objective of the program was to investigate heat transfer phenomena believed to occur in PWRs during accidents, including small and large break loss-of-coolant accidents.

Test 3.03.6AR was conducted to obtain transient film boiling data in rod bundle geometry under reactor accident-type conditions. The primary purpose of this report is to make the reduced instrument responses for THTF Test 3.03.6AR available. Included in the report are uncertainties in the instrument responses, calculated mass flows, and calculated rod powers.

NUREG/CR-2525 V03: ORNL ROD BUNDLE HEAT TRANSFER TEST DATA. Volume 3-Thermal-Hydraulic Test Facility Experimental Data Report For Test 3.06.6B-Transient Film Boiling In Upflow. MULLINS, C. B.; GOULD, S. S.; FELDE, D. K.; et al. Oak Ridge National Laboratory. June 1982. 260pp. 8206240036. ORNL/NUREG/TM-4. 13610:076.

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) Test 3.06.6B. This test was conducted by members of the Pressurized-Water-Reactor (PWR) Blowdown Heat Transfer (BDHT) Separate-Effects Program on August 29, 1980. The objective of the program was to investigate heat transfer phenomena believed to occur in PWR's during accidents, including small and large break loss-of-coolant accidents. Test 3.06.6B was conducted to obtain transient film boiling data in rod bundle geometry under reactor accident-type conditions. The primary purpose of this report is to make the reduced instrument responses for THTF Test 3.06.6B available. Included in the report are uncertainties in the instrument responses, calculated mass flows, and calculated rod power.

NUREG/CR-2525 V05: ORNL ROD BUNDLE HEAT TRANSFER TEST DATA. Volume 5-Thermal-Hydraulic Test Facility Experimental Data Report For Test 3.08.6C-Transient Film Boiling In Upflow. MULLINS, C. B.; FELDE, D. K.; SUTTON, A. G.; et al. Oak Ridge National Laboratory. June 1982. 259pp. 8207190013. ORNL/NUREG/TM-4. 13922:004.

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) Test 3.08.6C. This test was conducted by members of the Pressurized-Water-Reactor (PWR) Blowdown Heat Transfer (BDHT) Separate-Effects Program on October 1, 1980. The objective of the program was to investigate heat transfer phenomena believed to occur in PWR's during accidents, including small and large break loss-of-coolant accidents. Test 3.08.6C was conducted to obtain transient film boiling data in rod bundle geometry under reactor accident-type conditions. The primary purpose of this report is to make the reduced instrument responses for THTF test 3.08.6C available. Included in the report are uncertainties in the instrument responses, calculated mass flows, and calculated rod powers.

NUREG/CR-2525 V07: ORNL ROD BUNDLE HEAT TRANSFER TEST DATA. Volume 7-Thermal-Hydraulic Test Facility Experimental Data Support For Test Series 3.07.9-Steady-State Film Boiling In Upflow. MULLINS, C. B.; FELDE, D. K.; SUTTON, A. G.; et al. Oak Ridge National Laboratory. June 1982. 101pp. 8207190021. ORNL/NUREG/TM-4. 13923:058.

Thermal-Hydraulic Test Facility (THTF) test series 3.07.9 was conducted by members of the Pressurized-Water Reactor (PWR) Blowdown Heat Transfer (BDHT) Separate-Effects Program on September 11, September 18, and October 1, 1980. The objective of the program was to investigate heat transfer phenomena believed to occur in PWRs during accidents, including small- and large-break loss-of-coolant accidents. Test series 3.07.9 was designed to provide steady-state film boiling data in rod bundle geometry under reactor accident-type conditions. This report presents the reduced instrument responses for THTF test series 3.07.9. Also included are uncertainties in the instrument responses, calculated mass flows, and calculated rod powers.

NUREG/CR-2542: SENSITIVITY STUDY USING THE FRANTIC CODE FOR THE UNAVAILABILITY OF A SYSTEM TO THE FAILURE CHARACTERISTICS OF THE COMPONENTS AND THE OPERATING CONDITIONS. GINZBERG, T.; DICKEY, J. M.; HALL, R. E. Brookhaven National Laboratory. May 1982. 114pp. 8206100087. BNL-NUREG-51504. 13480:145.

The purpose of this report is to show how the code FRANTIC II can be used to explore the sensitivity of the calculated unavailability of a system to various parameters. Several generic systems are analyzed in detail. The study illustrates the effect of uncertainties in the empirical data and helps assess the relative importance of collecting more accurate data for particular components. The study also helps in deciding whether it is worth improving a particular component or whether such improvement would have little effect. In addition, the impact of changing operational procedures can be assessed, and different operating strategies can be compared. Change of some procedures may yield only slight improvement, whereas in other instances a small change in timing may have significant effect. Thus, this report shows how FRANTIC II may be a useful and powerful tool in the analysis of the reliability of complex systems, and in the determination of the more significant factors.

NUREG/CR-2543: A STUDY OF THE FEASIBILITY OF MICROWAVE DIELECTRIC HEATING FOR LMFBR TRANSITION PHASE ACCIDENT SEQUENCE BOILING STUDIES. MAKOWITZ, H.; GINSBERG, T. Brookhaven National Laboratory. May 1982. 121pp. 8206100017. BNL-NUREG-51506. 13475:234.

A study is reported on the feasibliity of the use of microwave dielectric heating to simulate the nuclear heat source in LMFBR "transition phase" accident sequence volume-boiling simulation experiments. The adequacy of microwave heating is judged based upon the criterion of heating uniformity per unit liquid volume and upon the ability to analytically characterize the liquid power density distribution. Two aspects of liquid power density uniformity are addressed. First, the effect of liquid geometry on power density is studied in order to determine whether millimeter-size droplets can be heated as efficiently as centimeter-scale masses which are exposed to the same source of radiation. Both analyses and experiments were performed in this portion of the study. Second, the spatial distribution of power density across liquid slabs is studied, in order to determine whether wave interference effects, which lead to severe power density gradients, can be minimized by choice of suitable dielectric liquids. The above analyses were carried out for a variety of wavelengths within the microwave radiation band, for several dielectric liquids and for a range of temperature.

NUREG/CR-2544: TWO-PHASE MASS FLUX UNCERTAINTY ANALYSIS FOR THERMAL-HYDRAULIC TEST FACILITY INSTRUMENTED SPOOL PIECES. CHEN, N.C.; FELDE, D.K. Oak Ridge National Laboratory. June 1982. 40pp. 8206220019. ORNL/TM-7859. 13584:324.

An analysis of two-phase mass flux uncertainties for the Thermal-Hydraulic Test Facility (THTF) instrumented spool pieces is presented. Comparisons are made between various homogeneous mass flux models based on hign-temperature and high-pressure water mass flux data from steady-state upflow film boiling tests run at the THTF. Subcooled flow at the test section inlet provides a well-defined standard for inplace evalution of the mass flux models at the high-quality, two-phase test section outlet. Additionally, a transient two-phase turbine meter model developed by Kamath and Lahey is applied to two transient THTF tests to assess the sensitivity of the calculated mass flux to uncertainties in two-phase flow parameters and transient response as applied specifically to THTF test conditions.

NUREG/CR-2545: DESIGN CONCEPT AND TESTING OF AN IN-BUNDLE GAMMA DENSITOMETER FOR SUBCHANNEL VOID FRACTION MEASUREMENTS IN THE THTF ELECTRICALLY HEATED ROD BUNDLE. FELDE, D.K. Oak Ridge National Laboratory. May 1982. 26pp. 8206110005. 13493:301.

A design concept is presented for an in-bundle gamma densitometer system for measurement of subchannel average fluid density and void fraction in rod or tube bundles. This report describes (1) the application of the design concept to the Thermal-Hydraulic Test Facility (THTF) electrically heated rod bundle and (2) results from tests conducted in the THTF.

NUREG/CR-2546: REACTOR SAFEGUARDS AGAINST INSIDE SABOTAGE. BENNETT, H.A. Sandia Laboratories. June 1982. 104pp. 8206290049. SAND82-0319. 13661:165.

A conceptual safeguards system is structured to show how both reactor operations and physical protection resources could be integrated to prevent release of radioactive material caused by insider sabotage. Operational recovery capabilities are addressed from the viewpoint of both detection of and response to disabled components. Physical protection capabilities for preventing insider sabotage through the application of work rules are analyzed. Recommendations for further development of safeguards system structures, operational recovery, and sabotage prevention are suggested.

NUREG/CR-2551: RANK ORDERING OF VITAL AREAS WITHIN NUCLEAR POWER PLANTS. RICHARDSON, J. M. Sandia Laboratories. June 1982. 38pp. 8206250040. SAND82-0332. 13628:168.

The conceptual development of a methodology for rank order of vital areas within nuclear power plants based upon times associated with sabotage events and their consequences is discussed. The important time parameters in the analysis include the time required to detect the perpetration of sabotage, the time required to repair or mitigate the consequences of the sabotage and the total time available to perform these functions before it is too late to reverse the damage. These time interval parameters are incorporated into an interruption analysis importance measure that provides information on the ability of the protection systems to cope with the results of the sabotage. A consequence analysis that considers categories of release characteristics is the next step in the ranking scheme. Results of the interruption and consequence analyses are combined to attain a risk index associated with each vital area. The final ranking can be used to order upgrade priorities and to allocate scarce protection resources effectively.

NUREG/CR-2559: RESULTS OF PHASE ONE OF PLANT ELECTRICAL SYSTEM (PES) STUDY. WYANT, F. J.; FURGAL, D. T. Sandia Laboratories. April 1982. 51pp. 8205130248. SAND82-0377. 13089:338.

This report summarizes initial scoping study efforts assessing nuclear power plant electrical system performance. Actual component failures and off-normal load and electrical power line conditions were determined. Sources of information (data bases) are discussed. A methodology for coding and classifying Plant Electrical System (PES) events is presented. Data from 9 LER monthly reports is categorized by component. This information is rank-ordered and cross-tabulated by frequency of occurrence, generic component, type of reactor, specific plant, component vendor and system interactive failure mode. Recommendations for further study are presented.

NUREG/CR-2564: ENVIRONMENTAL FACTORS AFFECTING LONG-TERM STABILIZATION OF RADON SUPPRESSION COVERS FOR URANIUM MILL TAILINGS. YOUNG, J. K.; LONG, L. W.; REIS, J. W. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1982. 110pp. 8205110103. PNL-4193. 13037:165.

Pacific Northwest Laboratory is investigating the use of a rock armoring blanket (riprap) to mitigate wind and water erosion of an earthen radon suppression cover applied to uranium mill tailings. To help determine design stresses for the tailings piles, environmental parameters are characterized for the five active uranium-producing regions on a site-specific basis. Only conventional uranium mills that are currently operating or that are scheduled to open in the mid 1980's are considered.

Available data indicate that flooding has the most potential for disrupting a tailings pile. The arid regions of the Wyoming Basins and the Colorado Plateau are subject to brief storms of high intensity. The Texas Gulf Coast has the highest potential for extreme precipitation from hurricane-related storms. Wind data indicate average wind speeds from 3 to 6 m/sec for sites, but extremes of 40 m/sec can be expected. Tornado risks range from low to moderate. The Colorado Plateau has the highest seismic potential, with maximum acceleration caused by earthquakes ranging from 0.2 to 0.4 g. Any direct effect from volcanic eruption is negligible, as all mills are located 90 km or more from an igenous or hydrothermal system.

NUREG/CR-2565: STRUCTURAL PERFORMANCE OF HEPA FILTERS UNDER SIMULATED TORNADO CONDITIONS. HORAK, H.L.; SMITH, P.R. Los Alamos Scientific Laboratory. GREGORY, W.S.; et al. Northeast Missouri State Univ. May 1982. 103pp. 8205180101. LA-9197-MS. 13134:165.

This report contains the results of structural tests to determine the response of High Efficiency Particulate Air filters to stimulated tornado conditions. The data include the structural limits of the filters, their resistance at high flow rates, and the effects of filter design features and tornado parameters.

Considering all the filters tested, the mean break pressure or structural limit was found to be 2.35 psi (16.2 kPa). The maximum valve was 2.87 psi (19.8 kPa), and the low value found was 1.31 psi (9.0 kPa). The type of failure was usually a medium break of the downstream filter fold.

The type of filters that we evaluated were nuclear grade with design flow rates of 1000 cfm (0.47 m(3)/s), standard separators, and folded medium design. The parameters evaluated that are characteristic of the filter included manufacturer, separator type, faceguards, pack tightness, and aerosol loading. Manufacturer and medium properties were found to have a large effect on the structural limits.

The tests results are independent of tornado type. The parameters we examined that are characteristic of tornados are pressurization rate and flow duration. These two parameters did not have a major effect on the break pressures.

NUREG/CR-2567: FINAL DATA REPORT FOR THE INSTRUMENTED FUEL ASSEMBLY (IFA)-432. BRADLEY, E.R.; CUNNINGHAM, M.E.; LANNING, D.D. Battelle Memorial Institute, Pacific Northwest Laboratory. June 1982. 82pp. 8207060339. PNL-4240. 13744:191.

This report presents the in-reactor data collected during the irradiation of the six-rod instrumented fuel assembly (IFA)-432 in the Halden (Norway) Boiling Water Reactor (HBWR) from June 1980 through June 1981. This assembly (designed by PNL) was one of a series of NRC sponsored tests to obtain data for the development and assessment of steady-state fuel performance computer codes. IFA-432 operated from December 1975 until June 1981, when it was removed from the reactor. Burnup levels in excess of 30,000 MWD/MTM were achieved. Data collected prior to June 1980 were reported in NUREG/CR-0560 and NUREG/CR-1950.

Fuel centerline temperatures, cladding elongations, internal fuel rod pressures and local powers were monitored during the irradiation. Detailed analysis of the data reported is not made.

NUREG/CR-2569: RESPONSE OF THE ZION & INDIAN POINT CONTAINMENT BUILDINGS TO SEVERE ACCIDENT PRESSURES. BUTLER, T.A.; FUGELSO, L.E. Los Alamos Scientific Laboratory. May 1982. 41pp. 8206170051. LA-9301-MS. 13555:240.

The failure modes and associated failure pressures for two common generic types of pressurized water reactor (PWR) containments are predicted. One building type is a lightly reinforced, post-tension structure represented by the Zion nuclear reactor containment. The other is the normally reinforced Indian Point Containment. Two-dimensional models of the buildings developed using the finite element method are used to predict the failure modes and failure pressures. A three-dimensional finite model is used to evaluate the Zion building's equipment hatch penetration. Predicted failure modes for both containments involve loss of structural integrity at the intersection of the cylindrical sidewall with the base slab. The response of the Indian Point building to postulated detonation of a hydrogen-air mixture in the containment dome is also calculated.

NUREG/CR-2570: EXPERIMENTAL INVESTIGATION OF TEARING INSTABILITY PHENOMENA FOR STRUCTURAL MATERIALS. VASSILAROS, M. G.; GUDAS, J. P.; JOYCE, J. A. David W. Taylor Naval Research & Development Center. April 1982. 40pp. 8205120122. 13053:263.

The objective of this investigation was to extend the range of tearing instability validation experiments utilizing the compact specimen to include high toughness alloys. J-Integral tests of ASTM A106; ASTM A516: Grade 70; ASTM A533; HY-80; and HY-130 steels were performed in a variable compliant screw-driven test machine. Results were analyzed with respect to the materials J(I)-R curves and various models of T(applied) for the compact specimen. Tearing instability theory was validated for these high toughness materials. For the cases of highly curved J(I)-R curves, it was shown that the actual value of T(material) at the point of instability should be employed rather than the average T(material) value. The T(applied) analysis of Paris coworkers applied to the compact specimen appears to be nonconservative in predicting the point of instability; whereas, the T(applied) analysis of Ernst and coworkers appears to be accurate, but requires precision beyond that displayed in this program. The generalized Paris analysis applied to the compact specimen and evaluated at maximum load was most consistent in predicting instability.

NUREG/CR-2581: SOME EFFECTS OF ELECTRONS SLOWING DOWN IN MATERIALS WITH APPLICATION TO SAFETY-RELATED EQUIPMENT QUALIFICATION. BUCKALEW, W. H.; WYANT, F. J. Sandia Laboratories. April 1982. 54pp. 8205120131. SAND82-0449. 13051:280.

Theoretical predictions have been made of the bremsstrahlung environments resulting from the slowing down of electrons in selected materials. Several materials, material thicknesses, and electron energies were considered. Parameters, of particular interest, obtained were transmitted photon energy and spectra. These data provide a means for estimating the effects of beta-emitting isotopes, released during a reactor loss of coolant accident (LOCA) or other accident scenario, on systems and components shielded by an enclosure or housing.

NUREG/CR-2582: RADIATION CAPABILITIES OF THE SANDIA HIGH INTENSITY ADJUSTABLE COBALT ARRAY. BUCKALEW, W. H.; THOME, F. V. Sandia Laboratories. June 1982. 62pp. 8206160054. SAND81-2655. 13538:042.

The High Intensity Adjustable Cobalt Array radiation facility has been characterized for several source strengths and geometries using a three-dimensional array of self-biasing photodiodes interfaced with automated data acquisition, reduction, and display equipment. Maximum dose rate achievable in a 24-in.-long x 22-in.-diameter volume is about 1.5 Mrd/h. Other source configurations can be selected also, e.g., fields 48 in. long x 22 in. diameter produce dose rates on the order of 0.8 Mrd/h. Even higher dose rates can be obtained by reducing the radiation volume.

NUREG/CR-2584: METEROLOGICAL CONSIDERATIONS IN THE DEVELOPMENT OF A REAL-TIME ATMOSPHERIC DISPERSION MODEL FOR REACTOR EFFLUENT EXPOSURE PATHWAY. VAN DER HOVEN Commerce, Dept. of, National Oceanographic & Atmospheric Administration. May 1982. 20pp. 8206220029. 13583:329.

Meteorological considerations, as part of an overall emergency plan in the event of an inadvertent atmospheric release of radioactive effluents from a nuclear reactor, are discussed in terms of the site meteorological measurement capability, the atmospheric transport and diffusion prediction requirements, the source term configuration, and the requirements posed by special site characteristics such as coastal, valley, and mountainous locations.

NUREG/CR-2586: A SURVEY OF METHODS FOR IMPROVING OPERATOR ACCEPTANCE OF COMPUTERIZED AIDS. FREY, P.R.; KISNER, R.A. Oak Ridge National Laboratory. April 1982. 30pp. 8205130260. ORNL/TM-8236. 13089:009.

The purpose of this report is to draw from the literature factors related to user acceptance of computerized equipment that may also be applicable to the acceptance of computerized aids used in the nuclear power plant control room. A review of the available literature revealed about seventy papers that deal with acceptance problems in computerized systems. Two attempts to define and measure the characteristics of a user-acceptable system in nonnuclear industries form a basis for future work on this subject in the nuclear industry. Operator acceptance of computerized aids can be influenced during design, operator training and system operation. Design methods for improving acceptance include allowing the user to participate in the design process, considering acceptance principles in the allocation of functions between the man and machine, minimizing the length and variation of the system response times, tailoring the dialogue to the task and use, integrating the system into the control room and providing usable system documentation. During operator training, acceptance considerations include providing adequate detail on the purposes and limitations of the system, ensuring that the training situations approximate the expected operational situations and providing training for subsequent generations of operators. The primary acceptance considerations during operation are system availability and system calibration.

NUREG/CR-2587: FUNCTIONS AND OPERATIONS OF NUCLEAR POWER PLANT CREWS. KISNER, R.A.; FREY, P.R. Oak Ridge National Laboratory. May 1982. 91pp. 8206100067. ORNL/TM-8237. 13472:301.

This report summarizes the results of work performed to define the functions, operations, and organization of nuclear power plant operating crews. The primary information sources used were ANS and IEEE standards, normal and emergency operating procedures from nuclear power plants, interviews, and literature reviews. The function and organization of operating crews for several plants are discussed generically. The report covers a wide spectrum of topics including review of standards affecting human factors in the control room, influences of automation on operator functions, classification of operator functions, function of operator at onset of emergency, crew organization, work-induced stress, and operator acceptance of his role.

NUREG/CR-2588: SECURITY OFFICER RESPONSE STRATEGIES (SECURORS). ROUNTREE, S. L. K. Sandia Laboratories. June 1982. 46pp. 8206240041. SAND82-0410. 13609:299.

The Security Officer Response Strategies (SECURORS) approach provides a method for deploying security officers within a nuclear power plant subsequent to an adversary intrusion detection. Under current nuclear power plant operating conditions, the number of vital areas generally exceeds the number of security officers. The SECURORS method allocates the available officers on the basis of numerical weights and ranking for each of the nuclear power plant vital areas and barriers. It is assumed that the numerical weights have been obtained previously from readily available techniques or from expert opinion on the vulnerability of vital areas and barriers. This paper does not establish any methodology for the derivation of weights, but vital area characteristics related to the numerical weights and ranking are reviewed. An example illustrates the integer programming problem formulation and solution process for several deployment strategies. The SECURORS approach builds on the results of several procedures and analytic techniques. It is assumed that the nuclear power plant has undergone a vital area analysis. Additional results can be obtained from the Safeguards Automated Facility Evaluation (SAFE) method. A glossary is provided to clarify safeguards and mathematical programming terminology.

NUREG/CR-2589: A GROUND-PENETRATING RADAR SURVEY OF THE MAXEY FLATS LOW-LEVEL NUCLEAR WASTE DISPOSAL SITE, FLEMING COUNTY, KENTUCKY. HORTON, K. A. Geo-Centers, Inc. June 1982. 48pp. 8207190065. GC-TR-82-171. 13920:058.

A ground-penetrating radar survey was conducted at the Maxey Flats Low-Level Nuclear Waste Disposal Site, Kentucky, to more accurately determine the location of burial trenches and pits, and to identify locations and depths of any prominent subsurface features.

A geologic/electromagnetic model of the site was developed and utilized for analysis of the acquired data. Depths of penetration derived from radar records correlated well with those calculated from the model. A final interpretation of the radar data is presented.

NUREG/CR-2591: ESTIMATING THE POTENTIAL INDUSTRIAL IMPACTS OF A NUCLEAR REACTOR ACCIDENT. CARTWRIGHT, J. V.; BEEMILLER, R. M.; TROTT, E. A.; et al. Commerce, Dept. of. April 1982. 136pp. 8205190035. 13185:047.

This NUREG describes an industrial impact model that can be used to estimate the regional industry-specific impacts of disasters, both natural and manmade. Special attention is given to the impacts of possible nuclear reactor accidents. The report also presents three applications of the model. The impacts estimated in the case studies are based on (1) general information and reactor-specific data, supplied by the U.S. Nuclear Regulatory Commission (NRC); regional economic models derived from the Regional Input-Output Modeling System (RIMS II) developed at the Bureau of Economic Analysis (BEA); and (3) additional methodology developed especially for taking into account the unique characteristics of a nuclear reactor accident with respect to regional industrial activity.

NUREG/CR-2593: A USER'S MANUAL FOR COMPUTER CODE RIBD/IRT. THAYER, D. D.; LURIE, N. A. Sandia Laboratories. April 1982. 76pp. 8205200260. SAND82-7013. 13195:291.

The computer code RIBD/IRT is a modified version of RIBD-II. It is a grid processor that calculates isotopic concentrations resulting from two fission sources with normal down-chain decay by beta emission and isomeric transfers and inter-chain coupling resulting from n-gamma reactions. Calculations can be made to follow an irradiation history through an unlimited number of step changes of unrestricted duration and variability including shutdown periods, restarts at different power levels and/or any other level changes. Output information includes time-dependent inventories, activities, decay powers, and energy releases for as many as 800 fission products. Modifications to RIBD-II were necessitated by Loss-of-Coolant Accident (LOCA) studies conducted by IRT Corporation regarding fission product source term definition. These modifications permit the user to track and modify the concentrations of individual elements as they decay with time following reactor shutdown. In essence, one can determine time-dependent fission product source terms resulting from any reactor operating history which then can be used as input into fission product transport codes. Other modifications to RIBD-II expanded the output information to assist the user in analyzing the source term. This manual describes the modifications to RIBD/II, input requirements and a sample problem. The appendicies give a listing of RIBD/IRT, sample output, and a listing of a code called ZIP which prepares the library tape for input to RIBD/IRT. The code is available in a UNIVAC 1100/81 version and a VAX 11/780 version.

NUREG/CR-2594: A USER'S MANUAL FOR THE GABAS SPECTRUM COMPUTER CODE. THAYER, D. D.; LURIE, N. A. Sandia Laboratories. April 1982. 55pp. 8205200270. SAND82-7014. 13203:117.

The Gamma and Beta Spectrum computer code (GABAS) was developed at IRT Corporation for calculating time-dependent beta and/or gamma spectra from decaying fission products. GABAS calculates composite fission product spectra based on the technique used by England, et al., in conjunction with the CINDER family of fission product codes.

Multigroup beta and gamma spectra for individual nuclides are folded with their corresponding time-dependent activities (usually generated by a fission product inventory code) to produce a composite time-dependent fission product spectrum. This manual contains the methodology employed by GABAS, input requirements for proper execution, a sample problem and a FORTRAN listing compatible with a UNIVAC machine

The code is available in a UNIVAC 1100/81 version and a VAX 11/780 version. The former may be obtained from the Radiation Shielding Information Center (RSIC); the latter may be obtained directly from IRT Corporation.

NUREG/CR-2597: STEADY-STATE PRESSURE LOSSES FOR MULTIROD BURST TEST (MRBT) BUNDLE B-5. BAILEY, P. T. Babcock & Wilcox Co. May 1982. 107pp. 8206100053. ORNL/SUB/80-404. 13472:004.

This report describes the water-flow-test of 64-rod PWR fuel assembly simulation which was tested under loss-of-coolant-accident (LOCA) conditions. The test, involving cladding deformation and rupture in the temperature region of the Zircaloy alpha phase, was performed on May 30, 1980. The average of burst temperatures and pressure differentials were 773 degrees and 8,806 kPa.

1. R. H. Chapman et al., Quick-look Report on MRBT B-5 (8 x 8)

Bundle Test, Internal Report ORNL/MRBT-5 (July 1980).
2. R. H. Chapman et al., Multirod Burst Test Program Prog. Rep. January-June 1980, NUREG/CR-1883 (ORNL/NUREG/TM-426).
3. A. W. Longest, Multirod Burst Test Program Prog. Rep. January-June 1981, NUREG/CR-2366, Vol. 1, ORNL/TM-8058.
4. J. L. Crowley, Multirod Burst Test Program Prog. Rep. July-December 1981, NUREG/CR-2366, Vol. 2 ORNL/TM-8190.

This report describes the work characterizing the hydraulic resistance of the B-5 bundle. In addition to the flow test of the deformed bundle, B&W assembled and flow tested an undeformed reference bundle (designated as B-5R) to provide comparative data. Magnetic tapes containing the raw test data, reduced test data, and calibration records of the B&W flow tests are on file at ORNL.

NUREG/CR-2600: END-OF-IRRADIATION DATA REPORT FOR THE INSTRUMENTED FUEL ASSEMBLY (IFA)-527. CUNNINGHAM, M. E.; LANNING, D. D. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1982. 116pp. 8205200293. PNL-4201. 13203:001.

This report presents data obtained during the irradiation of the six-rod instrumented fuel assembly (IFA)-527 in the Halden Boiling Water Reactor (HBWR), Halden, Norway. This assembly is the last in a series of U.S. Nuclear Regulatory Commission (NRC)-sponsored tests to obtain data for the development and verification of steady-state fuel performance computer codes. IFA-527 contains five identical rods with high-density stable fuel pellets and 230-um diametral gaps and one rod with similar fuel pellets but with a 60-um diametral gap. All six rods were xenon-filled to simulate the effects of fission gas and to enhance the observable effects of fuel cracking and relocation on fuel temperatures. The assembly operated successfully from July 1, 1980, to August 15, 1980; the reactor was then shut down until September 10, 1980. During the shutdown, at least four of the six rods suffered pressure boundary failures. Irradiation of the assembly continued with the failed rods from September 10, 1980, until April 8, 1981; the assembly was then removed from the reactor. This report presents both pre- and postfailure data for IFA-527.

NUREG/CR-2603: BUBBLE BEHAVIOR IN LMFBR CORE DISRUPTIVE ACCIDENTS. REYNOLDS, A. B.; ERDMAN, C. A.; BRADLEY, D. R.; et al. Virginia, Univ. of. April 1982. 90pp. 8205040021. 12971:008.

Research performed at the University of Virginia during FY '81 for the Advanced Reactor Safety Research Division of the U.S. Nuclear Regulatory Commission is reported. The research is part of the LMFBR Aerosol Release and Transport Program. Principal areas investigated were (1) analysis of ORNL FAST underwater tests, (2) pretest parametric analysis of ORNL under sodium tests, (3) axial motion of large expanding and collapsing bubbles, and (4) measurement of droplet sizes from flashing. Analysis of the FAST tests with the UVABUBL code showed the strong influence of water vapor during the bubble expansion; water vapor rapidly replaces UO(2) vapor as the vapor that drives the bubble. In the case of the under sodium tests, it is expected that entrained sodium will vaporize and influence bubble behavior, but, unlike water, sodium will not be vaporized from the bubble surface. Earlier analyses of axial motion of the French EXCOBULLE experiments were improved. Experimental methods in the experiment on droplet sizes from flashing were developed further.

NUREG/CR-2604: THE SNAP OPERATING SYSTEM (SOS) USER'S GUIDE. SABUDA, J. D. ; WALKER, J. L. ; POLITO, J. ; et al. Sandia Laboratories. May 1982. 438pp. 8205190010. SAND82-7018. 13185:183.

The SNAP Operating System (SOS) is a FORTRAN 77 program which provides assistance to the safeguards analyst who uses the Safeguards Automated Facility Evaluation (SAFE) and the Safeguards Network Analysis Procedure (SNAP) techniques. Features offered by SOS are a data base system for storing a library of SNAP applications, computer graphics representation of SNAP models, a computer graphics editor to develop and modify SNAP models, a SAFE-to-SNAP interface, automatic generation of SNAP input data, and a computer graphics post-processor for SNAP. The SOS User's Guide is designed to provide the user with the information necessary to use SOS effectively. Examples are used throughout to illustrate the concepts. The format of the user's guide follows the same sequence as would be used in executing an actual application.

NUREG/CR-2605: THE SNAP OPERATING SYSTEM REFERENCE MANUAL. SABUDA, J. D. ; POLITO, J. ; WALKER, J. L. Sandia Laboratories. May 1982. 268pp. 8206100058. SAND82-7019. 13467:137.

The SNAP Operating System (SOS) is a FORTRAN 77 program which provides assistance to the safeguards analyst who uses the Safeguards Automated Facility Evaluation (SAFE) and the Safeguards Network Analysis Procedure (SNAP) techniques. Features offered by SOS are a data base system for storing a library of SNAP applications, computer graphics representation of SNAP models, a computer graphics editor to develop and modify SNAP models, a SAFE-to-SNAP interface, automatic generation of SNAP input data, and a computer graphics post-processor for SNAP. The SOS Reference Manual provides detailed application information concerning SOS as well as a detailed discussion of all SOS components and their associated command input formats.

NUREG/CR-2610: RAGBEEF: A FORTRAN IV IMPLEMENTATION OF A TIME-DEPENDENT MODEL FOR RADIONUCLIDE CONTAMINATION OF BEEF. PLEASANT, J. C. ; MCDOWELL-BOYER; KILLOUGH, G. G. Oak Ridge National Laboratory. June 1982. 145PP. 8207190018. ORNL/TM-8011. 13922:278.

RAGBEEF is a FORTRAN IV program that calculates radionuclide concentrations in beef as a result of ingestion of contaminated feeds, pasture, and pasture soil by beef cattle. The model implemented by RAGBEEF is dynamic in nature, allowing the user to consider age-and season-dependent aspects of beef cattle management in estimating concentrations in beef. It serves as an auxiliary code to RAGTIME, previously documented by the authors, which calculates radionuclide concentrations in agricultural crops in a dynamic manner, but evaluates concentration in beef for steady-state conditions only. The time-dependent concentrations in feeds, pasture, and pasture soil generated by RAGTIME are used as input to the RAGBEEF code. RAGBEEF, as presently implemented, calculates radionuclide concentrations in the muscle of age-based cohorts in a beef cattle herd. Concentrations in the milk of lactating cows are also calculated, but are assumed age-independent as in RAGTIME. This report describes the age-and season-dependent considerations making up the RAGBEEF model, as well as presenting the equations which describe the model and a documentation of the associated computer code. Listing of the RAGBEEF and updated RAGTIME codes are provided in appendices, as are the results of a sample run of RAGBEEF and a description of recent modifications to RAGTIME.

NUREG/CR-2611: MGO AND 70 W% UO2-30W% Y2O3: THERMOPHYSICAL AND TRANSIENT PROPERTIES. PILCH,M. Sandia Laboratories. April 1982. 28pp. 8205060079. SAND81-1230. 13007:286.

Interactions between a molten core simulant and MgO bricks (Harklase) are being studied. A molten core simulant, consisting of 70 w% UO(2) and 30 w% Y(2)O(3), has been proposed for use in large scale experiments (200 kg) at Sandia's Large Melt Facility. This report documents the binary phase diagrams, thermophysical properties, and transport properties which are necessary for the analysis of these experiments.

NUREG/CR-2612: VARIABILITY IN DOSE ESTIMATES ASSOCIATED WITH THE FOOD CHAIN TRANSPORT AND INGESTION OF SELECTED RADIONUCLIDES. HOFFMAN,F.O.; GARDNER,R.H.; ECKERMAN,K.F. Oak Ridge National Laboratory. June 1982. 50pp. 8206240059. ORNL/TM-8099. 13608:290.

Dose predictions for the ingestion of 90Sr and 137Cs, using aquatic and terrestrial food chain transport models similar to those in the Nuclear Regulatory Commission's Regulatory Guide 1.109, are evaluated through estimating the variability of model parameters and determining the effect of this variability on model output. The variability in the predicted dose equivalent is determined using analytical and numerical procedures. In addition, a detailed discussion is included on 90Sr dosimetry. The overall estimates of uncertainty are most relevant to conditions where site-specific data is unavailable and when model structure and parameter estimates are unbiased.

NUREG/CR-2618: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A NATURAL CIRCULATION TEST S-NC-7C. LARSON,R.A. EG&G, Inc. April 1982. 48pp. 8205130237. EGG-2179. 13075:312.

This report presents test data recorded for Test S-NC-7C of the Semiscale Mod-2A Natural Circulation Test Series. This is one of several Semiscale tests that investigate the thermal-hydraulic phenomena resulting from operational transients or small-break loss-of-coolant accidents (LOCAs) involving loss of mechanical primary coolant circulation in a pressurized water reactor. These tests produce experimental data to develop and assess the analytical capability of computer models used to predict the results of such small-break LOCAs and operational transients.

The primary objectives of Test S-NC-7C were to experimentally characterize the relationship of natural circulation flow to primary system inventory, and to examine the influence on system behavior of imbalancing the secondary side of one loop.

This report presents the uninterpreted data from Test S-NC-7C for analysis. The data, presented as graphs in engineering units, have been analyzed only to the extent necessary to ensure that they are reasonable and consistent.

NUREG/CR-2622: ANALYSIS OF TRAC AND SCTF RESULTS FOR SYSTEM PRESSURE-EFFECTS TESTS UNDER FORCED FLOODING (RUNS 506,507 AND 508). SUDO,Y. Los Alamos Scientific Laboratory. May 1982. 64pp. 8205180117. LA-9258-MS. 13134:273.

The Transient Reactor Analysis Code (TRAC) and the Slab Core Test Facility (SCTF) results are compared for the three system pressure-effects tests (Runs 506, 507, and 508) with forced injection into the lower plenum. The results show that TRAC can predict well the overall transients of core rod temperature, core differential pressure, and the liquid carryover into the hot leg, as well as in the upper plenum, effects that are strongly dependent on the system pressure. Comparisons are also presented that show major differences between the SCTF test and the TRAC results that should be improved in the future.

NUREG/CR-2625: CRITICAL PATHWAYS OF RADIONUCLIDES TO MAN FROM AGRO-ECOSYSTEMS. Annual Progress Report, October 1980-September 1981. SMITH, M. H.; ALBERTS, J. J.; ADRIANO, D. C.; et al. Savannah River Laboratory. April 1982. 50pp. 8205110077. 13038:245.

The research has as its objective describing the fate and behavior in the environment of radionuclides from nuclear fuel reprocessing.

Greenhouse radionuclide uptake studies which examined factors possibly altering phytoavailability of radionuclides show only slight differences among crop species or soil treatments (lime or lime plus chelate) in Pu or Cm uptake. The temporal effect on Pu and Cm uptake, from a partial data set, is inconclusive, with variable effects from crop species, radionuclides, and soil treatments. Cesium uptake shows variable response with crop species generally decreasing with time.

Field grown broadleaf crops grown have differing Pu concentrations observed in wheat and soybean crops. In crops tending to trap aerially deposited Pu, washing removed more than 50% of the Pu.

Uranium contamination of a wheat crop grown near the separations facility appears to be strongly affected by root uptake. This is in contrast to the behavior of Pu where superficial pathways are the dominant modes of contamination, and is probably treated to (1) the ubiquitous presence of naturally ocurring U isotopes and (2) a greater concentration ratio for U than for Pu.

NUREG/CR-2629: INTERIM SOURCE TERM ASSUMPTIONS FOR EMERGENCY PLANNING AND EQUIPMENT QUALIFICATION. NIEMCZYK, S. J. Oak Ridge National Laboratory. June 1982. 141pp. 8206290519. ORNL/TM-8274. 13646:160.

The source terms recommended in the current regulatory guidance for considerations of light water reactor (LWR) accidents were developed a number of years ago when understanding of many of the phenomena pertinent to source term estimation was not well developed. The purpose of the work presented here was to review the literature on accident source term research and utilize the recent research to develop more realistic assumptions for calculation of accident source terms which could be used for regulatory purposes for two specific considerations, namely, equipment qualification and emergency planning.

The emphasis of this work was on developing appropriate assumptions for estimating the magnitude of the radionuclide releases for various groups of accidents in each of the accident spectra of concern. The overall approach taken was to adopt basic assumptions and models previously proposed for various aspects of source term estimation and to modify those assumptions and models to reflect recently gained insights into, and data describing the release and transport of radionuclides during and after light water reactor accidents. The report presents results of sample calculations of accident source terms and compares the results with the other published results. The report also presents peer review comments on this study.

NUREG/CR-2632: RESPONSE OF CENTRIFUGAL BLOWERS TO SIMULATED TORNADO TRANSIENTS. July-September 1981. IDAR, E. S. ; MARTIN, R. A. ; GREGORY, W. S. ; et al. Los Alamos Scientific Laboratory. May 1982. 22pp. 8206100062. LA-9276-SR. 13467:116.

During this quarter, quasi-steady and dynamic testing of the 24-in. centrifugal blower was completed using the blowdown facility located at New Mexico State University. The data were obtained using a new digital data-acquisition system. Software was developed at the Los Alamos National Laboratory to reduce the dynamic test data and create computer-generated movies showing the dynamic performance of the blower under simulated tornado transient pressure conditions relative to its quasi-steady-state performance.

Currently, quadrant-four (outrunning flow) data have been reduced for the most severe and a less severe tornado pressure transient. The results indicate that both the quasi-steady and dynamic blower performance are very similar. Some hysteresis in the dynamic performance occurs because of rotational inertia effects in the blower rotor and drive system. Currently quadrant-two (backflow) data are being transferred to the LTSS computer system at Los Alamos and will be reduced shortly.

NUREG/CR-2633: CONTAINMENT REACTOR CAVITY SUBCOMPARTMENT ANALYSIS PROCEDURES FOR A BOILING WATER REACTOR. TURK, W. V. ; GIDO, R. G. ; LI, C. Y. Los Alamos Scientific Laboratory. May 1982. 32pp. 8206110015. LA-9277-MS. 13492:163.

Procedures for the performance of Boiling Water Reactor (BWR) cavity subcompartment analysis are presented. The purpose of this presentation is to normalize the analysis procedures and to provide a standard approach for such analyses. As a result, differences in the manner of performing subcompartment analyses can 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 US Nuclear Regulatory Commission guidelines. A wide range of the effects of input and modeling variations on calculated sacrificial shield-wall (SSW) forces and moments were studied. The studies were for a representative BWR cavity geometry with the pipe break inside the SSW. The COMPARE subcompartment analysis code was used for the studies.

NUREG/CR-2636: EXPERIMENTAL DATA REPORT FOR AIR-WATER FLOODING TESTS OF THE FLECHT-SEASET PROGRAM SET FACILITY VESSEL UPPER PLENUM. ANDERSON, J. L. ; FOGDALL, S. P. EG&G, Inc. June 1982. 99pp. 8206240066. EGG-2183. 13609:007.

A test facility to investgate the flooding characteristics of the FLECHT-SEASET Program's SET Facility vessel upper plenum has been developed and installed in the Steam-Air-Water Test Facility of the Idaho National Engineering Laboratory. A series of countercurrent-flow-limited tests were peformed in the test facility using air-water at low pressure and room temperature. This report documents the experimental system and the testing program, presents tabulations of the data, develops the experimental uncertainty analysis and discusses the results of the testing.

NUREG/CR-2637: EMERGENCY RESPONSE CAPABILITIES AND EXAMPLE ASSESSMENTS FOR AIRBORNE RADIONUCLIDE DISCHARGES. START, G. E. ; CATE, J. H. ; ACKERMANN, G. R. ; et al. Commerce, Dept. of, National Oceanographic & Atmospheric Administration. May 1982. 32pp. 8206100036.

13474:354.

An existing emergency response capability has been developed for the Idaho National Engineering Laboratory by the National Oceanic and Atmospheric Admnistration Air Resources Laboratories Field Research Offices. The system consists of several existing computers and associated data collection facilities. This equipment has been coordinated into a useful capability providing initial and ongoing analysis of meteorological and radiological information. In the event of a radiological emergency, this information may be used to assist action plan formulations and decisions for the area in and around the Snake River Plain in Southeast Idaho.

NUREG/CR-2638: SNOW LOADS FOR THE DESIGN OF NUCLEAR POWER PLANT STRUCTURES. ELLINGWOOD, B. ; HARRIS, J. R. Commerce, Dept. of, National Bureau of Standards. April 1982. 51pp. 8204160038. 12714:015.

This report describes a research program to characterize snow loads on roofs of nuclear power plant structures and to develop recommendations for operating basis and extreme environmental loads. Snow surveys were conducted to gather field data about the distribution of snow on plant roofs and to correlate roof and ground snow loads. The survey data were integrated with data from similar studies to provide recommendations for structural design. Load conbinations involving rain and snow were analyzed probabilistically to provide a basis of comparison with other design basis environmental loads.

NUREG/CR-2639: HISTORICAL EXTREME WINDS FOR THE UNITED STATES--ATLANTIC AND GULF OF MEXICO COASTLINES. CHANGERY, M. J. Commerce, Dept. of, National Oceanographic & Atmospheric Administration. May 1982. 156pp. 8206140312. 13508:239.

Annual fastest mile wind data were extracted for the complete period of record for 53 locations along the Atlantic and Gulf of Mexico coastlines. Existing models were used to standardize the data to 10 meters for airport-type exposures and meters for city exposures. Selected probability estimates were developed from applicaton of the Fisher-Tippet Type I extreme value mode for non-tropical storms and the Weibll model for tropical storms. A mixed distribution was used for locations with a significant percentage of annual extremes caused by tropical storms.

NUREG/CR-2642: LONG-TERM SURVIVABILITY OF RIPRAP FOR ARMORING URANIUM MILL TAILINGS AND COVERS: A LITERATURE REVIEW. LINDSEY, C. G. ; LONG, L. W. ; BEGEJ, C. W. Battelle Memorial Institute, Pacific Northwest Laboratory. June 1982. 140pp. 8207140120. PNL-4225. 13845:347.
Pacific Northwest Laboratory (PNL) is investigating the use of a rock armoring blanket (riprap) to mitigate wind and water erosion of an earthen radon suppression cover applied to uranium mill tailings. Because the radon suppression cover and the tailings must remain intact for up to 1000 years or longer, the riprap must withstand natural weathering forces. This report is a review of information on rock weathering and riprap durability. Chemical and physical weathering processes, rock characteristics related to durability, climatic conditions affecting the degree and rate of weathering, and testing procedures used to measure weathering susceptibilities have been revised. Sampling and testing techniques, as well as analysis of physical and chemical weathering susceptibilities, are necessary to evaluate rock durability. Many potential riprap materials may not be able to survive 1000 years of weathering. Available techniques for

durability testing cannot adequately predict rock durability for the 1000-year period because they do not consider the issue of time (i.e., how long must riprap remain stable). This report includes an Appendix, which discusses rock weathering, written by Dr. Richard Jahns of Stanford University.

NUREG/CR-2644: AN ASSESSMENT OF OFFSITE, REAL-TIME DOSE-MEASUREMENT SYSTEMS FOR EMERGENCY SITUATIONS. MAECK, M. J.; HOFFMAN, L. G.; STAPLES, B. A.; et al. Exxon Nuclear Co., Inc. (subs. of Exxon Corp.). April 1982. 67pp. 8205040015. ENICO-1110. 12970:289.

An evaluation is made of the effectiveness of fixed, real-time monitoring systems around nuclear power stations in determining the magnitude of unmonitored releases. The effects of meteorological conditions on the accuracy with which the magnitude of unmonitored releases is determined and the uncertainties inherent in defining these meterological conditions are discussed. The number and placement of fixed field detectors in a system is discussed, and the data processing equipment required to convert field detector output data into release rate information is described. Cost data relative to the purchase and installation of specific systems are given, as well as the characteristics and information return for a system purchased at an arbitrary cost.

NUREG/CR-2647: CRITICAL HEAT FLUX EXPERIMENTS UNDER LOW FLOW CONDITIONS IN A VERTICAL ANNULUS. MISHIMA, K.; ISHII, M. Argonne National Laboratory. April 1982. 43pp. 8204290604. ANL-82-6. 12895:235.

An experimental study was performed on critical heat flux (CHF) at low flow conditions for low pressure steam-water upward flow in an annulus. The test section was transparent, therefore, visual observations of dryout as well as various instrumentations were made. The data indicated that a premature CHF occurred due to flow regime transition from churn-turbulent to annular flow. It is shown that the critical heat flux observed in the experiment is essentially similar to a flooding-limited burnout and the critical heat flux can be well reproduced by a nondimensional correlation derived from the previously obtained criterion for flow regime transition. The observed CHF values are much smaller than the standard high quality CHF criteria at low flow, corresponding to the annular flow film dryout. This result is very significant, because the coolability of a heater surface at low flow rates can be drastically reduced by the occurrence of this mode of CHF.

NUREG/CR-2648: EXPERIMENTAL DATA REPORT FOR SEMISCALE MOD-2A NATURAL CIRCULATION TEST SERIES (TESTS S-NC-8B AND S-NC-9). SACKETT, K. E.; CLEGG, L. B. EG&G, Inc. April 1982. 56pp. 8205130243. EGG-2184. 13090:028.

This report presents test data recorded for Tests S-NC-8B and S-NC-9 of the Semiscale Mod-2A Natural Circulation Test Series. These tests are part of a series of Semiscale tests that investigate the thermal-hydraulic phenomena resulting from operational transients involving loss of mechanical primary coolant circulation in a pressurized water reactor. The primary objective of Tests S-NC-8B and S-NC-9 was to experimentally characterize the thermal-hydraulic behavior of a system during single-phase, and reflux natural circulation conditions experienced in the course of an integral small break with and without the presence of emergency core cooling water. Of special interest were the effects of single-phase natural circulation flow caused by changes in core power, primary pressure, and external heater power.

This report presents the uninterpretated data from Tests S-NC-8B and S-NC-9 for future analysis. The data, presented as graphs in engineering units, have been analyzed only to the extent necessary to ensure that they are reasonable and consistent.

NUREG/CR-2651: ACCIDENT GENERATED PARTICULATE MATERIALS AND THEIR CHARACTERISITICS--A REVIEW OF BACKGROUND INFORMATION. SUTTER, S. L. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1982. 97pp. 8206170387. PNL-4154. 13557:195.

Safety assessments and environmental impact statements for nuclear fuel cycle facilities require an estimate of the amount of radioactive particulate material initially airborne (source term) during accidents. Pacific Northwest Laboratory (PNL) has surveyed the literature, gathering information on the amount and size of these particles that has been developed from limited experimental work, measurements made from operational accidents, and known aerosol behavior. Information useful for calculating both liquid and powder source terms is compiled in this report. Potential aerosol generating events discussed are spills, resuspension, aerodynamic entrainment, explosions and pressurized releases, communition, and airborne chemical reactions. A discussion of liquid behavior in sprays, sparging, evaporation, and condensation as applied to accident situations is also discussed.

NUREG/CR-2652: EVALUATION AND PERFORMANCE OF CLOSED-CIRCUIT BREATHING APPARATUS. HACK, A. ; TRUJILLO, A. ; CARTER, K. ; et al. Los Alamos Scientific Laboratory. June 1982. 23pp. 8206230328. LA-9266-MS. 13594:295.

Seven closed-circuit self-contained breathing apparatus were worn by a panel of anthropometrically selected test subjects to determine the protection provided by each. The types included those that supply breathing gas continuously, or on demand, or a combination of both. One unit maintained a positive pressure and provided higher protection than the others. Device performance by facial size is discussed.

NUREG/CR-2653: EARTH RESISTIVITY AS A TOOL FOR SHALLOW EXPLORATION IN THE REELFOOT LAKE AREA, TENNESSEE. STEARNS, R. G. ; HASELTON, T. M. ; TSAY, J. Vanderbilt Univ. May 1982. 131pp. 8206100048. 13478:330.

Surface earth resistivity techniques were successfully tested at a shallow (10's of feet) depth in the Reelfoot Lake area of Mississippi's alluvial plain. Profiling, Barnes Layer sections, Wenner sounding, and circle soundings proved useful.

Features of abandoned river channels (a central low resistivity clay 'plug' and lateral high resistivity, sandy natural levees) were readily located and mapped by profiling, and were located within 10 feet or less by circle soundings.

Approximately true resistivity columns were made by measuring the resistivity of samples from small diameter holes. For these columns, Wenner Array soundings gave nearly correct layer thickness estimated in contrast to erroneous Schlumberger soundings.

NUREG/CR-2664: SELECTED REVIEW OF FOREIGN LICENSING PRACTICES FOR NUCLEAR POWER PLANTS. STEVENSON, J. D. ; THOMAS, F. A. Structural Mechanics Associates. April 1982. 151pp. 8205030645. 12928:065.

A compilation and description of current U.S. and foreign licensing and regulatory practices are given. Also included is a brief description of nuclear power plant regulatory and licensing organizations involved. The particular countries surveyed are Canada, France, Japan, Sweden, the United Kingdom, the United States and the Federal Republic of Germany.

NUREG/CR-2671: THE MARVIKEN FULL SCALE CRITICAL FLOW TESTS. Summary Report. * MARVIKEN. May 1982. 26?pp. 8205200278. MXC-301. 13195:014.

The Marviken Full Scale Critical Flow Tests were conducted as a multi-national project at Marviken Power Station in Sweden. The program sought to provide the critical mass flow data necessary to form a link between the available small scale test data and full scale pipe geometries found in operating nuclear power stations.

This report summarizes the program objectives, test facility, instrumentation, procedure, matrix, data and error limits, and significant test results. The summary report is reprinted by USNRC under the multi-national agreement that allows public dissemination of the data two years after the tests.

NUREG/CR-2681: ESTIMATED RECURRENCE FREQUENCIES FOR INITIATING ACCIDENT CATEGORIES ASSOCIATED WITH THE CLINCH RIVER BREEDER REACTOR PLANT DESIGN. COPUS, E.R. Sandia Laboratories. June 1982. 170pp. 8206250048. SAND82-0720. 13627:031.

Estimated recurrence frequencies for each of twenty-five generic LMFBR initiating accident categories were quantified using the Clinch River Breeder Reactor Plant (CRBRP) design. These estimates were obtained using simplified systems fault trees and functional event tree models from the Accident Delineation Study Phase I Final Report coupled with order-of-magnitude estimates for the initiator-dependent failure probabilities of the individual CRBRP engineered safety systems. Twelve distinct protected accident categories where SCRAM is assumed to be successful are estimated to occur at a combined rate of 10(-3) times per year while thirteen unprotected accident categories in which SCRAM fails are estimated to occur at a combined rate on the order of 10(-5) times per year. These estimates are thought to be representative despite the fact that human performance factors, maintenance and repair, as well as input common cause uncertainties, were not treated explicitly. The overall results indicate that for the CRBRP design no single accident category appears to be dominant, nor can any be totally eliminated from further investigation in the areas of accident phenomenology for in-core events and post-accident phenomenology for containment.

NUREG/CR-2682: CITADEL: A COMPUTER CODE FOR THE ANALYSIS OF IODINE BEHAVIOR IN STEAM GENERATOR TUBE RUPTURE ACCIDENTS. RAGHURAM, S.; BAYBUTT, P.; DENNING, R.S.; et al. Battelle Memorial Institute, Columbus Laboratories. April 1982. 141pp. 8205110659. BMI-2093. 13037:001.

The computer code CITADEL was written to analyze iodine behavior during steam generator tube rupture accidents. The code models the transport and deposition of iodine from its point of escape at the steam generator primary break until its release to the environment. This report provides a brief description of the code including its input requirements and the nature and form of its output.

This report is in the form of a user's manual for the code. Only a brief discussion of the processes modeled in the code is provided herein. The interested reader is referred to a companion report for detailed technical description of the models that have been included in the code.

NUREG/CR-2683: IODINE BEHAVIOR IN STEAM GENERATOR TUBE RUPTURE ACCIDENTS. RAGHURAM, S. ; BAYBUTT, P. ; DENNING, R. S. ; et al. Battelle Memorial Institute, Columbus Laboratories. April 1982. 114pp. 8205110662. BMI-2094. 13036:169.

This report identifies the results of a program aimed at developing a computer code for use in the analysis of the behavior of iodine during steam generator tube rupture (SGTR) accidents in pressurized water reactors (PWR's). The program was directed towards the identification of the several processes that play a role in the transport and deposition behavior of iodine from its point of escape at the primary system break to its point of release to the environment, the development of models to describe these processes and the incorporation of these models into a computer code. Preliminary calculations performed using the computer code indicate that iodine contained in the water droplets that are formed as the primary coolant flashes could be a major source of iodine released to the atmosphere during an SGTR accident. Additionally, the assumed chemical form of the iodine, molecular or ionic, appears to be extremely important in determining the consequences of the accident.

NUREG/CR-2685: EVALUATION OF CONCURRENT PEAK RESPONSES. WANG, P. C. ; CURRERI, M. ; SHOOMAN, M. ; et al. Brookhaven National Laboratory. May 1982. 90pp. 8206170046. BNL-NUREG-51529. 13543:001.

This report deals with the problem of combining two or more concurrent responses which were induced by dynamic loads acting on nuclear power plant structures. Specifically, the acceptability of using the SRSS (square root of the sum of the squares) value of peak values as the combined response is investigated. Emphasis is placed on the establishment of a simplified criteron that is convenient and relatively easy to use by design engineers.

NUREG/CR-2686: REVIEW OF LOAD COMBINATIONS FOR NSSS AND BOP PIPING AND EQUIPMENT OF MARK III PLANTS. PHILIPPACOPOULO; REICH, M. ; WANG, P. C. Brookhaven National Laboratory. May 1982. 200pp. 8206100065. BNL-NUREG-51530. 13479:204.

This report describes a review conducted by the Structural Analysis Division of Brookhaven National Laboratory (BNL) for the Mechanical Engineering Branch of the Nuclear Regulatory Commission (MEB/NRC) on combinations of dynamic responses related to Nuclear Steam Supply Systems (NSSS) and Balance-of-Plant (BOP) piping and equipment components of Mark III plants. A total of 167 combination cases were considered. The response combinations reviewed in this report were compiled by Structural Mechanics Associates for the General Electric Company, using time-histories and other technical data supplied by various architect-engineering firms working for the Mark III containment owners. The objective of the (BNL) review was to verify the results presented by SMA.

NUREG/CR-2692: AN INTEGRATED SYSTEM FOR FORECASTING ELECTRIC ENERGY AND LOAD FOR STATES AND UTILITY SERVICE AREAS. CHERN, W. S. ; GALLAGHER, C. A. ; TEPEL, R. C. ; et al. Oak Ridge National Laboratory. May 1982. 58pp. 8206110006. ORNL/TM-7947. 13493:016.

This report documents the integrated system for forecasting electric energy and load. In the system, the service area models of electrical energy (kWh) and the load distribution (minimum and maximum loads and lead duration curve) are linked to the state-level model of electrical energy (kWh). Thus, the service area forecasts are conditional upon the state-level forecasts. Such a linkage reduces considerably the data requirements for modeling service area electricity demand.

Four utilities are selected to provide examples of the integrated forecasting system. The statistical results suggest that the use of selected, important demand determinants, such as price and income, to explain the differences in electricity demand growth between the service area and the remainder of the corresponding state is appropriate. The forecasting results show that the forecasted growth rates of electricity demand, in some cases, differ substantially between the service area and the corresponding state.

NUREG/CR-2696: CALCULATIONS OF TWO SERIES OF EXPERIMENTS PERFORMED AT THE POOLSIDE FACILITY USING THE OAK RIDGE RESEARCH REACTOR. MAERKER, R. E.; WILLIAMS, M. L. Oak Ridge National Laboratory. June 1982. 36pp. 8206240078. ORNL/TM-8326. 13607:244.

This report contains two papers that were presented at the Fourth ASTM-EURATOM Symposium on Reactor Dosimetry in Washington, D.C. on March 22-26, 1982 and serves as documentation of the analytical work performed by the Engineering Physics Division. These papers describe discrete ordinate calculations of two series of experiments that were performed at the Poolside Facility as part of the Surveillance Dosimetry Improvement Program, and are very similar in scope.

NUREG/CR-2699: TRANSPORTATION OF RADIOACTIVE MATERIAL IN MARYLAND. June 1980-June 1981. * Maryland, State of. April 1982. 89pp. 8205110072. 13038:156.

The Maryland Department of Health and Mental Hygiene, under a joint U.S. DOT and NRC contract, conducted a one-year study beginning June 6, 1980 to assess the transport of radioactive materials in Maryland. Highway surveillance indicated that less than one truck in 10,000 was hauling radioactive materials and that Low Specific Activity wastes constituted the primary material being transported. Routing data was developed from surveillance and industry-supplied information. Highway inspection and enforcement activities revealed that the level of transport and violations of radioactive materials indicate a minimum exposure risk to workers. Some violations of labeling and placarding regulations were, however, noted.

NUREG/CR-2700: PARAMETERS FOR CHARACTERIZING SITES FOR DISPOSAL OF LOW-LEVEL RADIOACTIVE WASTE. LUTTON, R. J.; MALONE, P. G.; MEADE, R. B.; et al. Army, Dept. of, Army Engineer Waterways Experiment Station. May 1982. 84pp.. 8206100024. 13475:051.

Sixty-seven site parameters and parameter groups are identified as important characteristics of sites for disposal of low-level radioactive waste and require detailed evaluation. Several of the most important parameters are needed for hydrological analysis while others are needed for facility design, construction, and operation. Still others are needed for baseline and detection stages of monitoring. It is recommended that all parameters be evaluated by technically qualified personnel. Appropriate tests and documentation methods are discussed in a second report, which will follow. However, site-specific testing or elaborate field measurement will not always be necessary, i.e., where indicated to be unnecessary on a technical basis. Much of this report, Appendices A through G, is directed to explaining the importance of parameters and to establishing site-specific limitions.

NUREG/CR-2704: U.S. REACTOR SPENT-FUEL STORAGE CAPABILITIES. LEE, W.J.; HOFFMAN, C.C.; CAVINESS, C.K. Nuclear Assurance Co.p. June 1982. 56pp. 8206290529. 13660:001.

This report describes the spent-fuel storage situation at reactors in the United States. The focus of the report is on the reactors that are developing a spent-fuel storage problem and the alternatives the utilities are utilizing and planning to use to minimize the problem. The alternatives the utilities are using and/or considering are described in the report and include:

High-Density Storage Racks
Double-Tiered Storage Racks
Rod Consolidation
Dry Storage Systems
Fuel Transshipments
At-Reactor Storage Pools

All of these alternatives are not available to every reactor and utility that is faced with a spent-fuel storage problem. Generally, utilities are reracking or are planning to rerack those spent-fuel pools that can be reracked with higher-density racks or double-tiered racks. Where reracking is not feasible, the fuel transshipments are being performed or considered. Since none of these other alternatives have been fully approved and licensed, these alternatives are all being evaluated.

NUREG/CR-2711: PERFORMANCE AND DESIGN REQUIREMENTS FOR A GRAPHICS DISPLAY RESEARCH FACILITY. TILLITT, D.N.; PETERSEN, R.J.; SMITH, R.L. EG&G, Inc. June 1982. 62pp. 8207190007. EGG-2194. 13921:233.

Performance and design requirements for a Graphics Display Research Facility (GDRF) are presented. The GDRF is an evolutionary, computer-based, human-engineering experimentation center that is specifically designed to address long-term research issues associated with automation, human performance, and risk in the operation of nuclear facilities. Research capabilities provided by this facility will directly support the licensing and regulations of nuclear facilities within the United States. This report discusses: the requirements, specifications, and implementation considerations for the facility; the necessary hardware, software, and personnel capabilities; and the potential costs of construction and operation for various levels of research activity. Research provided by this facilitiy is intended to satisfy NRC needs to: (a) confirm design adequacy of, and develop evaluation criteria for computerized graphic display and other information presentation mechanisms proposed for use in nuclear power plants, and (b) assess the possible effects on operator performance of computer-based operator-support concepts. The ultimate goal of this research is to support regulatory directives for minimizing the risk of human error in the operation of nuclear facilities.

NUREG/CR-2713: VAPOR DEPOSITION VELOCITY MEASUREMENTS AND CORRELATIONS FOR I(2)AND CsI. NICOLOSI, S.L.; BAYBUTT, P. Battelle Memorial Institute, Columbus Laboratories. May 1982. 42pp. 8206090117.

BMI-2091. 13457:008.

Vapor deposition velocities were measured for I(2) and CsI vapors depositing on prefilmed Type 304 stainless steel and Inconel 600 surfaces in steam atmospheres. This work was performed to extend the data base of the TRAP (Transport of Radioluclides in Primary systems) code. Arrenhius type vapor deposition velocity correlations were developed for I(2) and for CsI vapors depositing on these materials. The 300-900 C correlation for Inconel 600 is V(d) = 3.49 x 10(-6) exp(3940/RT), and the 300-1130 correlation for Type 304 stainless steel is V(d) = 2.53 x 10(-3) exp(-6670/RT). The I(2) vapor deposition velocity correlation for Inconel 600 should not be used for temperatures greater than 900 C since this correlation gives a decreasing trend with increasing temperature whereas our experiments showed some evidence that the temperature dependence of the vapor deposition velocity for this system may change to an increasing trend at 900 C. The 300-1130 C correlation for Type 304 stainless steel should not be used at lower temperatures since the low temperature vapor deposition velocities decrease with temperature whereas the high temperature vapor deposition velocities increase with temperature for this system. The correlation derived for CsI vapor depositing onto Type 304 stainless steel surfaces is V(d) = 1.65 x 10(-9) exp(21600/RT) for 550-1040 C. The correlation derived for CsI vapor depositing onto Inconel 600 surfaces is V(d) = 6.36 x 10(-8) exp(13670/RT) for 815-1040 C.

NUREG/CR-2717: EXPERIMENT DATA REPORT FOR LOFT ANTICIPATED TRANSIENT WITHOUT SCRAM EXPERIMENT L9-3. BAYLESS, P. D. ; DIVINE, J. M. EG&G, Inc. June 1982. 189pp. 8206240072. EGG-2195. 13609:106.

Selected pertinent and uninterpreted data from the third anticipated transient with multiple failures experiment (Experiment L9-3) conducted in the Loss-of-Fluid Test (LOFT) facility are presented. The LOFT facility is a 50-MW(t) pressurized water reactor (PWR) system with instruments that measure and provide data on the system thermal-hydraulic and nuclear conditions. The operation of the LOFT system is typical of large [approximately 1000 MW(e)], commercial PWR operations. Experiment L9-3 simulated a loss-of-feedwater anticipated transient without scram. The loss-of-feedwater accident led to an increase in the primary coolant system temperature and pressure. Both the experiment power-operated relief valve (PORV) and safety relief valve opened and were able to limit and control the pressure transient. The plant was then recovered with the control rods still withdrawn by injecting 7200-ppm borated water, manually cycling the PORV, and feeding and bleeding the steam generator.

NUREG/CR-2722: RADIOLOGICAL SURVEY OF THE WEST LAKE LANDFILL, ST. LOUIS COUNTY, MISSOURI. BOOTH, L. F. ; GROFF, D. W. ; MCDOWELL, G. S. ; et al. Radiation Management Corp. May 1982. 139pp. 8206100069. 13480:003.

This report presents the results of a radiological survey of the West Lake Landfill, St. Louis County, Missouri, performed by Radiation Management Corporation during the spring and summer of 1981. Measurements were made to determine external radiation levels, concentrations of airborne contaminants and the identity and concentrations of subsurface deposits. Results indicate that large volumes of uranium ore residues, probably originating from the Hazelwood, Missouri, Latty Avenue site, have been buried at the West Lake Landfill. Two areas of contamination, covering more than 15 acres and located at depths of up to 20 feet below the present surface, have been identified. There is no indication that significant quantities of contaminants are moving off-site at this time.

NUREG/CR-2727 V01: ECOLOGICAL STUDIES OF WOOD-BORING BIVALVES IN THE VICINITY OF THE OYSTER CREEK NUCLEAR GENERATING STATION. Progress Report, September-November 1981. HOAGLAND, K. E. ; CROCKET, L. Lehigh Univ. June 1982. 52pp. 8207220666. 14024:214.

The species composition, distribution, and population dynamics of wood-boring bivalves are being studied in the vicinity of the Oyster Creek Nuclear Generating Station, Barnegat Bay, New Jersey. Untreated wood test panels are used to collect organisms at 12 stations. Physiological tolerances of 3 species are also under investigation in the laboratory. Competition among the species is being analyzed. In the fall of 1981, Teredo bartschi remained in Oyster Creek despite continuous prolonged outages of the Oyster Creek Nuclear Generating Station. It did not spread to Forked River or Waretown as it had done in other years when the effluent was present. The peak in larval production and settlement of T. bartschi occurred between September and October. Settlement of shipworms occurred on no monthly panels except those in Oyster Creek during the period of this report. Laboratory experiments revealed that T. bartschi becomes inactive at 5 degrees C (24 parts/thousand) and T. navalis shows signs of osmotic stress below 10 parts/thousand at 18 degrees C. The shipworms in Barnegat Bay do not show a preference for settling at the mudline when the substrate is not limited.

NUREG/CR-2732: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A INTERMEDIATE BREAK TEST SERIES. (Tests S-IB-1 And S-IB-2). SACKETT, K. E. ; CLEGG, L. B. EG&G, Inc. June 1982. 58PP. 8207190010. EGG-2196. 13958:001.

This report presents test data recorded for Tests S-IB-1 and S-IB-2 of the Semiscale Mod-2A Intermediate Break Test Series. These tests are part of a series of Semiscale tests that investigate the thermal-hydraulic phenomena resulting from a hypothesized loss-of-coolant accident (LOCA) in a pressurized water reactor (PWR) system. These tests provide experimental data for assessing the analytical capability of computer codes used in LOCA analysis. Tests S-IB-1 and S-IB-2 were conducted from initial conditions closely approximating the specified initial conditions of: 15.5-MPa system pressure, 557-K cold leg temperature, and 1.95-MW core power level. This report presents uninterpreted data from both tests for future analysis. The data, presented as graphs in engineering units, have been analyzed only to the extent necessary to ensure that they are reasonable and consistent.

NUREG/CR-2736: TRANSPORTATION OF RADIOACTIVE MATERIAL IN MICHIGAN. September 1980-August 1981. MCCARTY, M. J. ; HENNIGAN, J. M. ; BRUCHMANN, G. W. Michigan, State of. May 1982. 98pp. 8206100050. 13479:100.

Most of the radioactive material transported into and through the State of Michigan is comprised of radiopharmaceuticals. The remainder includes rdioactive waste from nuclear power plants and hospitals, uranium ore concentrate (yellowcake) from Ontario, Canada, and periodic spent fuel shipments from a university research reactor. Investigations have revealed that minor violations of packaging and shipping paper regulations persist but to a lesser degree than in previous years. Major operational problems associated with two courier companies have substantially improved but still require improvement. Several minor transportation accidents are reported, none of which resulted in significant radiation exposure. Joint investigations with federal agencies were made, and some resulted in legal action of shippers. Future work performed will be under a contract with the U. S. Department of Transportation.

This report describes the fourth year's study by the state of Michigan of the transportation of radioactive material in Michigan, during the period September 1, 1980 to August 31, 1981. For the periods September 1, 1979 to August 31, 1980 see NUREG/CR-2034; for September 1, 1978 to August 31, 1979 see NUREG/CR-1194; and the first year is unpublished.

NUREG/CR-2737: EVALUATION OF BULK PROPERTIES OF RADWASTE GLASS AND CERAMIC CONTAINER MATERIALS TO DETERMINE LONG-TERM STABILITY. MACEDO, P. B. ; BARKATT, A. Catholic Univ. June 1982. 116pp. 8207140265. 13843:073.

The general objective is to investigate the characteristics of simulated HLW glass and ceramics with respect to surface corrosion, network dissolution and subsequent leaching under an aqueous environment. Based on these characteristics, a model has been proposed to predict the durability to these waste forms. Specific tasks are:

1. Leaching properties under neutral pH with relatively high dilution
2. Wet-dry cycling test
3. Variable pH under high dilution
4. Flow-rate dependency
5. MCC-1 round robin participation.

NUREG/CR-2741: A TECTONIC STUDY OF THE EXTENSION OF THE NEW MADRID FAULT ZONE NEAR ITS INTERSECTION WITH THE 38TH PARALLEL LINEAMENT. Final Technical Report, June 1979-June 1981. BRAILE, L. W. ; HINZE, W. J. ; SEXTON, J. L. ; et al. Purdue Univ. June 1982. 86pp. 8207140097. 13846:227.

Gravity, magnetic, geologic, and seismicity data have been combined in a seismotectonic analysis of the New Madrid seismic zone. Previous studies have presented evidence for several rift zones in this area (Upper Mississippi embayment), including the Reelfoot rift, a late Precambrian-early Paleozoic failed arm which extends north-northeast from the ancient continental margin. We suggest that the northern terminus of the Reelfoot rift forms a rift complex, with arms extending northeast ino southwesten Indiana, northwest along the Mississippi River, and east into western Kentucky, which appears to correlate well with the seismicity in the area. This correlation suggests that faults associated with this rift complex are being reactivated in the contemporary stress field (east-northeast compression). If this interpretation is valid, it represents a seismotectonic model which can be used to predict the extent of future seismicity in the New Madrid seismic zone. The proposed rift complex also provides a coherent model for the tectonic development of this region of the North American midcontinent.

NUREG/CR-2760: ASSESSMENT OF SCALE EFFECTS ON VORTEXING, SWIRL, AND INLET LOSSES IN LARGE SCALE SUMP MODELS. PADMANABHAN, M. ; HECKER, G. E. Alden Research Laboratory. June 1982. 81pp. 8207220641. ARL-48-82. 14024:290.

To verify the use of reduced scale hydraulic models of large scale ratios to demonstrate the performance of containment emergency sumps, a test program involving two geometric scale models (1:2 and 1:4) of a full size sump (1:1) was undertaken as a part of the total test program towards the resolution of Unresolved Safety Issue A-43, "Containment Emergency Sump Performance."

The test results substantiated that hydraulic models of large scale such as 1:2 to 1:4 reliably predicted the sump hydraulic performance. No scale effects on vortexing or air-withdrawals were apparent within the tested prediction range for both models. However, a good prediction of pipe flow swirl and inlet loss coefficient was found to require that the approach flow Reynolds number and pipe Reynolds number be above certain limits.

Based on the results of these tests, it is concluded that properly designed and operated, reduced scale hydraulic models of geometric scales 1:4 or larger can be used both by utilities and by regulatory agencies to prove the satisfactory hydraulic performance of sump designs.

NUREG/CR-2772: HYDRAULIC PERFORMANCE OF PUMP SUCTION INLETS FOR EMERGENCY CORE COOLING SYSTEMS IN BOILING WATER REACTORS. PADMANABHAN, M. Alden Research Laboratory. June 1982. 60pp. 8207220649. ARL-398A. 14024:124.

This document reports on the hydraulic performance of representative Boiling Water Reactor Residual Heat Removal suction inlet configurations; Mark I and Mark II and III designs. Parameters of interest were air-ingestion, vortex types, pipe swirl, and pressure loss coefficients. Tests were conducted with nearly uniform and non-uniform inlet approach flows. Flows and submergences ranged from 2000 to 12000 gpm per pipe and 2 to 5 ft, respectively, giving a Froude number range from 0.17 to 1.06.

Zero air-withdrawal was measured for both configurations for Froude numbers equal to or less than 0.8 even under non-uniform approach flows; no air-core vortices were observed for the same flow conditions. At a Froude number above 1.0 and with non-uniform approach flows, air-withdrawal up to 4% by volume was observed in the Mark I design and air-withdrawals up to 0.5% by volume were observed in the Mark II and III design.

Swirl levels in the pipe up to 7 degrees were measured for Mark II and III designs and up to 3 degrees for Mark I design. Inlet loss coefficients were about 1.7 for Mark II and III designs and about 1.0 for Mark I design.

NUREG/CR-2783: COUNTERCURRENT STEAM-WATER FLOW IN A FLAT PLATE GEOMETRY. BANKOFF, S.G.; KIM, H.J.; TANKIN, R.S.; et al. Northwestern Univ. June 1982. 51pp. 8207220672. NU-8201B. 14024:163.

The study of steam condensation in countercurrent stratified flow of steam and subcooled water has been carried out in a rectangular channel, with an inclination angle 33 degrees from the horizontal. The variables in this experiment were the inlet water and steam flow rates and the inlet water temperature. Condensation heat transfer coefficients were determined as functions of local steam and water flow rates and the degree of subcooling. Correlations are given for the local Nusselt number for the smooth and for the rough surface regimes, and also for the dimensionless wave amplitude. A turbulence-centered model is also considered. It is shown that better agreement with the data can be obtained if the characteristic lengths in the turbulent Nusselt number and turbulent Reynolds number are taken to be wave amplitude and the friction velocity, rather than the water layer thickness and 0.3 times the mean water velocity. A new correlation is presented based on the wave parameters.

NUREG/CR-2788: STRENGTH AND STIFFNESS OF UNIAXIALLY TENSIONED REINFORCED CONCRETE PANELS SUBJECTED TO MEMBRANE SHEAR. HILMY, S. I.; WHITE, R. N.; GERGELY, P. Cornell Univ. June 1982. 223pp. 8207220670. 14020:170.

This report presents experimental and analytical results on internal pressurization effects and seismic shear effects in a concrete containment vessel that is cracked by tension in one direction only. The investigation was a continuation of research reported in NUREG Reports CR-1602 and CR-2049. The experimental program, which was restricted to 6 in. thick flat specimens with two-way reinforcement, included establishment of (a) extensional stiffness for uniaxially tensioned specimens stressed to 0.6f(y), and (b) shear strength and stiffness of these cracked specimens with tension levels ranging from 0 to 0.9f(y); values were about 10 to 15 percent higher than in similar biaxially tensioned specimens. Eleven (11) specimens were tested (6 in monotonic shear and 5 in reversing cyclic shear).

Results are correlated with earlier experimental results from studies on similar specimens and on other simpler specimens that were tested in many different labs (Cornell, PCA, Toronto, Japan, and elsewhere). A finite element representation of behavior is developed for prediction of initial shear modulus. The report concludes with design recommendations.

NUREG/CR-2790: AUTOMOBILE IMPACT FORCES ON CONCRETE WALL PANELS. CHIAPETTA, R. L.; PANG, E. C. Chiapetta, Welch & Associates, Ltd. June 1982. 260pp. 8207060002. CWA 4010-FR. 13716:001.

The objective of this study was to develop force-time impact signature data for use in the design or evaluation of nuclear power plant structures subject to tornado-borne automotive vehicle impact. The approach was based on the use of analytical vehicle models to calculate impact forces. To assess the significance of vehicle/structure interaction for head-on impact force-histories, a lumped-mass model of a reinforced concrete wall panel was coupled to a one-dimensional vehicle model for numerous panel design configurations within the range of practical interest. Vehicle-structure interaction was found to have relatively little effect on the force-histories. The sensitivity of structural response to variations in force signature characteristics was established and idealized impact force-time relations were developed for five distinct impact speeds ranging from 20-60 meters/sec. The use of these relations produce less conservative estimates of structural deflection, for all impact speeds considered, than the currently accepted design procedure.

Contractor Report Number Index

This index lists, in alphabetical order, the contractor-issued report codes for the NRC contractor reports in this compilation. Each contractor code is cross-referenced to the NUREG/CR for the report and to the 10-digit NRC Document Control System accession number.

SECONDARY REPORT NUMBER	REPORT NUMBER	SECONDARY REPORT NUMBER	REPORT NUMBER
ANL-81-29	NUREG/CR-2181 V04	EGG-2196	NUREG/CR-2732
ANL-82-6	NUREG/CR-2647	EGG-CAAD-5629	NUREG/CR-2059
ANL-CT-81-35	NUREG/CR-1622	ENICO-1110	NUREG/CR-2644
ARL-398A	NUREG/CR-2772	EPRI NP-1584	NUREG/CR-2133
ARL-48-82	NUREG/CR-2760	EPRI NP-1781	NUREG/CR-2231
ASA NO. 635	NUREG/CR-2297	EPRI NP-1783	NUREG/CR-2229 V01
BMI-2091	NUREG/CR-2713	ERL-ARL-108	NUREG/CR-2521
BMI-2093	NUREG/CR-2682	GC-TR-82-171	NUREG/CR-2589
BMI-2094	NUREG/CR-2683	GEAP-24939	NUREG/CR-2133
BNL-NUREG-51328	NUREG/CR-1890	GEAP-24962-1	NUREG/CR-2229 V01
BNL-NUREG-51366	NUREG/CR-2039	GEAP-24964	NUREG/CR-2231
BNL-NUREG-51410	NUREG/CR-2193 V01 N2	LA-9116-MS	NUREG/CR-2434
BNL-NUREG-51449	NUREG/CR-2317 V01 N3	LA-9133-MS	NUREG/CR-2464
BNL-NUREG-51449	NUREG/CR-2317 V01 N4	LA-9197-MS	NUREG/CR-2565
BNL-NUREG-51480	NUREG/CR-2416	LA-9209-PR	NUREG/CR-2281 V02
BNL-NUREG-51481	NUREG/CR-2417	LA-9229-PR	NUREG/CR-2281 V03
BNL-NUREG-51499	NUREG/CR-2516 V01 N1	LA-9258-MS	NUREG/CR-2622
BNL-NUREG-51504	NUREG/CR-2542	LA-9266-MS	NUREG/CR-2652
BNL-NUREG-51506	NUREG/CR-2543	LA-9276-SR	NUREG/CR-2632
BNL-NUREG-51529	NUREG/CR-2685	LA-9277-MS	NUREG/CR-2633
BNL-NUREG-51530	NUREG/CR-2686	LA-9301-MS	NUREG/CR-2569
CONF-810372	NUREG/CP-0022	LMF-92	NUREG/CR-2512
CSNI REPT NO. 63	NUREG/CP-0031 V02	MTR-82W26	NUREG/CP-0026
CSNI REPT NO 63	NUREG/CP-0031 V01	MXC-301	NUREG/CR-2671
CWA 4010-FR	NUREG/CR-2790	NU-8201B	NUREG/CR-2783
DPST-80-4	NUREG/CR-1681	ORNL-5822	NUREG/CR-2435
DPST-80-6	NUREG/CR-2356	ORNL-5824	NUREG/CR-2493
EGG-2037	NUREG/CR-0169 V13	ORNL-5846	NUREG/CR-2455
EGG-2070	NUREG/CR-1826 V01	ORNL-5848	NUREG/CR-2456
EGG-2070	NUREG/CR-1826 V02	ORNL/CSD/TM-158	NUREG/CR-2223
EGG-2179	NUREG/CR-2618	ORNL/CSD/TM-160	NUREG/CR-2306
EGG-2183	NUREG/CR-2636	ORNL/CSD/TM-165	NUREG/CR-2494
EGG-2184	NUREG/CR-2648	ORNL/NSIC-182	NUREG/CR-2497 V01
EGG-2194	NUREG/CR-2711	ORNL/NSIC-182	NUREG/CR-2497 V02
EGG-2195	NUREG/CR-2717	ORNL/NSIC-195	NUREG/CR-2172

SECONDARY REPORT NUMBER	REPORT NUMBER	SECONDARY REPORT NUMBER	REPORT NUMBER
ORNL/NSIC-196	NUREG/CR-2173	PNL-4193	NUREG/CR-2564
ORNL/NSIC-196	NUREG/CR-2173 ERR	PNL-4201	NUREG/CR-2600
ORNL/NSIC-200	NUREG/CR-2000 V01 N2	PNL-4225	NUREG/CR-2642
ORNL/NSIC-200	NUREG/CR-2000 V01 N3	PNL-4240	NUREG/CR-2567
ORNL/NSIC-200	NUREG/CR-2000 V01 N4	SAI-288-82-PA	NUREG/CR-1672 V03
ORNL/NSIC-200	NUREG/CR-2000 V01 N5	SAND79-1909	NUREG/CR-1636 V04
ORNL/NUREG-85	NUREG/CR-2469	SAND80-0764	NUREG/CR-1851
ORNL/NUREG/TM-3	NUREG/CR-2220 V03	SAND80-1646	NUREG/CR-1594 V04
ORNL/NUREG/TM-3	NUREG/CR-2220 V02	SAND80-1897 V03	NUREG/CR-1659 V03
ORNL/NUREG/TM-4	NUREG/CR-2053	SAND80-2662	NUREG/CR-1820
ORNL/NUREG/TM-4	NUREG/CR-2525 V02	SAND81-0079	NUREG/CR-2431
ORNL/NUREG/TM-4	NUREG/CR-2525 V01	SAND81-0933	NUREG/CR-2473
ORNL/NUREG/TM-4	NUREG/CR-2525 V03	SAND81-1230	NUREG/CR-2611
ORNL/NUREG/TM-4	NUREG/CR-2525 V05	SAND81-1370	NUREG/CR-2194
ORNL/NUREG/TM-4	NUREG/CR-2525 V07	SAND81-1529 V01	NUREG/CR-2238 V01
ORNL/SUB/80-404	NUREG/CR-2597	SAND81-1622	NUREG/CR-2412
ORNL/TM-7859	NUREG/CR-2544	SAND81-1653	NUREG/CR-2343
ORNL/TM-7868	NUREG/CR-2184	SAND81-1732	NUREG/CR-2279
ORNL/TM-7921	NUREG/CR-2362	SAND81-1754	NUREG/CR-2283
ORNL/TM-7947	NUREG/CR-2692	SAND81-1906	NUREG/CR-2314
ORNL/TM-7985/P2	NUREG/CR-2353 V02	SAND81-1978	NUREG/CR-2350
ORNL/TM-8011	NUREG/CR-2610	SAND81-2655	NUREG/CR-2582
ORNL/TM-8024	NUREG/CR-2470	SAND81-7160	NUREG/CR-2377
ORNL/TM-8073	NUREG/CR-2392	SAND82-0006	NUREG/CR-2481
ORNL/TM-8085	NUREG/CR-2393	SAND82-0319	NUREG/CR-2546
ORNL/TM-8099	NUREG/CR-2612	SAND82-0332	NUREG/CR-2551
ORNL/TM-8172	NUREG/CR-2505	SAND82-0377	NUREG/CR-2559
ORNL/TM-8190	NUREG/CR-2366 V02	SAND82-0410	NUREG/CR-2588
ORNL/TM-8231	NUREG/CR-2204 V04	SAND82-0449	NUREG/CR-2581
ORNL/TM-8236	NUREG/CR-2586	SAND82-0720	NUREG/CR-2681
ORNL/TM-8237	NUREG/CR-2587	SAND82-0927	NUREG/CR-2403 S01
ORNL/TM-8252	NUREG/CR-2141 V04	SAND82-7013	NUREG/CR-2393 ERR
ORNL/TM-8260	NUREG/CR-2221 V04	SAND82-7013	NUREG/CR-2593
ORNL/TM-8274	NUREG/CR-2629	SAND82-7014	NUREG/CR-2394 ERR
ORNL/TM-8307	NUREG/CR-2299 V04	SAND82-7014	NUREG/CR-2594
ORNL/TM-8326	NUREG/CR-2696	SAND82-7017	NUREG/CR-1245 R01
PNL-3117	NUREG/CR-1030 V02	SAND82-7018	NUREG/CR-2604
PNL-3762	NUREG/CR-2019	SAND82-7019	NUREG/CR-2605
PNL-3772	NUREG/CR-2022	SAND82-7063	NUREG/CR-2760
PNL-4028	NUREG/CR-2387	SAND82-7064	NUREG/CR-2772
PNL-4039	NUREG/CR-2201	UCID-18146	NUREG/CR-1233 V04
PNL-4070	NUREG/CR-2432	UCRL-15490	NUREG/CR-2301
PNL-4086	NUREG/CR-2460	UCRL-53021 V04	NUREG/CR-2015 V04
PNL-4106	NUREG/CR-2413		
PNL-4154	NUREG/CR-2651		

Personal Author Index

This index lists the personal authors of NRC staff and contractor reports in alphabetical order. Each name is followed by the NUREG number and the title of the report(s) prepared by that author. If further information is needed, refer to the main citation by the NUREG number.

3

3

NUREG/CR-2494: OR-FLAW: A FINITE ELEMENT PROGRAM FOR DIRECT EVALUATION OF K-FACTORS FOR USER-DEFINED FLAWS IN PLATES, CYLINDERS AND PRESSURE-VESSEL NOZZLE CORNERS.

BATES, E. F.

NUREG-0849: STANDARD REVIEW PLAN FOR THE REVIEW AND EVALUATION OF EMERGENCY PLANS FOR RESEARCH AND TEST REACTORS.

BAYBUTT, P.

NUREG/CR-2682: CITADEL: A COMPUTER CODE FOR THE ANALYSIS OF IODINE BEHAVIOR IN STEAM GENERATOR TUBE RUPTURE ACCIDENTS.

NUREG/CR-2683: IODINE BEHAVIOR IN STEAM GENERATOR TUBE RUPTURE ACCIDENTS.

NUREG/CR-2713: VAPOR DEPOSITION VELOCITY MEASUREMENTS AND CORRELATIONS FOR I(2)AND CsI.

BAYLESS, P. D.

NUREG/CR-2717: EXPERIMENT DATA REPORT FOR LOFT ANTICIPATED TRANSIENT WITHOUT SCRAM EXPERIMENT L9-3.

BECKMAN, R. J.

NUREG/CR-2434: FRAC (FAILURE RATE ANALYSIS CODE): A COMPUTER PROGRAM FOR ANALYSIS OF VARIANCE OF FAILURE RATES. An Application User's Guide.

NUREG/CR-2464: METHODS FOR CLASSIFYING MIXTURES OF EXPONENTIAL DISTRIBUTIONS BASED ON EITHER EXPONENTIAL OR POISSON DATA.

BEEMILLER, R. M.

NUREG/CR-2591: ESTIMATING THE POTENTIAL INDUSTRIAL IMPACTS OF A NUCLEAR REACTOR ACCIDENT.

BEGEJ, C. W.

NUREG/CR-2642: LONG-TERM SURVIVABILITY OF RIPRAP FOR ARMORING URANIUM MILL TAILINGS AND COVERS: A LITERATURE REVIEW.

BELL, J. T.

NUREG/CR-2493: AQUEOUS IODINE CHEMISTRY IN LWR ACCIDENTS: Review And Assessment.

BENNETT, H. A.

NUREG/CR-2546: REACTOR SAFEGUARDS AGAINST INSIDE SABOTAGE.

BERANEK, F.

NUREG/CR-1681: WRAP-PWR VERIFICATION STUDIES.

BERMAN, M.

NUREG/CR-2481: LIGHT WATER REACTOR SAFETY RESEARCH PROGRAM. Semiannual Report, April-September 1981.

BLUHM, D.

NUREG/CR-2442: RELIABILITY ANALYSIS OF STEEL CONTAINMENT STRENGTH. Technical Report, August 1980-September 1981.

BOOTH, L. F.

NUREG/CR-2722: RADIOLOGICAL SURVEY OF THE WEST LAKE LANDFILL, ST. LOUIS COUNTY, MISSOURI.

BOUCHARD, D. A.

NUREG/CR-2314: AGING WITH RESPECT TO FLAMMABILITY AND OTHER PROPERTIES IN FIRE-RETARDED ETHYLENE PROPYLENE RUBBER AND CHLOROSULFONATED POLYETHYLENE.

BRADLEY, D. R.

NUREG/CR-2603: BUBBLE BEHAVIOR IN LMFBR CORE DISRUPTIVE ACCIDENTS.

BRADLEY, E. R.

NUREG/CR-2567: FINAL DATA REPORT FOR THE INSTRUMENTED FUEL ASSEMBLY (IFA)-432.

BRADLEY, O. D.

NUREG/CR-2652: EVALUATION AND PERFORMANCE OF CLOSED-CIRCUIT BREATHING APPARATUS.

BRAILE, L. W.

NUREG/CR-2741: A TECTONIC STUDY OF THE EXTENSION OF THE NEW MADRID FAULT ZONE NEAR ITS INTERSECTION WITH THE 38TH PARALLEL LINEAMENT. Final Technical Report, June 1979-June 1981.

CARUSO, S. C.
NUREG/CR-2133: BWR REFILL-RELOAD PROGRAM TASK 4.4 - 30 SSTF DESCRIPTION DOCUMENT.
CATE, J. H.
NUREG/CR-2637: EMERGENCY RESPONSE CAPABILITIES AND EXAMPLE ASSESSMENTS FOR AIRBORNE RADIONUCLIDE DISCHARGES.
CAVINESS, C. K.
NUREG/CR-2704: U. S. REACTOR SPENT-FUEL STORAGE CAPABILITIES.
CHANGERY, M. J.
NUREG/CR-2639: HISTORICAL EXTREME WINDS FOR THE UNITED STATES-ATLANTIC AND GULF OF MEXICO COASTLINES.
CHAPMAN, R. L.
NUREG/CR-2059: COMPILATION OF DATA CONCERNING KNOWN AND SUSPECTED WATER HAMMER EVENTS IN NUCLEAR POWER PLANTS (CY 1969-MAY 1981).
CHEN, J. C.
NUREG/CR-2015 V04: SEISMIC SAFETY MARGINS RESEARCH PROGRAM PHASE I FINAL REPORT - SOIL STRUCTURE INTERACTION (PROJECT III).
CHEN, N. C.
NUREG/CR-2544: TWO-PHASE MASS FLUX UNCERTAINTY ANALYSIS FOR THERMAL-HYDRAULIC TEST FACILITY INSTRUMENTED SPOOL PIECES.
CHERN, W. S.
NUREG/CR-2692: AN INTEGRATED SYSTEM FOR FORECASTING ELECTRIC ENERGY AND LOAD FOR STATES AND UTILITY SERVICE AREAS.
CHIAPETTA, R. L.
NUREG/CR-2790: AUTOMOBILE IMPACT FORCES ON CONCRETE WALL PANELS.
CHOU, C. K.
NUREG/CR-2301: FRACTURE MECHANICS MODELS DEVELOPED FOR PIPING RELIABILITY ASSESSMENT IN LIGHT WATER REACTORS.
CHOW, H.
NUREG/CR-1826 V01: RELAP5/MOD1 CODE MANUAL. Volume 1: System Models And Numerical Methods.
NUREG/CR-1826 V02: RELAP5/MOD1 CODE MANUAL. Volume 2: User's Guide And Input Requirements.
CHRISTENSEN, D.
NUREG/CR-2059: COMPILATION OF DATA CONCERNING KNOWN AND SUSPECTED WATER HAMMER EVENTS IN NUCLEAR POWER PLANTS (CY 1969-MAY 1981).
CHRISTENSEN, S.
NUREG/CR-2220 V02: THE IMPACT OF ENTRAINMENT AND IMPINGEMENT ON FISH POPULATIONS IN THE HUDSON RIVER ESTUARY.
NUREG/CR-2220 V03: THE IMPACT OF ENTRAINMENT AND IMPINGEMENT ON FISH POPULATIONS IN THE HUDSON RIVER ESTUARY.
CHUN, R. C.
NUREG/CR-2015 V04: SEISMIC SAFETY MARGINS RESEARCH PROGRAM PHASE I FINAL REPORT - SOIL STRUCTURE INTERACTION (PROJECT III).
CLAPP, N. E.
NUREG/CR-2221 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH. Quarterly Progress Report, October 1-December 31, 1981.
CLARK, R. G.
NUREG/CR-2460: TECHNICAL SUPPORT FOR IMPROVING THE LICENSING REGULATORY BASE FOR SELECTED FACILITIES ASSOCIATED WITH THE FRONT END OF THE FUEL CYCLE.
CLEGG, L. B.
NUREG/CR-2648: EXPERIMENTAL DATA REPORT FOR SEMISCALE MOD-2A NATURAL CIRCULATION TEST SERIES (TESTS S-NC-8B AND S-NC-9).
NUREG/CR-2732: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A INTERMEDIATE BREAK TEST SERIES. (Tests S-IB-1 And S-IB-2).
CLEVELAND, J. C.
NUREG/CR-2221 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH. Quarterly Progress

CYBULSKIS, P.
NUREG/CR-1659 V03: REACTOR SAFETY STUDY METHODOLOGY APPLICATIONS PROGRAM: Calvert Cliffs No. 2 PWR Power Plant.
CZYSCINSKI, K. S.
NUREG/CR-2192 V01 N2: EVALUATION OF ISOTOPE MIGRATION-LAND BURIAL. Quarterly Progress Report, April-June 1981.
DAFOE, R. E.
NUREG/CR-2059: COMPILATION OF DATA CONCERNING KNOWN AND SUSPECTED WATER HAMMER EVENTS IN NUCLEAR POWER PLANTS (CY 1969-MAY 1981).
DARKOW, G. L.
NUREG/CR-2359: ATMOSPHERIC STRUCTURE PRIOR TO TORNADOES AS DERIVED FROM PROXIMITY AND PRECEDENT UPPER AIR SOUNDINGS.
DATESMAN, G. H.
NUREG/CR-2297: SECURITY MANAGEMENT TECHNIQUES AND EVALUATIVE CHECKLISTS FOR SECURITY FORCE EFFECTIVENESS.
DAVENPORT, L. C.
NUREG/CR-2460: TECHNICAL SUPPORT FOR IMPROVING THE LICENSING REGULATORY BASE FOR SELECTED FACILITIES ASSOCIATED WITH THE FRONT END OF THE FUEL CYCLE.
DEDHIA, D. D.
NUREG/CR-2301: FRACTURE MECHANICS MODELS DEVELOPED FOR PIPING RELIABILITY ASSESSMENT IN LIGHT WATER REACTORS.
DELENE, J. G.
NUREG/CR-2392: SUMMARY OF ORNL WORK ON NRC-SPONSORED HTGR SAFETY RESEARCH, JULY 1974-SEPTEMBER 1980.
DENNING, R. S.
NUREG/CR-2682: CITADEL: A COMPUTER CODE FOR THE ANALYSIS OF IODINE BEHAVIOR IN STEAM GENERATOR TUBE RUPTURE ACCIDENTS.
NUREG/CR-2683: IODINE BEHAVIOR IN STEAM GENERATOR TUBE RUPTURE ACCIDENTS.
DICKEY, J. M.
NUREG/CR-2542: SENSITIVITY STUDY USING THE FRANTIC CODE FOR THE UNAVAILABILITY OF A SYSTEM TO THE FAILURE CHARACTERISTICS OF THE COMPONENTS AND THE OPERATING CONDITIONS.
DICKSON, C. R.
NUREG/CR-2637: EMERGENCY RESPONSE CAPABILITIES AND EXAMPLE ASSESSMENTS FOR AIRBORNE RADIONUCLIDE DISCHARGES.
DIGGS, B. R.
NUREG/CR-2306: CSRL-V: PROCESSED ENDF/B-V 227-NEUTRON-GROUP AND POINTWISE CROSS-SECTION LIBRARIES FOR CRITICALITY SAFETY, REACTOR AND SHIELDING STUDIES.
DISALVO, R.
NUREG-0863: SURVEY OF FOREIGN REACTOR OPERATOR QUALIFICATIONS, TRAINING, AND STAFFING REQUIREMENTS.
NUREG-0872: A FEASIBILITY STUDY OF USING LICENSEE EVENT REPORTS FOR A STATISTICAL ASSESSMENT OF THE EFFECT OF OVERTIME AND SHIFT WORK ON OPERATOR ERROR.
DIVINE, J. M.
NUREG/CR-2717: EXPERIMENT DATA REPORT FOR LOFT ANTICIPATED TRANSIENT WITHOUT SCRAM EXPERIMENT L9-3.
DOUGAN, J. R.
NUREG/CR-2362: RELATIONSHIPS BETWEEN CHARPY V-NOTCH IMPACT ENERGY AND FRACTURE TOUGHNESS.
DUNN, L. A.
NUREG/CR-2381: GEOLOGIC AND HYDROLOGIC RESEARCH AT THE WESTERN NEW YORK NUCLEAR SERVICE CENTER, WEST VALLEY, NEW YORK. Progress Report, August 1979-July 1981.
ECKERMAN, K. F.
NUREG/CR-2612: VARIABILITY IN DOSE ESTIMATES ASSOCIATED WITH THE FOOD CHAIN TRANSPORT AND INGESTION OF SELECTED RADIONUCLIDES.

ECKERT, R. M.
NUREG/CR-1030 V02: SEDIMENT AND RADIONUCLIDE TRANSPORT IN RIVERS. Phase 2-Field Sampling Program For Cattaraugus And Buttermilk Creeks, New York.
ELLINGWOOD, B.
NUREG/CR-2638: SNOW LOADS FOR THE DESIGN OF NUCLEAR POWER PLANT STRUCTURES.
ELRICK, R. M.
NUREG/CR-1820: STATUS REPORT ON THE FISSION-PRODUCT RESEARCH PROGRAM.
ERDMAN, C. A.
NUREG/CR-2603: BUBBLE BEHAVIOR IN LMFBR CORE DISRUPTIVE ACCIDENTS.
FANOUS, F.
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NUREG/CR-2435: DISPERSED FLOW FILM BOILING IN ROD BUNDLE GEOMETRY-STEADY STATE HEAT TRANSFER DATA AND CORRELATION COMPARISONS.
NUREG/CR-2469: AN ANALYSIS OF TRANSIENT FILM BOILING OF HIGH-PRESSURE WATER IN A ROD BUNDLE.
YOUNG, J. K.
NUREG/CR-2564: ENVIRONMENTAL FACTORS AFFECTING LONG-TERM STABILIZATION OF RADON SUPPRESSION COVERS FOR URANIUM MILL TAILINGS.
YOUNGER, J. M.
NUREG/CR-2591: ESTIMATING THE POTENTIAL INDUSTRIAL IMPACTS OF A NUCLEAR REACTOR ACCIDENT.
YUEN, M. C.
NUREG/CR-2334: INTERPHASE TRANSPORT IN HORIZONTAL STRATIFIED CONCURRENT FLOW.
NUREG/CR-2783: COUNTERCURRENT STEAM-WATER FLOW IN A FLAT PLATE GEOMETRY.
ZABRISKIE, W. L.
NUREG/CR-2204 V04: ADVANCED TWO-PHASE FLOW INSTRUMENTATION PROGRAM. Quarterly Progress Report, October-December 1981.

Subject Index

This index was developed from keywords and word strings in titles and abstracts. During this development period, there will be some redundancy, which will be removed later when a reasonable thesaurus has been developed through experience. Suggestions for improvements are welcome.

NRC Originating Organization Index (Staff Reports)

This index lists those NRC organizations that have published staff reports. The index is arranged alphabetically by major NRC organizations (e.g., program offices) and then by subsections of these (e.g., divisions, branches) where appropriate. Each entry is followed by a NUREG number and title of the report(s). If further information is needed, refer to the main citation by NUREG number.

5

NUREG-0540 V04 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. March 1-31, 1982. (FOIA Supplement)
NUREG-0540 V04 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. March 1-31, 1982.
NUREG-0750 V13 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-June 1981.
NUREG-0750 V14 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-September 1981.
NUREG/CP-0031 V01: PROCEEDINGS OF THE CSNI SPECIALIST MEETING ON OPERATOR TRAINING AND QUALIFICATIONS.
NUREG/CP-0031 V02: PROCEEDINGS OF THE CSNI SPECIALIST MEETING ON OPERATOR TRAINING AND QUALIFICATIONS.

EDO - OFFICE OF MANAGEMENT & PROGRAM ANALYSIS

OFFICE OF MANAGEMENT AND PROGRAM ANALYSIS
NUREG-0020 V06 N02: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of January 31, 1982. (Grey Book)
NUREG-0030 V05 N04: NUCLEAR POWER PLANTS-CONSTRUCTION STATUS REPORT. Data As Of December 31, 1981. (Yellow Book)
NUREG-0871 V01 N02: SUMMARY INFORMATION REPORT. January 1, 1982-March 31, 1982. (Brown Book)

INTERNAL INFORMATION SYSTEMS BRANCH
NUREG-0566 V02 N02: STANDARDS DEVELOPMENT STATUS SUMMARY REPORT. Data As Of February 28, 1982. (Green Book)

EDO - OFFICE OF STATE PROGRAMS

OFFICE OF STATE PROGRAMS, DIRECTOR
NUREG-0891: NUCLEAR PROPERTY INSURANCE: STATUS AND OUTLOOK.

5

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA

DIRECTOR'S OFFICE
NUREG-0090 V04 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. Quarterly Report, October-December 1981.

OFFICE OF INSPECTION & ENFORCEMENT (POST 12/11/80)

DIRECTOR'S OFFICE, OFFICE OF INSPECTION AND ENFORCEMENT
NUREG-0849: STANDARD REVIEW PLAN FOR THE REVIEW AND EVALUATION OF EMERGENCY PLANS FOR RESEARCH AND TEST REACTORS.
NUREG-0903: SURVEY OF INDUSTRY AND GOVERNMENT PROGRAMS TO COMBAT DRUG AND ALCOHOL ABUSE.

OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS

OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS, DIRECTOR
NUREG-0725 R02: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL.
NUREG-0904: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE DECOMMISSIONING OF THE RARE EARTHS FACILITY, WEST CHICAGO, ILLINOIS. Docket No. 40-2061. (Kerr-McGee Chemical Corporation)
NUREG-0923: ADVANCE NOTIFICATION OF SHIPMENTS OF NUCLEAR WASTE AND SPENT FUEL: Guidance.

NUREG-0925: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF THE TETON PROJECT. Docket No. 40-8781. (Teton Exploration Drilling Company, Incorporated)

DIVISION OF WASTE MANAGEMENT

NUREG-0902: SITE SUITABILITY, SELECTION AND CHARACTERIZATION BRANCH TECHNICAL POSITION - Low Level Waste Licensing Branch.

CONSOLIDATION OF THE OFFICES OF NUC. REG. RESEARCH & STANDARDS DEVELOPMEN

OFFICE OF NUCLEAR REGULATORY RESEARCH, DIRECTOR

NUREG-0435 V04 N01: RESEARCH PROJECT CONTROL SYSTEM (RPCS) STATUS SUMMARY REPORT. Research Results Utilization. Data From July 1981-March 1982. (Buff Book)

DIVISION OF FACILITY OPERATIONS

NUREG-0863: SURVEY OF FOREIGN REACTOR OPERATOR QUALIFICATIONS, TRAINING, AND STAFFING REQUIREMENTS.

NUREG-0872: A FEASIBILITY STUDY OF USING LICENSEE EVENT REPORTS FOR A STATISTICAL ASSESSMENT OF THE EFFECT OF OVERTIME AND SHIFT WORK ON OPERATOR ERROR.

EDO-RESOURCE MANAGEMENT

OFFICE OF RESOURCE MANAGEMENT, DIRECTOR

NUREG-0606 V04 N02: UNRESOLVED SAFETY ISSUES SUMMARY. Data As Of May 21, 1982. (Aqua Book)

MANAGEMENT INFORMATION BRANCH

NUREG-0020 V06 N03: LICENSED OPERATING REACTORS. Status Summary Report. February 1982. (Beige Book)

NUREG-0030 V06 N01: NUCLEAR POWER PLANTS-CONSTRUCTION STATUS REPORT. Data as of March 31, 1981. (Yellow Book)

NUREG-0485 V04 N03: SYSTEMATIC EVALUATION PROGRAM STATUS SUMMARY REPORT. Data As Of March 31, 1982. (Buff Book)

NUREG-0485 V04 N04: SYSTEMATIC EVALUATION PROGRAM STATUS SUMMARY REPORT. Data As Of April 30, 1982. (Buff Book)

NUREG-0485 V04 N05: SYSTEMATIC EVALUATION PROGRAM STATUS SUMMARY REPORT. Data As Of May 31, 1982. (Buff Book)

NUREG-0580 V11 N01-4: DRAFT REGULATORY LICENSING STATUS SUMMARY REPORT. Data as of January 1 - April 19, 1982. (Blue Book)

NUREG-0580 V11 N05: REGULATORY LICENSING STATUS SUMMARY REPORT. Data As Of May 15, 1982. (Blue Book)

NUREG-0748 V02 N03: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of March 31, 1982. (Orange Book)

NUREG-0748 V02 N04: OPERATING REACTOR LICENSING ACTIONS SUMMARY. Data As Of April 30, 1982. (Orange Book)

NUREG-0748 V02 N05: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of May 31, 1982. (Orange Book)

OFFICE OF NUCLEAR REACTOR REGULATION (POST 4/28/80)

OFFICE OF NUCLEAR REACTOR REGULATION, DIRECTOR

NUREG-0519 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF LASALLE COUNTY STATION, UNITS 1 AND 2. Docket Nos. 50-373 And 50-374. (Commonwealth Edison Company, et al.)

NUREG-0712 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 1 & 2. Docket Nos.

50-361 And 50-362. (Southern California Edison Company)

NUREG-0787 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATERFORD STEAM ELECTRIC STATION, UNIT NO. 3. Docket No. 50-382. (Louisiana Power & Light Company)

NUREG-0793: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF MIDLAND PLANT, UNITS 1 AND 2. Docket Nos. 50-329 And 50-330. (Consumers Power Company)

NUREG-0793 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF MIDLAND PLANT, UNITS 1 AND 2. Docket Nos 50-329 And 50-330. (Consumers Power Company)

NUREG-0831 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF GRAND GULF NUCLEAR STATION, UNITS 1 & 2. Docket Nos. 50-416 & 50-417. (Mississippi Power And Light Company)

NUREG-0842: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF ST. LUCIE PLANT, UNIT NO. 2. Docket No. 50-389. (Florida Power & Light Company)

NUREG-0847: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF THE WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)

NUREG-0848: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF BYRON STATION UNITS 1 AND 2. Docket Nos. STN 50-454 And STN 50-455. (Commonwealth Edison Company)

NUREG-0854: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF CLINTON POWER STATION, UNIT NO. 1. Docket No. 50-461. (Illinois Power Company, et al.)

NUREG-0857 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2 & 3. Docket Nos. STN 50-528, STN 50-529 & STN 50-530. (Arizona Public Service Company)

NUREG-0861: TECHNICAL SPECIFICATIONS FOR LA SALLE COUNTY STATION, UNIT NO. 1. Docket No. 50-373 (Commonwealth Edison Company).

NUREG-0878: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF WOLF CREEK GENERATING STATION, UNIT 1. Docket No. STN 50-482. (Kansas Gas And Electric Company, et al.)

NUREG-0881: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WOLF CREEK GENERATING STATION, UNIT NO. 1. Docket No. STN 50-482. (Kansas Gas And Electric Company, et al.)

NUREG-0887: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PERRY NUCLEAR POWER PLANTS, UNITS 1 & 2. Docket Nos. 50-440 And 50-441. (Cleveland Electric Illuminating Company)

NUREG-0894: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE CONSTRUCTION OF SKAGIT/HANFORD NUCLEAR PROJECT, UNITS 1 AND 2. Docket Nos. STN 50-522 And STN 50-523. (Puget Sound Power And Light Company, Pacific Power And Light Company, et al.)

NUREG-0895: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF SEABROOK STATION, UNITS 1 AND 2. Docket Nos. 50-443 & 50-444. (Public Service Company Of New Hampshire, et al.)

NUREG-0913: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE RESEARCH REACTOR AT THE UNIVERSITY OF FLORIDA. Docket No. 50-83.

NUREG-0916: SAFETY EVALUATION REPORT RELATED TO RESTART OF R.E. GINNA NUCLEAR POWER PLANT. Docket No. 50-244. (Rochester Gas And Electric Corporation)

NUREG-0916 ERR: SAFETY EVALUATION REPORT RELATED TO THE RESTART OF R.E. GINNA NUCLEAR POWER PLANT. Docket No 50-244. (Rochester Gas And Electric Corporation)

CLINCH RIVER BREEDER REACTOR PROGRAM OFFICE

NUREG-0786 R01: SITE-SUITABILITY REPORT IN THE MATTER OF THE CLINCH RIVER BREEDER REACTOR PLANT. Docket No. 50-537.

DIVISION OF ENGINEERING

NUREG-0868: A COLLECTION OF MATHEMATICAL MODELS FOR DISPERSION IN SURFACE WATER AND GROUNDWATER.

DIVISION OF LICENSING

NUREG-0820 DRFT: INTEGRATED PLANT SAFETY ASSESSMENT SYSTEMATIC EVALUATION PROGRAM FOR PALISADES PLANT. Docket No. 50-255. (Consumers Power Company)

NUREG-0821 DRFT: INTEGRATED PLANT SAFETY ASSESSMENT SYSTEMATIC EVALUATION PROGRAM FOR R. E. GINNA NUCLEAR POWER PLANT. Docket No. 50-244. (Rochester Gas & Electric Corporation)

NUREG-0911: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE WASHINGTON STATE UNIVERSITY TRIGA REACTOR. Docket No. 50-27.

DIVISION OF SAFETY TECHNOLOGY

NUREG-0926: TECHNICAL SPECIFICATIONS FOR GRAND GULF NUCLEAR STATION, UNIT NO. 1. Docket No. 50-416. (Mississippi Power and Light Company)

NRC Contract Sponsor Index (Contractor Reports)

This index lists the NRC organizations that sponsored the contractor reports listed in this compilation. It is arranged alphabetically by major NRC organization (e.g., program office) and then by subsections of these (e.g., divisions) where appropriate. The sponsor organization is followed by the NUREG/CR number and title of the report(s) prepared by that organization. If further information is needed, refer to the main citation by the NUREG/CR number.

NUREG/CR-2722: RADIOLOGICAL SURVEY OF THE WEST LAKE LANDFILL, ST. LOUIS COUNTY, MISSOURI.
DIVISION OF WASTE MANAGEMENT
NUREG/CR-2350: SENSITIVITY ANALYSIS TECHNIQUES: SELF-TEACHING CURRICULUM.
NUREG/CR-2700: PARAMETERS FOR CHARACTERIZING SITES FOR DISPOSAL OF LOW-LEVEL RADIOACTIVE WASTE.

CONSOLIDATION OF THE OFFICES OF NUC. REG. RESEARCH & STANDARDS DEVELOPMEN

OFFICE OF NUCLEAR REGULATORY RESEARCH, DIRECTOR
NUREG/CR-0169 V13: LOFT EXPERIMENTAL MEASUREMENTS UNCERTAINTY ANALYSES. Volume XIII. Temperature Measurements.
NUREG/CR-1245 R01: CORRECTIONS AND ADDITIONS TO USER'S GUIDE FOR SNAP. (NUREG/CR-1245, SAND80-0315).
NUREG/CR-1594 V04: ADVANCED REACTOR SAFETY RESEARCH QUARTERLY REPORT OCTOBER-DECEMBER 1980.
NUREG/CR-1622: FLOW MEASUREMENT BY PULSED-NEUTRON ACTIVATION TECHNIQUES AT THE PKL FACILITY AT ERLANGEN (GERMANY).
NUREG/CR-1659 V03: REACTOR SAFETY STUDY METHODOLOGY APPLICATIONS PROGRAM: Calvert Cliffs No. 2 PWR Power Plant.
NUREG/CR-1681: WRAP-PWR VERIFICATION STUDIES.
NUREG/CR-1820: STATUS REPORT ON THE FISSION-PRODUCT RESEARCH PROGRAM.
NUREG/CR-1826 V01: RELAP5/MOD1 CODE MANUAL. Volume 1: System Models And Numerical Methods.
NUREG/CR-1826 V02: RELAP5/MOD1 CODE MANUAL. Volume 2: User's Guide And Input Requirements.
NUREG/CR-1851: REACTOR PHYSICS DESIGN CALCULATIONS FOR THE ACPR UPGRADE.
NUREG/CR-2053: HEAT TRANSFER ANALYSIS OF THE LWR PRESSURE VESSEL STEEL IRRADIATION CAPSULES IN THE OAK RIDGE RESEARCH REACTOR-PRESSURE VESSEL BENCHMARK FACILITY.
NUREG/CR-2059: COMPILATION OF DATA CONCERNING KNOWN AND SUSPECTED WATER HAMMER EVENTS IN NUCLEAR POWER PLANTS (CY 1969-MAY 1981).
NUREG/CR-2141 V04: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM. Quarterly Progress Report For October-December 1981.
NUREG/CR-2172: SUMMARY AND BIBLIOGRAPHY OF SAFETY-RELATED EVENTS AT BOILING-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1980.
NUREG/CR-2172 ERR: SUMMARY AND BIBIOLOGRAPHY OF SAFETY-RELATED EVENTS AT BOILING-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1980.
NUREG/CR-2173: SUMMARY AND BIBLIOGRAPHY OF SAFETY-RELATED EVENTS AT PRESSURIZED-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1980.
NUREG/CR-2173 ERR: SUMMARY AND BIBLIOGRAPHY OF SAFETY-RELATED EVENTS AT PRESSURIZED-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1980.
NUREG/CR-2181 V04: PHYSICS OF REACTOR SAFETY. Quarterly Report, October-December 1981.
NUREG/CR-2192 V01 N2: EVALUATION OF ISOTOPE MIGRATION-LAND BURIAL. Quarterly Progress Report, April-June 1981.
NUREG/CR-2193 V01 N2: PROPERTIES OF RADIOACTIVE WASTES AND WASTE CONTAINERS. Quarterly Progress Report, April-June 1981.
NUREG/CR-2194: CONTAINMENT RESEARCH PRIORITIES.
NUREG/CR-2204 V04: ADVANCED TWO-PHASE FLOW INSTRUMENTATION PROGRAM. Quarterly Progress Report, October-December 1981.
NUREG/CR-2220 V02: THE IMPACT OF ENTRAINMENT AND IMPINGEMENT ON FISH POPULATIONS IN THE HUDSON RIVER ESTUARY.
NUREG/CR-2220 V03: THE IMPACT OF ENTRAINMENT AND IMPINGEMENT ON FISH POPULATIONS IN THE HUDSON RIVER ESTUARY.
NUREG/CR-2221 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH. Quarterly Progress

6

Report, October 1-December 31, 1981.
NUREG/CR-2231: BWR LOW FLOW BUNDLE UNCOVERY TEST AND ANALYSIS.
NUREG/CR-2238 V01: ADVANCED REACTOR SAFETY RESEARCH. Quarterly Report, January-March 1981.
NUREG/CR-2279: WATER RELEASE FROM HEATED CONCRETES.
NUREG/CR-2281 V02: NUCLEAR REACTOR SAFETY. April 1-June 30, 1981.
NUREG/CR-2281 V03: NUCLEAR REACTOR SAFETY. July 1-September 30, 1981.
NUREG/CR-2283: DIRECT OBSERVATION OF MELT BEHAVIOR DURING HIGH TEMPERATURE MELT/CONCRETE INTERACTIONS.
NUREG/CR-2299 V04: AEROSOL RELEASE AND TRANSPORT PROGRAM. Quarterly Progress Report For October-December 1981.
NUREG/CR-2300 V01 R1: DRAFT: PRA PROCEDURE GUIDE. A Guide To The Performance Of Probabilistic Risk Assessments For Nuclear Power Plants.
NUREG/CR-2306: CSRL-V: PROCESSED ENDF/B-V 227-NEUTRON-GROUP AND POINTWISE CROSS-SECTION LIBRARIES FOR CRITICALITY SAFETY, REACTOR AND SHIELDING STUDIES.
NUREG/CR-2314: AGING WITH RESPECT TO FLAMMABILITY AND OTHER PROPERTIES IN FIRE-RETARDED ETHYLENE PROPYLENE RUBBER AND CHLOROSULFONATED POLYETHYLENE.
NUREG/CR-2317 V01 N3: CONTAINER ASSESSMENT-CORROSION STUDY OF HLW CONTAINER MATERIALS. Quarterly Progress Report, July-September 1981.
NUREG/CR-2317 V01 N4: CONTAINER ASSESSMENT-CORROSION STUDY OF HLW CONTAINER MATERIALS. Quarterly Progress Report, October-December 1981.
NUREG/CR-2353 V02: SPECIFICATION AND VERIFICATION OF NUCLEAR POWER PLANT TRAINING SIMULATOR RESPONSE CHARACTERISTICS. Part II: Conclusions And Recommendations.
NUREG/CR-2356: UPDATED INPUT FOR THE WRAP-EM SYSTEM.
NUREG/CR-2362: RELATIONSHIPS BETWEEN CHARPY V-NOTCH IMPACT ENERGY AND FRACTURE TOUGHNESS.
NUREG/CR-2366 V02: MULTIROD BURST TEST PROGRAM PROGRESS REPORT FOR JULY-DECEMBER 1981.
NUREG/CR-2377: TESTS & CRITERIA FOR FIRE PROTECTION OF CABLE PENETRATIONS.
NUREG/CR-2392: SUMMARY OF ORNL WORK ON NRC-SPONSORED HTGR SAFETY RESEARCH, JULY 1974-SEPTEMBER 1980.
NUREG/CR-2393: FUEL AEROSOL SIMULANT TEST DATA RECORD REPORT: UNDERWATER TESTS.
NUREG/CR-2393 ERR: Errata, changing rept number to NUREG/CR-2593, to A USER'S MANUAL FOR COMPUTER CODE RIBD/IRT.
NUREG/CR-2394 ERR: Errata, changing rept number to NUREG/CR-2594, to A USER'S MANUAL FOR THE GABAS SPECTRUM COMPUTER CODE.
NUREG/CR-2412: HEAT REMOVAL FROM A STRATIFIED UO2-SODIUM PARTICLE BED.
NUREG/CR-2416: INITIAL QUANTIFICATION OF HUMAN ERROR ASSOCIATED WITH SPECIFIC INSTRUMENTATION AND CONTROL SYSTEM COMPONENTS IN LICENSED NUCLEAR POWER PLANTS.
NUREG/CR-2417: IDENTIFICATION AND ANALYSIS OF HUMAN ERRORS UNDERLYING PUMP AND VALVE RELATED EVENTS REPORTED BY NUCLEAR POWER PLANT LICENSEES.
NUREG/CR-2431: BURN MODE ANALYSIS OF HORIZONTAL CABLE TRAY FIRES.
NUREG/CR-2432: A UNIQUE CONCEPT FOR LIQUID LEVEL AND VOID FRACTION DETECTION IN SEVERE FUEL DAMAGE TESTS.
NUREG/CR-2434: FRAC (FAILURE RATE ANALYSIS CODE): A COMPUTER PROGRAM FOR ANALYSIS OF VARIANCE OF FAILURE RATES. An Application User's Guide.
NUREG/CR-2435: DISPERSED FLOW FILM BOILING IN ROD BUNDLE GEOMETRY-STEADY STATE HEAT TRANSFER DATA AND CORRELATION COMPARISONS.

NUREG/CR-2455: EXPERIMENTAL INVESTIGATIONS OF BUNDLE BOILOFF AND REFLOOD UNDER HIGH-PRESSURE LOW HEAT FLUX CONDITIONS.

NUREG/CR-2456: EXPERIMENTAL INVESTIGATIONS OF UNCOVERED-BUNDLE HEAT TRANSFER AND TWO-PHASE MIXTURE-LEVEL SWELL UNDER HIGH-PRESSURE LOW HEAT-FLUX CONDITIONS.

NUREG/CR-2464: METHODS FOR CLASSIFYING MIXTURES OF EXPONENTIAL DISTRIBUTIONS BASED ON EITHER EXPONENTIAL OR POISSON DATA.

NUREG/CR-2469: AN ANALYSIS OF TRANSIENT FILM BOILING OF HIGH-PRESSURE WATER IN A ROD BUNDLE.

NUREG/CR-2470: THERMOMETRY IN THE MULTIROD BURST TEST PROGRAM.

NUREG/CR-2473: SIMMER ANALYSIS OF PROMPT BURST ENERGETICS EXPERIMENTS.

NUREG/CR-2481: LIGHT WATER REACTOR SAFETY RESEARCH PROGRAM. Semiannual Report, April-September 1981.

NUREG/CR-2493: AQUEOUS IODINE CHEMISTRY IN LWR ACCIDENTS: Review And Assessment.

NUREG/CR-2494: OR-FLAW: A FINITE ELEMENT PROGRAM FOR DIRECT EVALUATION OF K-FACTORS FOR USER-DEFINED FLAWS IN PLATES, CYLINDERS AND PRESSURE-VESSEL NOZZLE CORNERS.

NUREG/CR-2496: HUMAN ENGINEERING DESIGN CONSIDERATIONS FOR CATHODE RAY TUBE-GENERATED DISPLAYS.

NUREG/CR-2505: ELECTRICAL IMPEDANCE STRING PROBES FOR TWO-PHASE VOIDS AND VELOCITY MEASUREMENTS.

NUREG/CR-2516 V01 N1: CHARACTERIZATION OF TMI-TYPE WASTES AND SOLID PRODUCTS. Quarterly Progress Report, April-September 1981.

NUREG/CR-2525 V01: ORNL ROD BUNDLE HEAT TRANSFER TEST DATA. Volume 1-ORNL Small Break LOCA Test Series I: Experimental Data Report.

NUREG/CR-2525 V03: ORNL ROD BUNDLE HEAT TRANSFER TEST DATA. Volume 3-Thermal-Hydraulic Test Facility Experimental Data Report For Test 3.06.6B-Transient Film Boiling In Upflow.

NUREG/CR-2525 V05: ORNL ROD BUNDLE HEAT TRANSFER TEST DATA. Volume 5-Thermal-Hydraulic Test Facility Experimental Data Report For Test 3.08.6C-Transient Film Boiling In Upflow.

NUREG/CR-2525 V07: ORNL ROD BUNDLE HEAT TRANSFER TEST DATA. Volume 7-Thermal-Hydraulic Test Facility Experimental Data Support For Test Series 3.07.9-Steady-State Film Boiling In Upflow.

NUREG/CR-2542: SENSITIVITY STUDY USING THE FRANTIC CODE FOR THE UNAVAILABILITY OF A SYSTEM TO THE FAILURE CHARACTERISTICS OF THE COMPONENTS AND THE OPERATING CONDITIONS.

NUREG/CR-2543: A STUDY OF THE FEASIBILITY OF MICROWAVE DIELECTRIC HEATING FOR LMFBR TRANSITION PHASE ACCIDENT SEQUENCE BOILING STUDIES.

NUREG/CR-2544: TWO-PHASE MASS FLUX UNCERTAINTY ANALYSIS FOR THERMAL-HYDRAULIC TEST FACILITY INSTRUMENTED SPOOL PIECES.

NUREG/CR-2545: DESIGN CONCEPT AND TESTING OF AN IN-BUNDLE GAMMA DENSITOMETER FOR SUBCHANNEL VOID FRACTION MEASUREMENTS IN THE THTF ELECTRICALLY HEATED ROD BUNDLE.

NUREG/CR-2546: REACTOR SAFEGUARDS AGAINST INSIDE SABOTAGE.

NUREG/CR-2551: RANK ORDERING OF VITAL AREAS WITHIN NUCLEAR POWER PLANTS.

NUREG/CR-2559: RESULTS OF PHASE ONE OF PLANT ELECTRICAL SYSTEM (PES) STUDY.

NUREG/CR-2565: STRUCTURAL PERFORMANCE OF HEPA FILTERS UNDER SIMULATED TORNADO CONDITIONS.

NUREG/CR-2581: SOME EFFECTS OF ELECTRONS SLOWING DOWN IN MATERIALS WITH APPLICATION TO SAFETY-RELATED EQUIPMENT QUALIFICATION.

NUREG/CR-2582: RADIATION CAPABILITIES OF THE SANDIA HIGH INTENSITY ADJUSTABLE COBALT ARRAY.

NUREG/CR-2586: A SURVEY OF METHODS FOR IMPROVING OPERATOR ACCEPTANCE OF COMPUTERIZED AIDS.

NUREG/CR-2587: FUNCTIONS AND OPERATIONS OF NUCLEAR POWER PLANT CREWS.
NUREG/CR-2588: SECURITY OFFICER RESPONSE STRATEGIES (SECURORS).
NUREG/CR-2593: A USER'S MANUAL FOR COMPUTER CODE RIBD/IRT.
NUREG/CR-2594: A USER'S MANUAL FOR THE GABAS SPECTRUM COMPUTER CODE.
NUREG/CR-2597: STEADY-STATE PRESSURE LOSSES FOR MULTIROD BURST TEST (MRBT) BUNDLE B-5.
NUREG/CR-2604: THE SNAP OPERATING SYSTEM (SOS) USER'S GUIDE.
NUREG/CR-2605: THE SNAP OPERATING SYSTEM REFERENCE MANUAL.
NUREG/CR-2610: RAGBEEF: A FORTRAN IV IMPLEMENTATION OF A TIME-DEPENDENT MODEL FOR RADIONUCLIDE CONTAMINATION OF BEEF.
NUREG/CR-2611: MGO AND 70 W% UO2-30W% Y2O3: THERMOPHYSICAL AND TRANSIENT PROPERTIES.
NUREG/CR-2618: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A NATURAL CIRCULATION TEST S-NC-7C.
NUREG/CR-2622: ANALYSIS OF TRAC AND SCTF RESULTS FOR SYSTEM PRESSURE-EFFECTS TESTS UNDER FORCED FLOODING (RUNS 506, 507 AND 508).
NUREG/CR-2632: RESPONSE OF CENTRIFUGAL BLOWERS TO SIMULATED TORNADO TRANSIENTS. July-September 1981.
NUREG/CR-2636: EXPERIMENTAL DATA REPORT FOR AIR-WATER FLOODING TESTS OF THE FLECHT-SEASET PROGRAM SET FACILITY VESSEL UPPER PLENUM.
NUREG/CR-2647: CRITICAL HEAT FLUX EXPERIMENTS UNDER LOW FLOW CONDITIONS IN A VERTICAL ANNULUS.
NUREG/CR-2649: EXPERIMENTAL DATA REPORT FOR SEMISCALE MOD-2A NATURAL CIRCULATION TEST SERIES (TESTS S-NC-8B AND S-NC-9).
NUREG/CR-2652: EVALUATION AND PERFORMANCE OF CLOSED-CIRCUIT BREATHING APPARATUS.
NUREG/CR-2671: THE MARVIKEN FULL SCALE CRITICAL FLOW TESTS. Summary Report.
NUREG/CR-2681: ESTIMATED RECURRENCE FREQUENCIES FOR INITIATING ACCIDENT CATEGORIES ASSOCIATED WITH THE CLINCH RIVER BREEDER REACTOR PLANT DESIGN.
NUREG/CR-2692: AN INTEGRATED SYSTEM FOR FORECASTING ELECTRIC ENERGY AND LOAD FOR STATES AND UTILITY SERVICE AREAS.
NUREG/CR-2696: CALCULATIONS OF TWO SERIES OF EXPERIMENTS PERFORMED AT THE POOLSIDE FACILITY USING THE OAK RIDGE RESEARCH REACTOR.
NUREG/CR-2711: PERFORMANCE AND DESIGN REQUIREMENTS FOR A GRAPHICS DISPLAY RESEARCH FACILITY.
NUREG/CR-2717: EXPERIMENT DATA REPORT FOR LOFT ANTICIPATED TRANSIENT WITHOUT SCRAM EXPERIMENT L9-3.
NUREG/CR-2732: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A INTERMEDIATE BREAK TEST SERIES. (Tests S-IB-1 And S-IB-2).

DIVISION OF ACCIDENT EVALUATION

NUREG/CR-2133: BWR REFILL-RELOAD PROGRAM TASK 4.4 - 30 SSTF DESCRIPTION DOCUMENT.
NUREG/CR-2229 V01: BWR LARGE BREAK SIMULATION TESTS--BWR BLOWDOWN/EMERGENCY CORE COOLING PROGRAM.
NUREG/CR-2518: THERMODYNAMIC PROPERTIES OF WATER FOR COMPUTER SIMULATION OF POWER PLANTS.
NUREG/CR-2567: FINAL DATA REPORT FOR THE INSTRUMENTED FUEL ASSEMBLY (IFA)-432.
NUREG/CR-2600: END-OF-IRRADIATION DATA REPORT FOR THE INSTRUMENTED FUEL ASSEMBLY (IFA)-527.
NUREG/CR-2603: BUBBLE BEHAVIOR IN LMFBR CORE DISRUPTIVE ACCIDENTS.
NUREG/CR-2682: CITADEL: A COMPUTER CODE FOR THE ANALYSIS OF IODINE BEHAVIOR IN STEAM GENERATOR TUBE RUPTURE ACCIDENTS.
NUREG/CR-2683: IODINE BEHAVIOR IN STEAM GENERATOR TUBE RUPTURE ACCIDENTS.
NUREG/CR-2685: EVALUATION OF CONCURRENT PEAK RESPONSES
NUREG/CR-2713: VAPOR DEPOSITION VELOCITY MEASUREMENTS AND

CORRELATIONS FOR I(2)AND CsI.

NUREG/CR-2783: COUNTERCURRENT STEAM-WATER FLOW IN A FLAT PLATE GEOMETRY.

DIVISION OF FACILITY OPERATIONS

NUREG/CR-2019: THIRD PHASE OF POCKET-SIZED ELECTRONIC DOSIMETER TESTING.

NUREG/CR-2297: SECURITY MANAGEMENT TECHNIQUES AND EVALUATIVE CHECKLISTS FOR SECURITY FORCE EFFECTIVENESS.

DIVISION OF HEALTH, SITING & ENVIRONMENT

NUREG/CR-2343: RISK METHODOLOGY FOR GEOLOGIC DISPOSAL OF RADIOACTIVE WASTE: THE DNET COMPUTER CODE USER'S MANUAL.

NUREG/CR-2359: ATMOSPHERIC STRUCTURE PRIOR TO TORNADOES AS DERIVED FROM PROXIMITY AND PRECEDENT UPPER AIR SOUNDINGS.

NUREG/CR-2381: GEOLOGIC AND HYDROLOGIC RESEARCH AT THE WESTERN NEW YORK NUCLEAR SERVICE CENTER, WEST VALLEY, NEW YORK. Progress Report, August 1979-July 1981.

NUREG/CR-2521: METHOD FOR ESTIMATING WAKE FLOW AND EFFLUENT DISPERSION NEAR SIMPLE BLOCK-LIKE BUILDINGS.

NUREG/CR-2564: ENVIRONMENTAL FACTORS AFFECTING LONG-TERM STABILIZATION OF RADON SUPPRESSION COVERS FOR URANIUM MILL TAILINGS.

NUREG/CR-2584: METEROLOGICAL CONSIDERATIONS IN THE DEVELOPMENT OF A REAL-TIME ATMOSPHERIC DISPERSION MODEL FOR REACTOR EFFLUENT EXPOSURE PATHWAY.

NUREG/CR-2589: A GROUND-PENETRATING RADAR SURVEY OF THE MAXEY FLATS LOW-LEVEL NUCLEAR WASTE DISPOSAL SITE, FLEMING COUNTY, KENTUCKY.

NUREG/CR-2625: CRITICAL PATHWAYS OF RADIONUCLIDES TO MAN FROM AGRO-ECOSYSTEMS. Annual Progress Report, October 1980-September 1981.

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NUREG/CR-2638: SNOW LOADS FOR THE DESIGN OF NUCLEAR POWER PLANT STRUCTURES.

NUREG/CR-2639: HISTORICAL EXTREME WINDS FOR THE UNITED STATES-ATLANTIC AND GULF OF MEXICO COASTLINES.

NUREG/CR-2642: LONG-TERM SURVIVABILITY OF RIPRAP FOR ARMORING URANIUM MILL TAILINGS AND COVERS: A LITERATURE REVIEW.

NUREG/CR-2653: EARTH RESISTIVITY AS A TOOL FOR SHALLOW EXPLORATION IN THE REELFOOT LAKE AREA, TENNESSEE.

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DIVISION OF RISK ANALYSIS

NUREG/CR-1672 V03: RISK ASSESSMENT METHODOLOGY DEVELOPMENT FOR WASTE ISOLATION IN GEOLOGIC MEDIA: Technical Review of NUREG/CR-1636, Vols 1, 2 and 3, December 1, 1981-March 31, 1982.

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NUREG/CR-2651: ACCIDENT GENERATED PARTICULATE MATERIALS AND THEIR CHARACTERISITICS--A REVIEW OF BACKGROUND INFORMATION.

DIVISION OF ENGINEERING TECHNOLOGY

NUREG/CR-2015 V04: SEISMIC SAFETY MARGINS RESEARCH PROGRAM PHASE I FINAL REPORT - SOIL STRUCTURE INTERACTION (PROJECT III).

NUREG/CR-2223: AN EVALUATION OF THE SOLID ANGLE METHOD USED IN NUCLEAR CRITICALITY SAFETY.

NUREG/CR-2301: FRACTURE MECHANICS MODELS DEVELOPED FOR PIPING RELIABILITY ASSESSMENT IN LIGHT WATER REACTORS.

NUREG/CR-2522: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS PROGRAM PLAN.

NUREG/CR-2570: EXPERIMENTAL INVESTIGATION OF TEARING INSTABILITY PHENOMENA FOR STRUCTURAL MATERIALS.

NUREG/CR-2788: STRENGTH AND STIFFNESS OF UNIAXIALLY TENSIONED REINFORCED CONCRETE PANELS SUBJECTED TO MEMBRANE SHEAR.

NUREG/CR-2790: AUTOMOBILE IMPACT FORCES ON CONCRETE WALL PANELS.

EDO-RESOURCE MANAGEMENT

DIVISION OF DATA AUTOMATION & MANAGEMENT INFORMATION

NUREG/CR-2201: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1978.

OFFICE OF NUCLEAR REACTOR REGULATION (POST 4/28/80)

DIVISION OF ENGINEERING

NUREG/CR-1890: ABS, SRSS AND CDF RESPONSE COMBINATION EVALUATION FOR MARK III CONTAINMENT AND DRYWELL STRUCTURES.

NUREG/CR-2039: DYNAMIC COMBINATIONS FOR MARK II CONTAINMENT STRUCTURES.

NUREG/CR-2442: RELIABILITY ANALYSIS OF STEEL CONTAINMENT STRENGTH. Technical Report, August 1980-September 1981.

NUREG/CR-2569: RESPONSE OF THE ZION & INDIAN POINT CONTAINMENT BUILDINGS TO SEVERE ACCIDENT PRESSURES.

NUREG/CR-2591: ESTIMATING THE POTENTIAL INDUSTRIAL IMPACTS OF A NUCLEAR REACTOR ACCIDENT.

NUREG/CR-2686: REVIEW OF LOAD COMBINATIONS FOR NSSS AND BOP PIPING AND EQUIPMENT OF MARK III PLANTS.

DIVISION OF SYSTEMS INTEGRATION (POST 811005)

NUREG/CR-2387: CREDIBLE ACCIDENT ANALYSES FOR TRIGA AND TRIGA-FUELED REACTORS.

NUREG/CR-2413: SURVEY OF REMOTE AREA MONITORING SYSTEMS AT U. S. LIGHT-WATER-COOLED POWER REACTORS.

NUREG/CR-2495: CHARACTERIZATION OF SOIL TO PLANT TRANSFER COEFFICIENTS FOR STABLE CESIUM AND STRONTIUM.

NUREG/CR-2612: VARIABILITY IN DOSE ESTIMATES ASSOCIATED WITH THE FOOD CHAIN TRANSPORT AND INGESTION OF SELECTED RADIONUCLIDES.

NUREG/CR-2629: INTERIM SOURCE TERM ASSUMPTIONS FOR EMERGENCY PLANNING AND EQUIPMENT QUALIFICATION.

NUREG/CR-2633: CONTAINMENT REACTOR CAVITY SUBCOMPARTMENT ANALYSIS PROCEDURES FOR A BOILING WATER REACTOR.

NUREG/CR-2644: AN ASSESSMENT OF OFFSITE, REAL-TIME DOSE-MEASUREMENT SYSTEMS FOR EMERGENCY SITUATIONS.

NUREG/CR-2760: ASSESSMENT OF SCALE EFFECTS ON VORTEXING, SWIRL, AND INLET LOSSES IN LARGE SCALE SUMP MODELS.

NUREG/CR-2772: HYDRAULIC PERFORMANCE OF PUMP SUCTION INLETS FOR EMERGENCY CORE COOLING SYSTEMS IN BOILING WATER REACTORS.

DIVISION OF LICENSING
NUREG/CR-2664: SELECTED REVIEW OF FOREIGN LICENSING PRACTICES FOR NUCLEAR POWER PLANTS.

DIVISION OF SAFETY TECHNOLOGY
NUREG/CR-2403 S01: SURVEY OF INSULATION USED IN NUCLEAR POWER PLANTS AND THE POTENTIAL FOR DEBRIS GENERATION.

Contractor Index

This index lists, in alphabetical order, the contractors that prepared the NUREG/CR reports listed in this compilation. Listed below each contractor are the NUREG/CR numbers and titles of their reports. If further information is needed, refer to the main citation by the NUREG/CR number.

7

NUREG/CR-1030 V02: SEDIMENT AND RADIONUCLIDE TRANSPORT IN RIVERS. Phase 2-Field Sampling Program For Cattaraugus And Buttermilk Creeks, New York.
NUREG/CR-2019: THIRD PHASE OF POCKET-SIZED ELECTRONIC DOSIMETER TESTING.
NUREG/CR-2022: TECHNICAL REVIEW OF THE DISPERSION AND DOSE MODELS USED IN THE MILDOS COMPUTER PROGRAM.
NUREG/CR-2201: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1978.
NUREG/CR-2387: CREDIBLE ACCIDENT ANALYSES FOR TRIGA AND TRIGA-FUELED REACTORS.
NUREG/CR-2413: SURVEY OF REMOTE AREA MONITORING SYSTEMS AT U. S. LIGHT-WATER-COOLED POWER REACTORS.
NUREG/CR-2432: A UNIQUE CONCEPT FOR LIQUID LEVEL AND VOID FRACTION DETECTION IN SEVERE FUEL DAMAGE TESTS.
NUREG/CR-2460: TECHNICAL SUPPORT FOR IMPROVING THE LICENSING REGULATORY BASE FOR SELECTED FACILITIES ASSOCIATED WITH THE FRONT END OF THE FUEL CYCLE.
NUREG/CR-2564: ENVIRONMENTAL FACTORS AFFECTING LONG-TERM STABILIZATION OF RADON SUPPRESSION COVERS FOR URANIUM MILL TAILINGS.
NUREG/CR-2567: FINAL DATA REPORT FOR THE INSTRUMENTED FUEL ASSEMBLY (IFA)-432.
NUREG/CR-2600: END-OF-IRRADIATION DATA REPORT FOR THE INSTRUMENTED FUEL ASSEMBLY (IFA)-527.
NUREG/CR-2642: LONG-TERM SURVIVABILITY OF RIPRAP FOR ARMORING URANIUM MILL TAILINGS AND COVERS: A LITERATURE REVIEW.
NUREG/CR-2651: ACCIDENT GENERATED PARTICULATE MATERIALS AND THEIR CHARACTERISITICS--A REVIEW OF BACKGROUND INFORMATION.

BROOKHAVEN NATIONAL LABORATORY
NUREG/CR-1890: ABS, SRSS AND CDF RESPONSE COMBINATION EVALUATION FOR MARK III CONTAINMENT AND DRYWELL STRUCTURES.
NUREG/CR-2039: DYNAMIC COMBINATIONS FOR MARK II CONTAINMENT STRUCTURES.
NUREG/CR-2192 V01 N2: EVALUATION OF ISOTOPE MIGRATION-LAND BURIAL. Quarterly Progress Report, April-June 1981.
NUREG/CR-2193 V01 N2: PROPERTIES OF RADIOACTIVE WASTES AND WASTE CONTAINERS. Quarterly Progress Report, April-June 1981.
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NUREG/CR-2317 V01 N4: CONTAINER ASSESSMENT-CORROSION STUDY OF HLW CONTAINER MATERIALS. Quarterly Progress Report, October-December 1981.
NUREG/CR-2416: INITIAL QUANTIFICATION OF HUMAN ERROR ASSOCIATED WITH SPECIFIC INSTRUMENTATION AND CONTROL SYSTEM COMPONENTS IN LICENSED NUCLEAR POWER PLANTS.
NUREG/CR-2417: IDENTIFICATION AND ANALYSIS OF HUMAN ERRORS UNDERLYING PUMP AND VALVE RELATED EVENTS REPORTED BY NUCLEAR POWER PLANT LICENSEES.
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NUREG/CR-2542: SENSITIVITY STUDY USING THE FRANTIC CODE FOR THE UNAVAILABILITY OF A SYSTEM TO THE FAILURE CHARACTERISTICS OF THE COMPONENTS AND THE OPERATING CONDITIONS.
NUREG/CR-2543: A STUDY OF THE FEASIBILITY OF MICROWAVE DIELECTRIC HEATING FOR LMFBR TRANSITION PHASE ACCIDENT SEQUENCE BOILING STUDIES.
NUREG/CR-2685: EVALUATION OF CONCURRENT PEAK RESPONSES.
NUREG/CR-2686: REVIEW OF LOAD COMBINATIONS FOR NSSS AND BOP PIPING AND EQUIPMENT OF MARK III PLANTS.

BURNS & ROE CO.
NUREG/CR-2403 S01: SURVEY OF INSULATION USED IN NUCLEAR POWER PLANTS AND THE POTENTIAL FOR DEBRIS GENERATION.

CATHOLIC UNIV.

7

NUREG/CR-2737: EVALUATION OF BULK PROPERTIES OF RADWASTE GLASS AND CERAMIC CONTAINER MATERIALS TO DETERMINE LONG-TERM STABILITY.

CHIAPETTA, WELCH & ASSOCIATES, LTD.

NUREG/CR-2790: AUTOMOBILE IMPACT FORCES ON CONCRETE WALL PANELS.

COMMERCE, DEPT. OF, NATIONAL OCEANOGRAPHIC & ATMOSPHERIC ADMINISTRATION

NUREG/CR-2521: METHOD FOR ESTIMATING WAKE FLOW AND EFFLUENT DISPERSION NEAR SIMPLE BLOCK-LIKE BUILDINGS.

NUREG/CR-2584: METEROLOGICAL CONSIDERATIONS IN THE DEVELOPMENT OF A REAL-TIME ATMOSPHERIC DISPERSION MODEL FOR REACTOR EFFLUENT EXPOSURE PATHWAY.

NUREG/CR-2637: EMERGENCY RESPONSE CAPABILITIES AND EXAMPLE ASSESSMENTS FOR AIRBORNE RADIONUCLIDE DISCHARGES.

NUREG/CR-2639: HISTORICAL EXTREME WINDS FOR THE UNITED STATES-ATLANTIC AND GULF OF MEXICO COASTLINES.

COMMERCE, DEPT. OF

NUREG/CR-2591: ESTIMATING THE POTENTIAL INDUSTRIAL IMPACTS OF A NUCLEAR REACTOR ACCIDENT.

COMMERCE, DEPT. OF, NATIONAL BUREAU OF STANDARDS

NUREG/CR-2638: SNOW LOADS FOR THE DESIGN OF NUCLEAR POWER PLANT STRUCTURES.

CORNELL UNIV.

NUREG/CR-2788: STRENGTH AND STIFFNESS OF UNIAXIALLY TENSIONED REINFORCED CONCRETE PANELS SUBJECTED TO MEMBRANE SHEAR.

DAVID W. TAYLOR NAVAL RESEARCH & DEVELOPMENT CENTER

NUREG/CR-2570: EXPERIMENTAL INVESTIGATION OF TEARING INSTABILITY PHENOMENA FOR STRUCTURAL MATERIALS.

EG&G, INC.

NUREG/CR-0169 V13: LOFT EXPERIMENTAL MEASUREMENTS UNCERTAINTY ANALYSES. Volume XIII. Temperature Measurements.

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NUREG/CR-2496: HUMAN ENGINEERING DESIGN CONSIDERATIONS FOR CATHODE RAY TUBE-GENERATED DISPLAYS.

NUREG/CR-2618: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A NATURAL CIRCULATION TEST S-NC-7C.

NUREG/CR-2636: EXPERIMENTAL DATA REPORT FOR AIR-WATER FLOODING TESTS OF THE FLECHT-SEASET PROGRAM SET FACILITY VESSEL UPPER PLENUM.

NUREG/CR-2648: EXPERIMENTAL DATA REPORT FOR SEMISCALE MOD-2A NATURAL CIRCULATION TEST SERIES (TESTS S-NC-8B AND S-NC-9).

NUREG/CR-2711: PERFORMANCE AND DESIGN REQUIREMENTS FOR A GRAPHICS DISPLAY RESEARCH FACILITY.

NUREG/CR-2717: EXPERIMENT DATA REPORT FOR LOFT ANTICIPATED TRANSIENT WITHOUT SCRAM EXPERIMENT L9-3.

NUREG/CR-2732: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A INTERMEDIATE BREAK TEST SERIES. (Tests S-IB-1 And S-IB-2).

EXXON NUCLEAR CO., INC. (SUBS. OF EXXON CORP.)

NUREG/CR-2495: CHARACTERIZATION OF SOIL TO PLANT TRANSFER COEFFICIENTS FOR STABLE CESIUM AND STRONTIUM.

NUREG/CR-2644: AN ASSESSMENT OF OFFSITE, REAL-TIME DOSE-MEASUREMENT SYSTEMS FOR EMERGENCY SITUATIONS.

GENERAL ELECTRIC CO.

NUREG/CR-2133: BWR REFILL-RELOAD PROGRAM TASK 4.4 - 30 SSTF DESCRIPTION DOCUMENT.

NUREG/CR-2229 V01: BWR LARGE BREAK SIMULATION TESTS--BWR BLOWDOWN/EMERGENCY CORE COOLING PROGRAM.
NUREG/CR-2231: BWR LOW FLOW BUNDLE UNCOVERY TEST AND ANALYSIS.

GEO-CENTERS, INC.
NUREG/CR-2589: A GROUND-PENETRATING RADAR SURVEY OF THE MAXEY FLATS LOW-LEVEL NUCLEAR WASTE DISPOSAL SITE, FLEMING COUNTY, KENTUCKY.

INSTITUTE OF ELECTRICAL & ELECTRONIC ENGINEERS
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NUREG/CR-2464: METHODS FOR CLASSIFYING MIXTURES OF EXPONENTIAL DISTRIBUTIONS BASED ON EITHER EXPONENTIAL OR POISSON DATA.
NUREG/CR-2565: STRUCTURAL PERFORMANCE OF HEPA FILTERS UNDER SIMULATED TORNADO CONDITIONS.
NUREG/CR-2569: RESPONSE OF THE ZION & INDIAN POINT CONTAINMENT BUILDINGS TO SEVERE ACCIDENT PRESSURES.
NUREG/CR-2622: ANALYSIS OF TRAC AND SCTF RESULTS FOR SYSTEM PRESSURE-EFFECTS TESTS UNDER FORCED FLOODING (RUNS 506, 507 AND 508).
NUREG/CR-2632: RESPONSE OF CENTRIFUGAL BLOWERS TO SIMULATED TORNADO TRANSIENTS. July-September 1981.
NUREG/CR-2633: CONTAINMENT REACTOR CAVITY SUBCOMPARTMENT ANALYSIS PROCEDURES FOR A BOILING WATER REACTOR.
NUREG/CR-2652: EVALUATION AND PERFORMANCE OF CLOSED-CIRCUIT BREATHING APPARATUS.

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MARVIKEN
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MICHIGAN, STATE OF
NUREG/CR-2736: TRANSPORTATION OF RADIOACTIVE MATERIAL IN MICHIGAN. September 1980-August 1981.

MISSOURI, UNIV. OF, COLUMBIA
NUREG/CR-2359: ATMOSPHERIC STRUCTURE PRIOR TO TORNADOES AS DERIVED FROM PROXIMITY AND PRECEDENT UPPER AIR SOUNDINGS.

MITRE CORP.
NUREG/CP-0026: WORKSHOP ON PSYCHOLOGICAL STRESS ASSOCIATED WITH THE PROPOSED RESTART OF THREE MILE ISLAND, UNIT 1.
NEW YORK, STATE OF
NUREG/CR-2381: GEOLOGIC AND HYDROLOGIC RESEARCH AT THE WESTERN NEW YORK NUCLEAR SERVICE CENTER, WEST VALLEY, NEW YORK. Progress Report, August 1979-July 1981.
NORTHEAST MISSOURI STATE UNIV.
NUREG/CR-2565: STRUCTURAL PERFORMANCE OF HEPA FILTERS UNDER SIMULATED TORNADO CONDITIONS.
NORTHWESTERN UNIV.
NUREG/CR-2334: INTERPHASE TRANSPORT IN HORIZONTAL STRATIFIED CONCURRENT FLOW.
NUREG/CR-2783: COUNTERCURRENT STEAM-WATER FLOW IN A FLAT PLATE GEOMETRY.
NUCLEAR ASSURANCE CORP.
NUREG/CR-2704: U.S. REACTOR SPENT-FUEL STORAGE CAPABILITIES.
OAK RIDGE NATIONAL LABORATORY
NUREG/CP-0022: PROCEEDINGS OF THE SYMPOSIUM ON UNCERTAINTIES ASSOCIATED WITH THE REGULATION OF THE GEOLOGIC DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTE.
NUREG/CR-0200 ERR: SCALE: MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION.
NUREG/CR-2000 V01 N2: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of February 1982.
NUREG/CR-2000 V01 N3: LICENSEE EVENT REPORT (LER) COMPILATION: For Month of March 1982.
NUREG/CR-2000 V01 N4: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of April 1982.
NUREG/CR-2000 V01 N5: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of May 1982.
NUREG/CR-2053: HEAT TRANSFER ANALYSIS OF THE LWR PRESSURE VESSEL STEEL IRRADIATION CAPSULES IN THE OAK RIDGE RESEARCH REACTOR-PRESSURE VESSEL BENCHMARK FACILITY.
NUREG/CR-2141 V04: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM. Quarterly Progress Report For October-December 1981.
NUREG/CR-2172: SUMMARY AND BIBLIOGRAPHY OF SAFETY-RELATED EVENTS AT BOILING-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1980.
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NUREG/CR-2173: SUMMARY AND BIBLIOGRAPHY OF SAFETY-RELATED EVENTS AT PRESSURIZED-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1980.
NUREG/CR-2173 ERR: SUMMARY AND BIBLIOGRAPHY OF SAFETY-RELATED EVENTS AT PRESSURIZED-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1980.
NUREG/CR-2184: COMPARISON OF THE RADIOLOGICAL IMPACTS OF THORIUM AND URANIUM NUCLEAR FUEL CYCLES.
NUREG/CR-2204 V04: ADVANCED TWO-PHASE FLOW INSTRUMENTATION PROGRAM. Quarterly Progress Report, October-December 1981.
NUREG/CR-2220 V02: THE IMPACT OF ENTRAINMENT AND IMPINGEMENT ON FISH POPULATIONS IN THE HUDSON RIVER ESTUARY.
NUREG/CR-2220 V03: THE IMPACT OF ENTRAINMENT AND IMPINGEMENT ON FISH POPULATIONS IN THE HUDSON RIVER ESTUARY.
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NUREG/CR-2223: AN EVALUATION OF THE SOLID ANGLE METHOD USED IN NUCLEAR CRITICALITY SAFETY.
NUREG/CR-2299 V04: AEROSOL RELEASE AND TRANSPORT PROGRAM. Quarterly Progress Report For October-December 1981.
NUREG/CR-2306: CSRL-V: PROCESSED ENDF/B-V 227-NEUTRON-GROUP AND

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NUREG/CR-2353 V02: SPECIFICATION AND VERIFICATION OF NUCLEAR POWER PLANT TRAINING SIMULATOR RESPONSE CHARACTERISTICS. Part II: Conclusions And Recommendations.
NUREG/CR-2362: RELATIONSHIPS BETWEEN CHARPY V-NOTCH IMPACT ENERGY AND FRACTURE TOUGHNESS.
NUREG/CR-2366 V02: MULTIROD BURST TEST PROGRAM PROGRESS REPORT FOR JULY-DECEMBER 1981.
NUREG/CR-2392: SUMMARY OF ORNL WORK ON NRC-SPONSORED HTGR SAFETY RESEARCH, JULY 1974-SEPTEMBER 1980.
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NUREG/CR-2435: DISPERSED FLOW FILM BOILING IN ROD BUNDLE GEOMETRY-STEADY STATE HEAT TRANSFER DATA AND CORRELATION COMPARISONS.
NUREG/CR-2455: EXPERIMENTAL INVESTIGATIONS OF BUNDLE BOILOFF AND REFLOOD UNDER HIGH-PRESSURE LOW HEAT FLUX CONDITIONS.
NUREG/CR-2456: EXPERIMENTAL INVESTIGATIONS OF UNCOVERED-BUNDLE HEAT TRANSFER AND TWO-PHASE MIXTURE-LEVEL SWELL UNDER HIGH-PRESSURE LOW HEAT-FLUX CONDITIONS.
NUREG/CR-2469: AN ANALYSIS OF TRANSIENT FILM BOILING OF HIGH-PRESSURE WATER IN A ROD BUNDLE.
NUREG/CR-2470: THERMOMETRY IN THE MULTIROD BURST TEST PROGRAM.
NUREG/CR-2493: AQUEOUS IODINE CHEMISTRY IN LWR ACCIDENTS: Review And Assessment.
NUREG/CR-2494: OR-FLAW: A FINITE ELEMENT PROGRAM FOR DIRECT EVALUATION OF K-FACTORS FOR USER-DEFINED FLAWS IN PLATES, CYLINDERS AND PRESSURE-VESSEL NOZZLE CORNERS.
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NUREG/CR-2545: DESIGN CONCEPT AND TESTING OF AN IN-BUNDLE GAMMA DENSITOMETER FOR SUBCHANNEL VOID FRACTION MEASUREMENTS IN THE THTF ELECTRICALLY HEATED ROD BUNDLE.
NUREG/CR-2586: A SURVEY OF METHODS FOR IMPROVING OPERATOR ACCEPTANCE OF COMPUTERIZED AIDS.
NUREG/CR-2587: FUNCTIONS AND OPERATIONS OF NUCLEAR POWER PLANT CREWS.
NUREG/CR-2610: RAGBEEF: A FORTRAN IV IMPLEMENTATION OF A TIME-DEPENDENT MODEL FOR RADIONUCLIDE CONTAMINATION OF BEEF.
NUREG/CR-2612: VARIABILITY IN DOSE ESTIMATES ASSOCIATED WITH THE FOOD CHAIN TRANSPORT AND INGESTION OF SELECTED RADIONUCLIDES.
NUREG/CR-2629: INTERIM SOURCE TERM ASSUMPTIONS FOR EMERGENCY PLANNING AND EQUIPMENT QUALIFICATION.
NUREG/CR-2692: AN INTEGRATED SYSTEM FOR FORECASTING ELECTRIC ENERGY AND LOAD FOR STATES AND UTILITY SERVICE AREAS.

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NUREG/CR-2314: AGING WITH RESPECT TO FLAMMABILITY AND OTHER PROPERTIES IN FIRE-RETARDED ETHYLENE PROPYLENE RUBBER AND CHLOROSULFONATED POLYETHYLENE.

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NUREG/CR-2350: SENSITIVITY ANALYSIS TECHNIQUES: SELF-TEACHING CURRICULUM.

NUREG/CR-2377: TESTS & CRITERIA FOR FIRE PROTECTION OF CABLE PENETRATIONS.

NUREG/CR-2393 ERR: Errata, changing rept number to NUREG/CR-2593, to A USER'S MANUAL FOR COMPUTER CODE RIBD/IRT.

NUREG/CR-2394 ERR: Errata, changing rept number to NUREG/CR-2594, to A USER'S MANUAL FOR THE GABAS SPECTRUM COMPUTER CODE.

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NUREG/CR-2559: RESULTS OF PHASE ONE OF PLANT ELECTRICAL SYSTEM (PES) STUDY.

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UNITED NUCLEAR CORP.
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VANDERBILT UNIV.
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VIRGINIA, UNIV. OF
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8

8

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1 Main Citations and Abstracts

2 Contractor Report Number Index

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5 NRC Originating Organization Index

6 NRC Contractor Sponsor Index

7 Contractor Index

8 Licensed Facility Index