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Vol. 7, No. 4

Regulatory and Technical Reports

Compilation for 1982

U.S. Nuclear Regulatory Commission

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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
Attn: Ann W. Savolainen
Landow 212
U.S. Nuclear Regulatory Commission
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

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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-0011 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SEQUOYAH NUCLEAR PLANT UNITS 1 & 2. Docket Nos. 50-327 And 50-328. (Tennessee Valley Authority) * Office of Nuclear Reactor Regulation, Director. December 1982. 42pp. 8301190438. 16849:076.

Supplement No. 6 to the Safety Evaluation Report (SER) related to the operation of the Tennessee Valley Authority's Sequoyah Nuclear Plant, Units 1 and 2, located in Hamilton County, Tennessee, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The purpose of this supplement is to update the staff's evaluations of the issues related to the hydrogen mitigation system identified in the SER and previous supplements as needing resolution.

NUREG-0020 V05 N11: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of October 31, 1981. (Grey Book) * Office of Management and Program Analysis. February 1982. 250pp. 8203040428. 12127:001.

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 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 V05 N12: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of November 30, 1981. (Grey Book) * Office of Management and Program Analysis. March 1982. 200pp. 8203190036. 12344:046.

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 N01: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of December 31, 1981. (Grey Book) * Office of Management and Program Analysis. March 1982. 300pp. 8204290412. 12900:342.

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 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 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. Data As Of February 28, 1982. (Grey Book) * Management Information Branch. June 1982. 363pp. 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-0020 V06 N04: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As of March 31, 1982. (Grey Book) * Management Information Branch. August 1982. 419pp. 8208240424. 14543:064.

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 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 N05: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of April 30, 1982. (Grey Book) * Management Information Branch. September 1982. 375pp. 8209270092. 15515:001.

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 N06: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data as of May 31, 1982. (Grey Book) * Management Information Branch. October 1982. 2pp. 8211040525. 16029:080.

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 Renional 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 N07: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of June 30, 1982. (Grey Book) * Management Information Branch. November 1982. 376pp. 8212270505. 16547:001.

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 N08: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of July 31, 1982. (Grey Book) * Management Information Branch. December 1982. 395pp. 8301100004. 16717:014.

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. 170pp. 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-0030 V06 N02: NUCLEAR POWER PLANTS-CONSTRUCTION STATUS REPORT. Data As Of June 30, 1982. (Yellow Book) * Management Information Branch. October 1982. 163pp. 8210220142. 15795:325.
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 V05 N02ER: Errata changing report number to NUREG-0040, Volume 5, Numbers 2, 3, LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, April 1981-September 1981. * Director's Office, Office of Inspection and Enforcement. January 7, 1982. 1p. 8201210216. 11648:186.
This periodical provides the results of inspections performed under the NRC's Licensee Contractor and Vendor Inspection Program that have been distributed to the inspected organizations during the period from October 1981 through December 1981. Also included in this issue are the results of certain inspections performed prior to October 1981 that were not included in previous issues of NUREG-0040.

NUREG-0040 V05 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, October 1981-December 1981. (White Book) * Director's Office, Office of Inspection and Enforcement. January 1982. 81pp. 8201210169. 11646:354.
This periodical provides the results of inspections performed under the NRC's Licensee Contractor and Vendor Inspection Program that have been distributed to the inspected organizations during the period from October 1981 through December 1981. Also included in this issue are the results of certain inspections performed prior to October 1981 that were not included in previous issues of NUREG-0040.

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-0040 V06 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, April 1982-June 1982. (White Book) * Region 4, Office of Director. July 1982. 178pp. 8207220652. 14023:034.
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 April 1982 through June 1982. Also included in this issue are the results of certain

inspections performed prior to April 1982 that were not included in previous issues of NUREG-0040.

NUREG-0040 V06 N03: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, July 1982-September 1982. (White Book) * Region 4, Office of Director. October 1982. 244pp. 8211160508. 16113:017.

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 July 1982 through September 1982. Also included in this issue are the results of certain inspections performed prior to July 1982 that were not included in previous issues of NUREG-0040.

NUREG-0090 V04 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. July-September 1981. * Director's Office. January 1982. 25pp. 8203020158. 12090:073.

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 July 1 to September 30, 1981.

During the report period, there were two abnormal occurrences at the nuclear power plants licensed to operate. One involved a misalignment of a high head safety injection isolation valve. The other involved a failure of the high pressure safety injection system. There were two abnormal occurrences at other licensee facilities. Both involved calculated radiation exposures in excess of 10 CFR 20 limits. There were two abnormal occurrences reported by the Agreement States. One involved excessive radiation doses to hospital patients. The second involved overexposures of a radiographer and two barge crew members.

The report also contains information updating some previously reported abnormal occurrences.

NUREG-0090 V04 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. 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-0090 V05 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. January-March 1982. * Director's Office. August 1982.

54pp. 8209020111. 14730:050.

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 January 1 to March 31, 1982.

During the report period, there were four abnormal occurrences at the nuclear power plants licensed to operate. The first involved diesel generator engine cooling system failures. The second involved pressure transients during shutdown. The third involved major deficiencies in management controls. The fourth involved a steam generator tube rupture. There were no abnormal occurrences for the other NRC licensees during the report period. The Agreement States reported no abnormal occurrences to the NRC.

The report also contains information updating some previously reported abnormal occurrences.

NUREG-0090 V05 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1982. * Director's Office. December 1982. 50pp. 8301100056. 16757:040.

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 April 1 to June 30, 1982.

During the report period, there were no abnormal occurrences at the nuclear power plants licensed to operate. There were no abnormal occurrences for the other NRC licensees. The Agreement States reported no abnormal occurrences to the NRC.

The report also contains information updating some previously reported abnormal occurrences. Some of the updates have been given more generalized titles (as compared to their former more specific titles) to include some new events which are associated in some respects to previously reported abnormal occurrences. The items which have been retitled are discussed in Appendix B of the report.

NUREG-0139 V01 S01: SUPPLEMENT TO FINAL ENVIRONMENTAL STATEMENT RELATED TO CONSTRUCTION AND OPERATION OF CLINCH RIVER BREEDER REACTOR PLANT.Main Report.Docket No. 50-537.(U.S. Department of Energy,Tennessee Valley Authority & Project Management Corp) LEECH,P.H. Clinch River Breeder Reactor Program Office. October 1982. 318pp. 8211160500. 16113:261.

This is a supplement to the 1977 Final Evironmental Statement (FES) relative to construction and operation of the proposed Clinch River Breeder Reactor Plant at Oak Ridge, Tennessee. It provides the staff's assessment of additional data relative to the site and environs and modifications of the plant design and its fuel cycle which have occurred since the FES was issued. The staff's overall conclusion is unchanged; that is, that the action called for is the issuance of a construction permit subject to certain limitations for protection of the environment.

NUREG-0267 R01: PRINCIPLES AND PRACTICES FOR KEEPING OCCUPATIONAL RADIATION EXPOSURES AT MEDICAL INSTITUTIONS AS LOW AS REASONABLY ACHIEVABLE. BRODSKY,A. Division of Facility Operations. October

1982. 90pp. 8210270331. 15848:022.

This report is a companion document to Regulatory Guide 8.18, "Information Relevant to Ensuring that Occupational Radiation Exposures at Medical Institutions Will Be As Low As Reasonably Achievable." Both documents have now been revised to incorporate many good suggestions received after the original documents were published for comment. This report is a compendium of good practices and helpful information derived from the experience of the radiological and health physics professions and is not to be construed in any way as additional regulatory requirements of the Nuclear Regulatory Commission. The information presented, including comprehensive checklists of facilities, equipment, and procedures that should be considered for working with NRC-licensed materials in all types of hospital activities, is intended to aid the NRC licensee in fulfilling the philosophy of maintaining radiation exposures of employees, patients, visitors, and the public as low as reasonably achievable (ALARA). Each subsection of this report is designed to include the major radiation safety considerations pertaining to the respective hospital function. Thus, the busy health professional will need to read only a few pages of this document at any one time to obtain the information needed.

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-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-0304 V07 N02: REGULATORY AND TECHNICAL REPORTS. Compilation For Second Quarter 1982. SAVOLAINEN, A. Division of Technical Information & Document Control. August 1982. 155pp. 8209010450. 14717:001.

This compilation lists all NRC regulatory and technical reports published under the series during the second quarter of 1982.

NUREG-0309 S03: SAFETY EVALUATION REPORT RELATED TO THE CONSTRUCTION OF SKAGIT/HANFORD NUCLEAR PROJECT, UNITS 1 AND 2. Docket Nos. 50-522 And 50-523. (Puget Sound Power And Light Company et al.) * Office of Nuclear Reactor Regulation, Director. December 1982. 221pp. 8301190470. 16850:126.

Supplement 3 to the Safety Evaluation Report for the application filed by Puget Sound Power and Light Company on behalf of itself, the

Pacific Power and Light Company, the Washington Water Power Company, and The Portland General Electric Company for construction permits to build the Skagit/Hanford Nuclear Project has been issued by the Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. This supplement is an evaluation of the site relocation amendment to the Preliminary Safety Analysis Report. The proposed site has been relocated from Skagit County, Washington, to the Department of Energy's Hanford Reservation.

NUREG-0325: U.S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS. * Office of Management and Program Analysis. February 1982. 110pp. 8203040433. 12158:267.

Functional organization charts for the NRC Commission Offices, Divisions, Staffs and Branches are presented.

NUREG-0327 R03: OWNERS OF NUCLEAR POWER PLANTS:PERCENTAGE OWNERSHIP OF COMMERCIAL NUCLEAR POWER PLANTS BY UTILITY COMPANIES. WOOD,R.S. Office of State Programs, Director. November 1982. 39pp. 8212140471. 16418:285.

The report indicates percentage ownership of commercial nuclear power plants by utility companies. The report includes all plants operating, under construction, docketed for NRC safety and environmental reviews, or under NRC antitrust review, but does not include those plants announced but not yet under review or those plants formally cancelled. Part I of the report lists plants alphabetically with their associated applicants and percentage ownership. Part II lists applicants alphabetically with their associated plants and percentage ownership. Part I also indicates which plants have received operating licenses (OLs).

NUREG-0386 S03 D02: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.January 1979-December 1980. * Office of the Executive Legal Director. August 1982. 85pp. 8209270033. 15519:126.

The third supplement to the second edition of NRC Staff Practice and Procedure Digest contains digests of Commission, Atomic Safety and Licensing Appeal Board and Atomic Safety and Licensing Board decisions issued during the period 1/1/79 to 12/31/80 interpreting the NRC's Rules of Practice (10 CFR Part 2). The supplement includes a number of new subsections and topics not covered in the Digest. These new sections are noted in the index. The Practice and Procedures Digest and its supplements were prepared by attorneys in the NRC Office of the Executive Legal Director. It has been published as a reference tool for all persons interested in NRC proceedings.

NUREG-0390 V05 N02: TOPICAL REPORT REVIEW STATUS. (Blue Book) * Office of Management and Program Analysis. January 1982. 150pp. 8202050093. 11832:206.

The primary purpose of this document is to provide periodic progress reports of on-going topical report reviews, to identify those topical reports for which the Nuclear Regulatory Commission (NRC) staff review has been completed and, to the extent practicable, to provide NRC management with sufficient information regarding the conduct of the topical report program to permit taking whatever actions deemed necessary or appropriate.

NUREG-0390 V06 N01: TOPICAL REPORT REVIEW STATUS. (Blue Book) * Management Information Branch. August 1982. 211pp. 8208230001. 14507:081.

The primary purpose of this document is to provide periodic progress reports of on-going topical report reviews, to identify those topical reports for which the Nuclear Regulatory Commission (NRC) staff review has been completed and, to the extent practicable, to provide NRC management with sufficient information regarding the conduct of the topical report program to permit taking whatever actions deemed necessary or appropriate.

NUREG-0420 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF THE SHOREHAM NUCLEAR POWER STATION, UNIT NO. 1. Docket No. 50-322. (Long Island Lighting Company) * Office of Nuclear Reactor Regulation, Director. February 1982. 32pp. 8203190075. 12352:333.

Supplement No. 2 to the Safety Evaluation Report of Long Island Lighting Company's application for a license to operate the Shoreham Nuclear Power Station, Unit 1, located in Suffolk County, New York, 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-0430 V02 N01: LICENSED FUEL FACILITY STATUS REPORT INVENTORY DIFFERENCE. Data As Of June 1982. (Grey Book) * Director's Office, Office of Inspection and Enforcement. July 1982. 15pp. 8208090004. 14302:152.

NRC is committed to the periodic release of inventory difference data from the licensed fuel facilities after the agency has had an opportunity to review the data and has performed any related investigations associated with the data. This report, NUREG-0430, Vol. 2, is a continuation of NUREG-0430, Vol. 1, for reporting inventory difference data for active, licensed fuel facilities.

NUREG-0430 V02 N02: LICENSED FUEL FACILITY STATUS SUMMARY REPORT. Inventory Difference Data. July 1981-December 1981. * Director's Office, Office of Inspection and Enforcement. October 1982. 15pp. 8211080008. 15974:192.

NRC is committed to the periodic publication of licensed fuel facilities inventory difference data, following agency review of the information and completion of any related investigations. Information included in this report includes inventory difference data for active fuel fabrication facilities possessing more than one effective kilogram of high enriched uranium, low enriched uranium, plutonium, or uranium-233.

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 N01: SYSTEMATIC EVALUATION PROGRAM STATUS SUMMARY REPORT. Data As Of January 31, 1982. (Buff Book) * Office of Management and Program Analysis. February 1982. 92pp. 8203040140. 12120:244.
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 N02: SYSTEMATIC EVALUATION PROGRAMS STATUS SUMMARY REPORT. Data As Of February 28, 1982. (Buff Book) * Office of Management and Program Analysis. March 1982. 50pp. 8203300305. 12460:092.
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 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.

NUREG-0485 V04 N06: SYSTEMATIC EVALUATION PROGRAM STATUS SUMMARY REPORT. Data As Of June 30, 1982. (Buff Book) * Management Information Branch. July 1982. 88pp. 8208090012. 14283:214.
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.

NUREG-0485 V04 N07: SYSTEMATIC EVALUATION PROGRAM STATUS SUMMARY REPORT. Data As Of July 31, 1982. (Buff Book) * Management Information Branch. August 1982. 96pp. 8209020198. 14726:277.

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.

NUREG-0485 V04 N08: SYSTEMATIC EVALUATION PROGRAM STATUS SUMMARY REPORT. Data As Of August 31, 1982. (Beige Book) * Management Information Branch. September 1982. 95pp. 8209280337. 15548:242.

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.

NUREG-0485 V04 N10: SYSTEMATIC EVALUATION PROGRAM STATUS SUMMARY REPORT. Data As Of October 31, 1982. (Beige Book) * Management Information Branch. November 1982. 97pp. 8212010138. 16293:139.

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.

NUREG-0485 V04 N11: SYSTEMATIC EVALUATION PROGRAM STATUS SUMMARY REPORT. Data As Of November 30, 1982. (Beige Book) * Management Information Branch. December 1982. 59pp. 8301100021. 16749:152.

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 overal completion status.

NUREG-0519 S02: 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) * Office of Nuclear Reactor Regulation, Director. February 1982. 125pp. 8203030262. 12111:001.

Supplement No. 2 to the Safety Evaluation Report of Commonwealth Edison Company's application for licenses to operate its LaSalle County Station, Units 1 and 2, located in Brookfield Township, LaSalle County, Illinois has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement provides further information on outstanding items from the Safety Evaluation Report and addresses a new sub-section entitled, "Assurance of Proper Design and Construction."

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-0519 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF LA SALLE COUNTY STATION, UNITS 1 AND 2. Docket Nos. 50-373 And 50-374. (Commonwealth Edison Company) * Office of Nuclear Reactor Regulation, Director. July 1982. 36pp. 8207210136. 13993:037.

Supplement No. 4 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. On April 17, 1982, we issued a license, NPF-11, to allow Unit 1 operation at power levels not to exceed 5 percent of rated power. This supplement addresses matters for proceeding to full power.

NUREG-0525 R05: SAFEGUARDS SUMMARY EVENT LIST. MACMURDY, P.; DAVIDSON, J.; LIN, H. Office of Nuclear Material Safety & Safeguards, Director. July 1982. 59pp. 8208230417. 14518:297.

The Safeguards Summary Event List (SSEL) provides brief summaries of several hundred safeguards-related events involving nuclear material or facilities regulated by the U.S. Nuclear Regulatory Commission (NRC). Events are described under the categories of bomb-related, intrusion, missing and/or allegedly stolen, transportation, vandalism, arson, firearms, sabotage and miscellaneous. The information contained in the event descriptions is derived primarily from official NRC reporting channels.

NUREG-0528 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF THE WM. H. ZIMMER NUCLEAR POWER STATION, UNIT 1. Docket No. 50-358. (Cincinnati Gas and Electric Company) * Office of Nuclear Reactor Regulation, Director. August 1982. 41pp. 8209130286. 14780:313.

The Safety Evaluation Report for the Zimmer Nuclear Power Station, Unit 1 was issued in January 1979. At the time of issuance there were two outstanding issues. Supplement No. 1, issued in June 1981 discussed the resolution of these issues and the concerns of the Advisory Committee on Reactor Safeguards, which issued a favorable report on March 13, 1979. Supplement No. 2, issued in October 1981 discussed subsequent outstanding issues since June 1981. This Supplement closes out outstanding issues and concludes that the facility can be operated by the applicant without endangering the health and safety of the public. The Zimmer Station is located in Washington Township, Clermone County, Ohio.

NUREG-0537: DRAFT ENVIROMENTAL STATEMENT 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. February 1982. 115pp. 8203010408. 12076:060.

This draft environmental statement contains the second assessment of the environmental impact associated with the operation of the Midland Plant, Units 1 and 2, 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 Midland project, alternatives to the project, the affected environment, environmental consequences and mitigating actions, and environmental and economic benefits and costs. Land-use and terrestrial and aquatic-ecological impacts will be small. Air quality impacts will also be small. Impacts to historic and prehistoric sites will be negligible. Chemical discharges are expected to further decrease the existing marginal water quality of the Tittabawassee River and may adversely affect future downstream water use, but will be required to meet conditions of the plant's NPDES permit. The effects of routine operations, energy transmission, and periodic maintenance of rights-of-way and transmission line facilities should not jeopardize any populations of endangered or threatened species. No significant impacts are anticipated from normal operational releases of radioactivity. The risk associated with accidental radiation exposure is very low. The net socioeconomic effects of the project will be beneficial. The action called for is for issuance of operating licenses for the Midland Plant, Units 1 and 2.

NUREG-0537: FINAL ENVIRONMENTAL STATEMENT 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. July 1982. 480pp. 8208170006. 14375:082.

This final environmental statement contains the second assessment of the environmental impact associated with operation of the Midland Plant, Units 1 and 2 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 Midland project, alternatives to the project, the affected environment, environmental consequences and mitigating actions, and environmental and economic benefits and costs. Land-use and terrestrial- and aquatic-ecological impacts will be small. Air-quality impacts will also be small. Impacts to historic and prehistoric sites will be negligible. Chemical discharges are expected to further decrease the existing marginal water quality of the Tittabawassee River and may adversely affect future downstream water use, but will be required to meet conditions of the plant's NPDES permit. The effects of routine operations, energy transmission, and periodic maintenance of rights-of-way and transmission line facilities should not jeopardize any populations of endangered or threatened species. No significant impacts are anticipated from normal operational releases of radioactivity. The risk associated with accidental radiation exposure is very low. The net socioeconomic effects of the project will be beneficial. The action called for is the issuance of operating licenses for the Midland Plant, Units 1 and 2.

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. 404pp. 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 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE NOVEMBER 1-30, 1981. * Division of Technical Information & Document Control. March 1982. 390pp. 8203170172. 12326:001.

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 Sources Index, and Report Number Index.

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 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. 112pp. 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. 681pp. 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-0540 V04 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30 1982. * Division of Technical Information & Document Control. October 1982. 655pp. 8211030511. 15931:056.

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 following indexes are included: Personal Author Index, Corporate Source Index, Report Number Index, and Cross Reference to Principal Documents Index.

NUREG-0540 V04 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. May 1-31, 1982. * Division of Technical Information & Document Control. October 1982. 554pp. 8211110067. 16045:183.

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 following indexes are included: Personal Author

Index, Corporate Source Index, Report Number Index, and Cross Reference to Principal Documents Index.

NUREG-0540 V04 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. June 1-30, 1982. * Division of Technical Information & Document Control. November 1982. 610pp. 8212140475. 16419:001.

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 following indexes are included: Personal Author Index, Corporate Source Index, Report Number Index, and Cross Reference to Principal Documents Index.

NUREG-0540 V04 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE July 1-31, 1982. * Division of Technical Information & Document Control. November 1982. 580pp. 8212270068. 16567:001.

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 following indexes are included: Personal Author Index, Corporate Source Index, Report Number Index, and Cross Reference to Principal Documents Index.

NUREG-0540 V04 N08: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. August 1-31, 1982. * Division of Technical Information & Document Control. December 1982. 587pp. 8301100002. 16715:001.

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 following indexes are included: Personal Author Index, Corporate Source Index, Report Number Index, and Cross Reference to Principal Documents Index.

NUREG-0566 V02 N01: STANDARDS DEVELOPMENT STATUS SUMMARY REPORT. Data As of December 31, 1981 And January 31, 1982. (Green Book) * Office of Management and Program Analysis. February 1982. 250pp. 8203040426. 12126:001.

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

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-0566 V02 N03: STANDARDS DEVELOPMENT STATUS SUMMARY REPORT. Data As Of June 1982. (Green Book) * Management Information Branch. July 1982. 211pp. 8208130471. 14354:059.

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 April 19, 1982. (Blue Book) * Management Information Branch. May 1982. 75pp. 8206110008. 13492:196.

Provides 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 a review 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-0580 V11 N06: REGULATORY LICENSING STATUS SUMMARY REPORT. Data As Of June 16, 1982. (Blue Book) * Management Information Branch. July 1982. 73pp. 8207220654. 13996:278.

Provides a review 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-0580 V11 N07: REGULATORY LICENSING STATUS SUMMARY REPORT. Data As Of July 10, 1982. (Blue Book) * Management Information Branch. July 1982. 72pp. 8208180120. 14391:215.

Provides a review 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-0580 V11 N08: REGULATORY LICENSING STATUS SUMMARY REPORT. Data As Of August 15, 1982. (Blue Book) * Management Information Branch. August 1982. 69pp. 8209100220. 14767:223.

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

NUREG-0580 V11 N09: REGULATORY LICENSING STATUS SUMMARY REPORT. Data As Of September 15, 1982. (Blue Book) * Management Information Branch. September 1982. 63pp. 8210120119. 15689:084.

Provides a review 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-0580 V11 N10: REGULATORY LICENSING STATUS SUMMARY REPORT. Data As Of October 15, 1982. (Blue Book) * Management Information Branch. October 1982. 70pp. 8211160476. 16111:115.

Provides a review 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-0580 V11 N11: REGULATORY LICENSING STATUS SUMMARY REPORT. Data As Of November 30, 1982. (Blue Book) * Management Information Branch. December 1982. 70pp. 8212270174. 16551:203.

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

NUREG-0606 V04 N01: UNRESOLVED SAFETY ISSUES SUMMARY. Data As of February 19, 1982. (Aqua Book) * Office of Management and Program Analysis. March 1982. 48pp. 8203170181. 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-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-0606 V04 N03: UNRESOLVED SAFETY ISSUES SUMMARY. Data As Of August 20, 1982. (Aqua Book) * Office of Resource Management, Director. September 1982. 51pp. 8209270450. 15527:317.

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-0606 V04 N04: UNRESOLVED SAFETY ISSUES SUMMARY. Data As Of November 19, 1982. (Aqua Book) * Office of Resource Management, Director. December 1, 1982. 50pp. 8212270243. 16550:186.

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-0650 S01: TECHNICAL WRITING STYLE GUIDE: AN ALTERNATIVE REFERENCE SYSTEM FOR NRC PUBLICATIONS. * Division of Technical Information & Document Control. February 1982. 17pp. 8203100012. 12191:335.

This supplement to the U.S. Nuclear Regulatory Commission (NRC) "Technical Writing Style Guide" (NUREG-0650) provides an alternative system for referencing documents in NRC publications. Instead of being listed according to the order in which they first appear in the text of a report, references in this alternative system are grouped into categories by types of documents referenced. This supplement also updates information on referencing classified and proprietary material and provides guidance on how to cite codes and standards in NRC publications.

NUREG-0652 V01 N03: FACILITIES LICENSE APPLICATION RECORD. Data As Of December 31, 1981. (White Book) * Office of Management and Program Analysis. February 1982. 88pp. 8203040435. 12125:314.

The FLAR is divided into three parts as follows:

Part I - The first part is a listing of all operating licenses, authorizations, and construction permits in effect plus applications pending.

Part II - The second part is a listing of applications that have been withdrawn or denied, and license and construction permits terminated or revoked.

Part III - The third part is a listing of research, test, and power reactors for export.

NUREG-0661 S01: SAFETY EVALUATION REPORT FOR THE MARK 1 CONTAINMENT LONG-TERM PROGRAM. Resolution Of Generic Technical Activity A-7. * Division of Safety Technology. August 1982. 19pp. 8209100232. 14767:296.

NUREG-0661 was originally issued in July 1980 with four open items which have now been resolved. The four open items consisted of the downcomer oscillation load definition and the confirmatory analyses and test programs which were intended to justify the adequacy of the load specification. The confirmatory efforts concern the assessment of compressible flow effects in the scaled pool swell tests and the confirmation of condensation oscillation load magnitude and global symmetry.

The staff has reviewed the improved downcomer condensation oscillation load definition provided by the Mark I Owners' Group and finds it acceptable. The staff has also reviewed the confirmatory experimental and analytical programs conducted by the Mark I Owners' Group and has concluded that the adequacy of the various load specifications have been justified.

This Supplement No. 1 to NUREG-0661 therefore serves as the staff's final resolution of Unresolved Safety Issue A-7.

NUREG-0698 R01: NRC PLAN FOR CLEANUP OPERATIONS AT THREE MILE ISLAND UNIT 2: Revision 1. LO,R.; SNYDER,B. TMI Program Office. February 1982. 30pp. 8204160027. 12719:238.

This NRC Plan, which defines NRC's functional role in cleanup operations at Three Mile Island Unit 2 and outlines NRC's regulatory responsibilities in fulfilling this role, is the first revision to the initial plan issued in July 1980 (NUREG-0698).

Since 1980, a number of policy developments have occurred which will have an impact on the course of cleanup operations. This revision reflects these developments in the area of NRC's review and approval process with regard to cleanup operations as well as NRC's interface with the Department of Energy's involvement in the cleanup and waste disposal. This revision is also intended to update the cleanup schedule by presenting the cleanup progress that has taken place and NRC's role in ongoing and future cleanup activities.

NUREG-0698 R01 ERR: NRC PLAN FOR CLEANUP OPERATIONS AT THREE MILE ISLAND UNIT 2. * TMI Program Office. March 16, 1982. 3pp. 8204150568. 12708:048.

This NRC Plan, which defines NRC's functional role in cleanup operations at Three Mile Island Unit 2 and outlines NRC's regulatory responsibilities in fulfilling this role, is the first revision to the initial plan issued in July 1980 (NUREG-0698).

Since 1980, a number of policy developments have occurred which will have an impact on the course of cleanup operations. This revision reflects these developments in the area of NRC's review and approval process with regard to cleanup operations as well as NRC's interface with the Department of Energy's involvement in the cleanup and waste disposal. This revision is also intended to update the cleanup schedule by presenting the cleanup progress that has taken place and NRC's role in ongoing and future cleanup activities.

NUREG-0712 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3. Docket Nos. 50-361 And 50-362. (Southern California Edison Company, et al.) * Office of Nuclear Reactor Regulation, Director. February 1982. 40pp. 8203090193. 12176:176.

Supplement No. 5 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 addresses several items that have come to light since issuance of Supplement No. 4, including an additional applicant request for relief from certain dated requirements of NUREG-0737.

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-0713 V03: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS - 1981. BROOKS, B.G. Management Information Branch. November 1982. 118pp. 8212270459. 16568:221.

This report summarizes the occupational radiation exposure information that has been reported to the U.S. N.R.C. by commercial nuclear power reactors during the years 1969 through 1981. The bulk of the data presented in the report was obtained from annual radiation exposure reports submitted in accordance with the requirements of 10 CFR 20.407 and licensee technical specifications. Data on workers terminating their employment at nuclear power facilities was obtained from reports submitted pursuant to 10 CFR 20.408. The annual reports submitted by the 71 nuclear power plants that had completed at least one full year of operation as of December 31, 1981, indicated that the number of personnel monitored during 1981 was 124,506 persons and the annual collective dose incurred by these individuals was 54,142 man-rems. The average annual dose for each worker that received a measurable dose was 0.7 rems, and the average collective dose per reactor was 773 man-rems. The termination reports revealed that some 64,500 individuals completed their employment with one or more reactor facilities during 1980.* Approximately 5,500 of these workers could be considered transients and they received an average dose of about one rem.

* The most recent year for which all of the termination data are available for analysis.

NUREG-0714 V01: OCCUPATIONAL RADIATION EXPOSURE REPORT. Twelfth Annual Report 1979. BROOKS, B.G.; MCDONALD, S.; RICHARDSON, E. Division of Data Automation & Management Information. August 1982. 108pp. 8209010444. 14716:004.

This report summarizes the information reported for calendar year 1979 by all NRC licensees to the Commission's centralized repository of personnel occupational radiation exposure information. The bulk of the information in the report is derived from annual reports that were required to be submitted by all NRC licensees pursuant to 10CFR20.407. This is the second year that all NRC licensees were required to submit an annual exposure report. Previously only certain categories - commercial nuclear power reactors, industrual radiographers, fuel fabricators and processors and commercial distributors of byproduct materials - of NRC licensees had submitted such reports. The requirement of 10CFR20.408 for the submission of termination reports continued to apply to only these four categories, and some analysis of the data contained in these reports is also presented. A brief description of personnel overexposures reported by NRC licensees is included as well.

NUREG-0717 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1. Docket No. 50-395. (South Carolina Electric and Gas Company) * Office of Nuclear Reactor Regulation, Director. January 1982. 50pp. 8201250333. 11676:021.

Supplement No. 3 to the Safety Evaluation Report related to the operation of the Virgil C. Summer Station, Unit 1 updates the information contained in the Safety Evaluation Report, dated February 1981, Supplement No. 1, dated April 1981 and Supplement No. 2 dated May 1981. This supplement also closes out previously outstanding items. Supplement No. 4 will be issued prior to plant operation expected in the spring of 1982 and will also update any outstanding issues and any new issues.

The Safety Evaluation Report and its supplements pertain to the application for a license to operate the Virgil C. Summer Nuclear Station filed by the South Carolina Electric & Gas Company on December 10, 1976.

NUREG-0717 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF THE VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1. Docket No. 50-395. (South Carolina Electric & Gas Company) * Office of Nuclear Reactor Regulation, Director. August 1982. 112pp. 8208230411. 14528:251.

The Safety Evaluation Report and its supplements pertain to the application for a license to operate the Virgil C. Summer Nuclear Station filed by the South Carolina Electric & Gas Company on December 10, 1976. The site is located in Fairfield County, South Carolina.

The Safety Evaluation Report related to operation was issued in February 1981. Supplement No. 1 containing updated information since issuance of the Safety Evaluation Report was issued in April 1981. Supplement No. 2 updating the emergency planning information was issued in May 1981. Supplement No. 3 updating previous information and closing out some outstanding items was issued in January 1982. This supplement discusses the resolution of all previously identified open items and closes them out.

NUREG-0717 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1. DOCKET No. 50-395. (South Carolina Electric And Gas Company) * Office of Nuclear Reactor Regulation, Director. November 1982. 22pp. 8212270211. 16551:138.

The Safety Evaluation Report and its supplements pertain to the application for a license to operate the Virgil C. Summer Nuclear Station filed by the South Carolina Electric & Gas Company on December 10, 1976. The site is located in Fairfield County, South Carolina.

The Safety Evaluation Report related to operation was issued in February 1981. Supplement No. 1 containing updated information since issuance of the Safety Evaluation Report was issued in April 1981. Supplement No. 2 updating the emergency planning information was issued in May 1981. Supplement No. 3 and Supplement No. 4 updated previous information and closed out formerly outstanding issues and were issued in January 1982 and August 1982, respectively. This Supplement (5) discusses operation of the station above 5 percent power.

NUREG-0725 R02: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL. * Office of Nuclear Material Safety & Safeguards, Director. June 1982. 53pp. 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-0741: TECHNICAL SPECIFICATIONS FOR SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 2. Docket No. 50-361. (Southern California Edison Company, et al.) WEINKAM, E. J. Office of Nuclear Reactor Regulation, Director. February 1982. 400pp. 8203120001. 12253:001.

The San Onofre Nuclear Generating Station, Unit 2 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-0744 V01-02 R1: RESOLUTION OF THE TASK A-11 REACTOR VESSEL MATERIALS TOUGHNESS SAFETY ISSUE. JOHNSON, R. Division of Safety Technology. October 1982. 182pp. 8211110094. 16047:125.

This report provides the NRC position with respect to the reactor pressure vessel safety analysis required according to the rules given in the Code of Federal Regulations, Title 10. An analysis is required whenever neutron irradiation reduces the Charpy V-notch upper shelf energy level in the vessel steel to 50 ft-lb or less. Task A-11 was needed because the available engineering methodology for such an analysis utilized linear elastic fracture mechanics principles, which could not fully account for the plastic deformation or stable crack extension expected at upper shelf temperatures. The Task A-11 goal was to develop an elastic-plastic fracture mechanics methodology, applicable to the beltline region of a pressurized water reactor vessel, which could be used in the required safety analysis. The goal was achieved with the help of a team of recognized experts. Part I of this volume contains the "For Comment" NUREG-1744 originally published in September 1981 and edited to accommodate comments from the public and the NRC staff. Part II of this volume contains the staff's responses to, and resolution of, the public comments received. This report completed the staff resolution of the Unresolved Safety Issue A-11, "Reactor Vessel Materials Toughness."

NUREG-0748 V02 N01: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of January 31, 1982. (Orange Book) * Office of Management and Program Analysis. February 1982. 250pp. 8203040437. 12124:001.

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 N02: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of February 28, 1982. (Orange Book) * Office of Management and Program Analysis. March 1982. 356pp. 8204150562. 12689:001.

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 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. 349pp. 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-0748 V02 N07: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of July 31, 1982. (Orange Book) * Management Information Branch. August 1982. 197pp. 8209100218. 14769:165.
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 N08: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of August 31, 1982. (Orange Book) * Management Information Branch. September 1982. 355pp. 8210120116. 15685:319.
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 N09: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of September 30, 1982. (Orange Book) * Management Information Branch. October 1982. 358pp. 8211160514. 16114:316.
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 N10: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of October 31, 1982. (Orange Book) * Management Information Branch. November 1982. 286pp. 8212160780. 16465:066.
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 N11: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of November 30, 1982. (Orange Book) * Management Information Branch. December 1982. 150pp. 8302030239. 17022:001.

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-0750 V14 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES. November 1981. Pp 933-1,089. * Division of Technical Information & Document Control. March 18, 1982. 164pp. 8203180455. 12339:089.

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

NUREG-0750 V15 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-March 1982. * Division of Technical Information & Document Control. March 1982. 72pp. 8211120037. 16064:061.

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 V15 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES. January 1982. Pp 1-224. * Division of Technical Information & Document Control. January 1982. 232pp. 8207140091. 13859:051.

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

NUREG-0750 V15 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES. February 1982. Pp 225-357. * Division of Technical Information & Document Control. February 1982. 139pp. 8207140086. 13859:283.

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

NUREG-0750 V15 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES. March 1982. Pp 359-672. * Division of Technical Information & Document Control. March 1982. 325pp. 8208130476. 14350:016.

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

NUREG-0750 V15 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES. April 1982. Pp 673-1,093. * Division of Technical Information & Document Control. April 1982. 429pp. 8209270441. 15550:028.

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

NUREG-0750 V15 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES. May 1982. Pp 1,095-1,362. * Division of Technical Information & Document Control. May 1982. 275pp. 8210150547. 15721:321.

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

NUREG-0750 V15 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES. June 1982. Pp 1,363-1,768. * Division of Technical Information & Document Control. June 1982. 411pp. 8211190313. 16166:001.

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

NUREG-0769 ADD01: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF ENRICO FERMI ATOMIC POWER PLANT, UNIT NO. 2. Docket No. 50-341. (Detroit Edison Company) * Office of Nuclear Reactor Regulation, Director. March 1982. 30pp. 8203190064. 12354:001.

The Final Environmental Statement for the Erico Fermi Atomic Power Plant related to operation was issued during August 1981. The Final Environmental Statement was the second assessment of the environmental impact associated with the construction and operation of the Fermi 2 Nuclear Power Plant, located on Lake Erie in Monroe County, Michigan. The Draft Environmental Statement was issued in April 1981. The first assessment was the Final Environmental Statement related to construction issued in July 1972 prior to issuance of the Fermi 2 construction permit.

This Addendum includes the NRC staff response to comments by the Department of the Interior that were not included in the Final Environmental Statement, dated August 1981.

NUREG-0769 ADD01 ERR: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF ENRICO FERMI ATOMIC POWER PLANT, UNIT NO. 2. Docket No. 50-341. (Detroit Edison Company) * Office of Nuclear Reactor Regulation, Director. March 23, 1982. 1p. 8204150235. 12687:345.

The Final Environmental Statement for the Erico Fermi Atomic Power Plant related to operation was issued during August 1981. The Final Environmental Statement was the second assessment of the environmental impact associated with the construction and operation of the Fermi 2 Nuclear Power Plant, located on Lake Erie in Monroe County, Michigan. The Draft Environmental Statement was issued in April 1981. The first assessment was the Final Environmental Statement related to construction issued in July 1972 prior to issuance of the Fermi 2 construction permit.

This Addendum includes the NRC staff response to comments by the Department of the Interior that were not included in the Final Environmental Statement, dated August 1981.

NUREG-0773: THE DEVELOPMENT OF SEVERE REACTOR ACCIDENT SOURCE TERMS: 1957 - 1981. BLOND, R.; TAYLOR, M.; MARGULIES, T.; et al. Office of Nuclear Regulatory Research, Director. November 1982. 116pp. 8212140476. 16431:202.

This report presents the currently available information on potential reactor accidents that has been analyzed by Probabilistic Risk Assessment (PRA) for various reactor designs and develops a group of hypothetical radioactive source terms to represent the spectrum of accidents for light water reactor designs. The set of source term estimates are fission product release fractions of the core inventory of radionuclides and release characteristics. The chosen release scenarios range from a class of relatively benign accident releases to one which represents the most severe potential consequences. These source terms are essentially independent of particular design options and may be useful for comparative siting analyses or emergency planning. The source terms given herein are based upon 1978 MARCH and CORRAL code estimates. These models have known deficiencies which would tend to give overestimates of the magnitudes of the releases. Detailed discussions of the conservatisms in the source term are given in NUREG-0771 and NUREG-0772. As such this report documents the current conservative estimates. Therefore caution should be used when applying these estimates.

NUREG-0776 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2. Docket Nos. 50-387 And 50-388. (Pennsylvania Power And Light Company) * Office of Nuclear Reactor Regulation, Director. July 1982. 174pp. 8208040193. 14233:183.

In April 1981, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-0776) regarding the application of the Pennsylvania Power & Light Company (the applicant) and the Allegheny Electric Cooperative, Inc. (co-applicant) for licenses to operate the Susquehanna Steam Electric Station, Units 1 and 2 located on a site in Luzerne County, Pennsylvania.

Supplement No. 1 was issued in June 1981 and addressed outstanding issues. Supplement No. 2 was issued in September 1981 and contains the Report of the Advisory Committee on Reactor Safeguards, dated August 11, 1981. Supplement No. 2 also contains the responses to the comments made by the Advisory Committee on Reactor Safeguards in its report.

This supplement discusses the resolution of all of the five items previously identified as open and closes them out.

NUREG-0776 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 & 2. Docket Nos. 50-387 And 50-388 (Pennsylvania Power And Light Company, Allegheny Electric Cooperative, Inc.) * Division of Licensing. October 1982. 39pp. 8212060462. 16343:311.

The Nuclear Regulatory Commission issued its Safety Evaluation Report relating to the application of the Pennsylvania Power & Light Company for operating licenses for the Susquehanna Steam Electric Station, Units 1 and 2 in April 1981. The facility is located on a site in Luzerne County, Pennsylvania.

Supplement No. 1 issued in June 1981 addressed outstanding issues. Supplement No. 2 issued September 1981 contained the Report of the Advisory Committee on Reactor Safeguards (ACRS), dated August 11, 1981. Supplement No. 3 issued in July 1982 discussed the resolution of previously identified open items and closed them out prior to issuance of an operating license for Unit No. 1. Operating License NPF-14 for Unit No. 1 was issued July 17, 1982 and restricted the power level to 5% of full power.

This Supplement addresses items contained in the 5% license that must be resolved prior to exceeding 5% power.

NUREG-0784: LONG-RANGE RESEARCH PLAN. FY 1984-1988. * Office of Nuclear Regulatory Research, Director. August 1982. 284pp. 8209270042. 15513:001.

The Long-Range Research Plan (LRRP) was prepared by the Office of Nuclear Regulatory Research (RES) to assist the NRC in coordinating its long-range research planning with the short-range budget cycles. The LRRP lays out programmatic approaches for research to help resolve regulatory issues. This year's plan reflects the new organization resulting from the consolidation of RES and the Office of Standards Development. The plan will be updated annually.

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 site is suitable for a facility of the general size and type as the CRBRP from 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-0787 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATERFORD STEAM ELECTRIC STATION, UNIT 3. Docket No. 50-302. (Louisiana Power And Light Company) * Office of Nuclear Reactor Regulation, Director. October 1982. 31PP. 8211190358. 16167:271.

Supplement No. 4 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 update the Safety Evaluation Report by providing the staff's evaluation of information submitted by the applicant since the Safety Evaluation Report and its three supplements were issued.

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: 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. October 1982. 168pp. 8210210030. 15785:015.

This report supplements the Safety Evaluation 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. Most of the open items are associated with soils-related problems at the Midland site.

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-0797 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2. Docket Nos. 50-445 and 50-446. (Texas Utilities Generating Company, et al.) * Office of Nuclear Reactor Regulation, Director. January 1982. 46pp. 8202020086. 11780:028.

Supplement No. 2 to the Safety Evaluation Report (SER) related to the operation of the Comanche Peak Steam Electric Station, Units 1 and 2, has been prepared by the Office of Nuclear Reactor Regulation of the U.S Nuclear Regulatory Commission. The facility is located in Somervell County, Texas. Subject to favorable resolution of the items identified in this supplement, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public. This document provides the NRC staff's evaluation of the outstanding and confirmatory issues that have been resolved since Supplement No. 1 was issued in October 1981 and addresses changes to the SER and Supplement No. 1, which have resulted from the receipt of additional information from the applicant. This document also addresses those items that are identified in the Advisory Committee on Reactor Safeguards letter, dated November 17, 1981.

NUREG-0798 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF ENRICO FERMI ATOMIC POWER PLANT, UNIT NO. 2. Docket No. 50-341. (Detroit Edison Company) * Office of Nuclear Reactor Regulation, Director. January 1982. 67pp. 8203030004. 12109:013.

Supplement No. 2 to the Safety Evaluation Report related to the operation of the Enrico Fermi Atomic Power Plant, Unit 2 provides the staff's evaluation of additional information submitted by the applicant regarding outstanding review issues identified in Supplement No. 1 to the Safety Evaluation Report, dated September 1981.

NUREG-0802: SAFETY/RELIEF VALVE QUENCHER LOADS: EVALUATION FOR BWR MARK II AND III CONTAINMENTS. SU, T. M. Division of Safety Technology. October 1982. 132pp. 8211030504. 15930:284.

Boiling water reactor (BWR) plants are equipped with safety/relief valves (SRVs) to protect the reactor from overpressurization. Plant operational transients, such as turbine trips, will actuate the SRV. Once the SRV opens, the air column within the partially submerged discharge line is compressed by the high-pressure steam released from the reactor. The compressed air discharged into the suppression pool produces high-pressure bubbles. Oscillatory expansion and contraction of these bubbles create hydrodynamic loads on the containment structures, piping, and equipment inside containment. This report presents the results of the staff's evaluation of SRV loads. The evaluation, however is limited to the quencher devices used in Mark II and III containments. With respect to Mark I containments, the SRV acceptance criteria are presented in NUREG-0661 issued July 1980. The staff acceptance criteria for SRV loads for Mark II and III containments are presented in this report. In conjunction with NUREG-0661, NUREG-0763, and NUREG-0783, the issuance of this report concludes NRC Unresolved Safety Issue A-39, Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWRs.

NUREG-0813: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF CALLAWAY PLANT, UNIT NO. 1. Docket No. STN 50-483. (Union Electric Company) * Office of Nuclear Reactor Regulation, Director. January 1982. 220pp. 8203010022. 12073:157.

The information in this Final Environmental Statement is the second assessment of the environmental impact associated with the construction and operation of the Callaway Plant, located in Callaway County, Missouri. The Draft Environmental Statement was issued in September 1981. The first assessment was the Final Environmental Statement related to construction issued in March 1975 prior to issuance of the Callaway 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-0816: POWER PLANT SITING AND DESIGN: INTAKE AND DISCHARGE EFFECTS AT POINT BEACH NUCLEAR PLANT ON LAKE MICHIGAN BIOTA AND FISHERIES. HICKEY, C. R. Division of Engineering. February 1982. 63pp. 8203120004. 12252:188.

The impact of the operation of the Point Beach Nuclear Plant on aquatic biota and fisheries of Lake Michigan is examined. Significant adverse impacts have not been detected. Localized effects are identified and appear to be related primarily to thermal discharges. The recreational fishery for trout and salmon is better as a result of (1) thermal plume attraction and (2) development of fishing facilities at the power plant. Angler catches averaged more than 10,000 salmonids per year. This fact should enhance the Lake Michigan fishery stocking program and is considered a benefit derived from power plant operation. The effects observed have been primarily on the exotic lake fishes (that is, alewife, smelt, and stocked salmonids) rather than on native lake species. Design features of both the offshore intake and the shoreline discharge (and the potential interaction of the two) contributed to localized effects together rather than separately. The review of the operational experience at Point Beach also showed that some station design features apparently contributed to greater effects than anticipated. This knowledge should prove useful in the feedback process from operational experience to siting and design of future power plants on Lake Michigan, and from operational experience to impact assessment and prediction.

NUREG-0820: INTEGRATED PLANT SAFETY ASSESSMENT, SYSTEMATIC EVALUATION PROGRAM. Palisades Plant. Docket No. 50-255. (Consumers Power Company) * Division of Licensing. October 1982. 531pp. 8211190341. 16170:001.

The Nuclear Regulatory Commission has published its Final Integrated Plant Safety Assessment Report (IPSAR) (NUREG-0820), under the scope of the Systematic Evaluation Program (SEP), for Consumers Power Company's Palisades Plant located in Covert, Van Buren County, Michigan. The SEP was initiated by the NRC to review the design of older operating nuclear reactor plants to reconfirm and document their safety. This report documents the review completed under the SEP for the Palisades Plant. The review has provided for (1) an assessment of the significance of differences between current technical positions on selected 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 documented evaluation of plant safety when all supplements to the Final IPSAR and the Safety Evaluation report for converting the license from a provisional to a full-term license have been issued. The report also addresses the comments and recommendations made by the Advisory Committee on Reactor Safeguards in connection with its review of the Draft Report, issued in April 1982. The Final IPSAR and its supplements will form part of the bases for considering the conversion of the existing provisional operating license to a full-term operating license.

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: INTEGRATED PLANT SAFETY ASSESSMENT REPORT-SYSTEMATIC EVALUATION PROGRAM. R. E. Ginna Nuclear Power Plant. Docket No. 50-244. (Rochester Gas & Electric Corporation) * Division of Licensing. December 1982. 511pp. 8301100023. 16748:001.

The Nuclear Regulatory Commission has published its Final Integrated Plant Safety Assessment Report (IPSAR) (NUREG-0821), under the scope of the Systematic Evaluation Program (SEP), for Rochester Gas & Electric Corporation's R. E. Ginna Nuclear Power Plant located in Wayne County, New York. The SEP was initiated by the NRC in February 1977, to review the design of older operating nuclear reactor plants to reconfirm and document their safety. This report documents the review completed under the SEP for the Ginna plant. The review has provided for (1) an assessment of the significance of differences between current technical positions on safety issues and those that existed when the Ginna plant was licensed, (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 when all supplements to the Final IPSAR and the Safety Evaluation Report for converting the license from a provisional to a full-term license have been issued. The report also addresses the comments and recommendations made by the Advisory Committee on Reactor Safeguards (ACRS) in connection with its review of the Draft Report, issued in May 1982. The Final IPSAR and its supplements will form part of the bases for considering the conversion of the existing 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-0822 DRFT: INTEGRATED PLANT SAFETY ASSESSMENT, SYSTEMATIC EVALUATION PROGRAM. Oyster Creek Nuclear Generating Station. Docket 50-219. (GPU Nuclear Corporation And Jersey Central Power And Light Company) * Division of Licensing. September 1982. 537pp. 8210270345. 15850:146.

The Systematic Evaluation Program was initiated in February 1977 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 Oyster Creek Nuclear Generating Station, owned by Jersey Central Power and Light Company and operated by GPU Nuclear Corporation located in Ocean County, New Jersey. Oyster Creek is one of ten plants reviewed under Phase II of the 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-0823 DRFT: INTEGRATED PLANT SAFETY ASSESSMENT, SYSTEMATIC EVALUATION PROGRAM. Dresden Nuclear Power Station, Unit 2. Docket No. 50-237. (Commonwealth Edison Company) * Division of Licensing. October 1982. 551pp. 8211160529. 16111:186.

The Systematic Evaluation Program was initiated in February 1977 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 Dresden Nuclear Power Station, Unit 2, owned and operated by the Commonwealth Edison Company and located in Grundy County, Illinois. Dresden Unit 2 is one of ten plants reviewed under Phase II of this program, which 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-0824 DRFT: INTEGRATED PLANT SAFETY ASSESSMENT. SYSTEMATIC EVALUATION PROGRAM. Millstone Nuclear Power Station, Unit 1. Docket No. 50-245. (Northeast Nuclear Energy Company) * Office of Nuclear Reactor Regulation, Director. November 1982. 498pp. 8212010100. 16292:001.

The Systematic Evaluation Program was initiated in February 1977 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 Millstone Nuclear Power Station, Unit 1, operated by Northeast Nuclear Energy Company located in Waterford, Connecticut. Millstone Nuclear Power Station, Unit 1 is one of ten plants reviewed under Phase II of this program. This report 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-0830 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF CALLAWAY PLANT, UNIT NO. 1. Docket No. STN 50-483. (Union Electric Company) * Office of Nuclear Reactor Regulation, Director. January 1982. 39pp. 8202020088. 11779:359.

Supplement No. 1 to the Safety Evaluation Report related to the operation of the Callaway Plant, Unit No. 1 updates the information contained in the Safety Evaluation Report, dated October 1981. This supplement also addresses the ACRS Report issued on November 17, 1981.

The Safety Evaluation Report and its supplement pertain to the application for a license to operate the Callaway Plant filed by the Union Electric Company on October 19, 1979.

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. 215pp. 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-0831 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF THE GRAND GULF NUCLEAR STATION, UNITS 1 AND 2. Docket Nos. 50-416 and 50-417. (Mississippi Power and Light). * Office of Nuclear Reactor Regulation, Director. July 1982. 45pp. 8208040131. 14236:232.

Supplement No. 3 to the Safety Evaluation Report of 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 Clairborne County, Mississippi, 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-0833 ERR: ENVIRONMENTAL IMPACT STATEMENT ON THE SITING OF NUCLEAR POWER PLANTS-SCOPING SUMMARY REPORT. * Waste Management Branch. January 19, 1982. 1p. 8202040085. 11818:07?

The NRC staff has completed its scoping process for the Environmental Impact Statement for the revision of its regulations on the siting of nuclear power plants. The rulemaking and environmental review have been focused to concentrate on significant issues and alternatives and to delete items from the rulemaking on which it is not appropriate to proceed at this time. A brief discussion of the major comments is included.

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. 169pp. 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-0837 V02 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report, January-March 1982. COSTELLO, F.; THOMPSON, T.; COHEN, L. K. Region 1, Office of Director. July 1982. 187pp. 82081[illegible]0469. 14355:318.

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 first quarter of 1982.

NUREG-0837 V02 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report, April-June 1982. COSTELLO, F.; THOMPSON, T.; COHEN, L. K. Region 1, Office of Director. November 1982. 190pp. 8301120108. 16784:032.

This report provides the status and results of the NRC ThermoluminescentDosimeter (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 quarter of 1982.

NUREG-0841: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2 AND 3. Docket Nos. STN 50-528, STN 50-529 & STN 50-530. (Arizona Public Service Company, et al.) * Office of Nuclear Reactor Regulation, Director. February 1982. 200pp. 8203190056. 12353:001.

The Final Environmental Statement related to the operation of the Palo Verde Nuclear Generating Station, Units 1, 2 and 3 by Arizona Public Service Company, et al (Docket Nos. STN 50-528, STN 50-529 and STN 50-530), located in Maricopa County, Arizona, 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 operating licenses could be granted.

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-0845: AGENCY PROCEDURES FOR THE NRC INCIDENT RESPONSE PLAN. * Division of Emergency Preparedness (Pre 830103). March 1982. 200pp. 8203290061. 12443:001.

The NRC Incident Response Plan describes the function of the NRC during an incident and the kinds of actions that comprise an NRC response. The NRC response plan will be activated in accordance with threshhold criteria described in the plan for incidents occurring at nuclear reactors, fuel facilities and materials licensees, during transportation of licensed material, and for threats against facilities or licensed material. In contrast to the general overview provided by the Plan, the purpose of these agency procedures is to delineate:

1. The manner in which each planned response function is performed;

2. The criteria for making those response decisions which can be preplanned;

3. The information and other resources needed during a response.

An inexperienced but qualified person should be able to develop the ability to perform functions assigned by the Plan and make necessary decisions, given the specified information, by becoming familiar with these procedures. This rule of thumb has been used to determine the amount of detail in which the agency procedures are described. These procedures form a foundation for the training of response personnel both in their normal working environment and during planned emergency exercises. These procedures also form a ready reference or reminder checklist for technical team members and managers during a response.

NUREG-0846: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE DECOMMISSIONING OF EDGEMONT URANIUM MILL. Docket No. 40-1341. (Tennessee Valley Authority) * Office of Nuclear Material Safety & Safeguards, Director. June 1982. 320pp. 8210050422. 15623:074.

A Final Environmental Statement (FES) related to the proposed decommissioning of the existing uranium milling facilities at Edgemont, South Dakota (Docket 40-1341) including removal or cleanup of contaminated soil from the mill site and local environs. This statement describes and evaluates (1) purpose of and need for action, (2) alternative methods of tailings disposal, (3) alternative tailings disposal sites, and (4) environmental consequences for the proposed action. Also included are comments of governmental agencies and other organizations on the Draft Environmental Statement for this project, and staff responses to their comments. The NRC has concluded that the action called for under the National Envionmental Policy Act of 1969 (NEPA) and 10 CFR Part 51 is to permit the applicant to proceed with the project as described in this statement, subject to at least certain conditions as stated in the Summary and Conclusions of the DES.

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. 512pp. 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-0847 S01: 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. September 1982. 29pp. 8210050405. 15625:299.

This report supplements the Safety Evaluation Report, NUREG-0847, issued in June 1982 by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by the Tennessee Valley Authority, as applicant and owner, for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391). The facility is located in Rhea County, Tennessee, near the Watts Bar Dam on the Tennessee River. 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 report dated August 16, 1982.

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-0853: SAFETY EVALUATION REPORT 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. February 1982. 250pp. 8203020168. 12091:001.

This Safety Evaluation Report for the application filed by Illinois Power Company, Soyland Power Cooperative, Inc. and Western Illinois Power Cooperative, Inc. as applicants and owners, for a license to operate the Clinton Power Station, Unit 1 (Docket No. 50-461), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Harp Township, DeWitt County, Illinois. Subject to favorable resolution of the items discussed in this report, the staff concludes that the facility can be operated by the Illinois Power Company without endangering the health and safety of the public.

NUREG-0853 S01: SAFETY EVALUATION REPORT 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. July 1982. 84pp. 8208130470. 14351:001.

Supplement No. 1 to the Safety Evaluation Report on the application filed by Illinois Power Company, Soyland Power Cooperative, Inc., and Western Illinois Power Cooperative, Inc., as applicants and owners, for a license to operate the Clinton Power Station, Unit No. 1 has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Harp Township, Dewitt County, Illinois. This supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report.

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-0855: HEALTH PHYSICS APPRAISAL PROGRAM. CUNNINGHAM, L. J.; WIGGINTON, J. E.; FLACK, E. D. Director's Office, Office of Inspection and Enforcement. March 1982. 108pp. 8204150578. 12686:267.

The accident at Three Mile Island in March 1979 and subsequent investigations identified, among other items, serious concerns involving several aspects of the radiation protection program. Significantly, some concerns involved areas not addressed by regulations or facility technical specifications. This in turn led to initiation of a major effort to evaluate the adequacy and effectiveness of radiation protection programs at all currently operating nuclear power facilities during calendar year 1980 by the Office of Inspection and Enforcement (IE), Nuclear Regulatory Commission. This inspection effort was termed an appraisal since it was structured to facilitate an integrated look at the total radiation protection program, delve into matters for which explicit regulatory requirements did not exist, and emphasized evaluation of capability and performance rather than compliance with regulations. This report discusses the results of the 48 appraisals and the anticipated regulatory actions that may be taken to further address the concerns.

NUREG-0857: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PALO VERDE NUCLEAR GENERATING STATION, UNITS 1,2 AND 3. Docket Nos. STN 50-528, STN 50-529 And STN 50-530. (Arizona Public Service Company, et al.) * Office of Nuclear Reactor Regulation, Director. February 1982. 50pp. 8203020137. 12090:114.

Supplement No. 1 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. STN 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 by providing (1) the evaluation of additional information submitted by the applicant since the Safety Evaluation Report was issued, (2) the evaluation of the matters the staff had under review when the Safety Evaluation Report was issued, and (3) the response to comments made by the Advisory Committee on Reactor Safeguards.

NUREG-0857 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PALO VERDE NUCLEAR GENERATING STATION, UNITS 1,2 AND 3. Docket Nos. STN 50-528, STN 50-529 And STN 50-530. (Arizona Public Service Company, et al.) * Office of Nuclear Reactor Regulation, Director. February 1982. 50pp. 8203020137. 12090:114.

Supplement No. 1 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. STN 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 by providing (1) the evaluation of additional information submitted by the applicant since the Safety Evaluation Report was issued, (2) the evaluation of the matters the staff had under review when the Safety Evaluation Report was issued, and (3) the response to comments made by the Advisory Committee on Reactor Safeguards.

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-0857 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PALO VERDE NUCLEAR GENERATING STATION, UNITS 1,2, AND 3. Docket Nos. 50-528, 50-529, And 50-530. (Arizona Public Service Company, et al.) * Office of Nuclear Reactor Regulation, Director. September 1982. 38pp. 8210060006. 15636:268.

Supplement No. 3 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. STN 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 applicants since Supplement No. 2 was issued, and (2) the evaluation of the matters that the staff had under review when Supplement No. 2 was issued.

NUREG-0858: COMPARISON BETWEEN FIELD DATA AND ULTIMATE HEAT SINK COOLING POND AND SPRAY POND MODELS. CODELL, R.B. Division of Engineering. September 1982. 67pp. 8210050431. 15626:108.

Two previously published reports, NUREG-0693 and NUREG-0733, presented models and methods by which ultimate heat sink cooling ponds and spray ponds used for safety-related water supplies in nuclear power plants could be analyzed for design-basis conditions of heat load and meteorology. These models were only partially verified with field data. The present report compares the NRC models to data collected for NRC by Battelle Pacific Northwest Laboratories on the performance of small geothermally heated ponds and spray ponds. These comparisons generally support the conclusion that the NRC models are useful tools in predicting ultimate heat sink performance.

NUREG-0859: COMPLIANCE DETERMINATION PROCEDURES FOR ENVIRONMENTAL RADIATION PROTECTION STANDARDS FOR URANIUM RECOVERY FACILITIES 40CFR PART 190. * Office of Nuclear Material Safety & Safeguards, Director. * Office of Nuclear Regulatory Research, Director. March 1982. 21pp. 8204150576. 12687:124.

Uranium Milling operations are licensed by the Nuclear Regulatory Commission and by some States in agreement with the Commission. The radiation dose to any individual from the operation of facilities within the uranium fuel cycle is limited to levels set by the Environmental Protection Agency. These levels are contained in the EPA Environmental Radiation Protection Standards for Nuclear Power Operations, in Part 190 of Title 40 of the Code of Federal Regulations (40 CFR Part 190). This report describes the procedures used within

NRC's Uranium Recovery Licensing Branch for evaluating compliance with these regulations for uranium milling operations. The report contains descriptions of these procedures, dose factors for evaluating environmental measurement data, and guidance to the NRC staff reviewer.

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-0862 I02: SPECIAL INSPECTION OF "PRELIMINARY REPORT, SEISMIC REVERIFICATION PROGRAM" AT DIABLO CANYON UNITS 1 AND 2. Docket Nos. 50-275 and 50-323. (Pacific Gas and Electric Company) FAULKENBERRY, B.; SHACKLETON, D. C. Reactor Construction Projects Branch. January 18, 1982. 618pp. 8202160053. 11946:001.

This report covers phase one of an investigation directed by Mr. William J. Dircks, Executive Director for Operations of the NRC, into the circumstances surrounding the development of the "Preliminary Report, Seismic Reverification Program" prepared by R.L. Cloud Associates, Inc. (R.L. Cloud), for PG&E. Statements made at a meeting on November 3, 1981 between representatives from PG&E and the NRC led the NRC to believe that no circulation of the results of the Diablo Canyon seismic reverification study by R.L. Cloud had taken place preliminary to the draft report submitted to the NRC on November 18, 1981. In late November and early December 1981, the NRC received information that draft reports of the results of the R.L. Cloud reverification study were circulated within PG&E prior to submittal to the NRC. NRC Region V initiated a special investigation on December 16, 1981, to obtain all pertinent facts related to: (1) the statements made at the November 3, 1981 meeting, and (2) PG&E's reviews and comments on draft reports of the results of the R.L. Cloud study prior to a draft report being submitted to the NRC on November 18, 1981.

NUREG-0862 I03: SPECIAL INSPECTION OF "PRELIMINARY REPORT, SEISMIC REVERIFICATION PROGRAM" AT DIABLO CANYON UNITS 1 AND 2. Docket Nos. 50-275 And 50-323. (Pacific Gas And Electric Company) FAULKENBERRY, B.; MORRILL, P. J.; FAIR, J. R.; et al. Division of Resident, Reactor Project & Engineering Programs. February 1982. 640pp. 8203040080. 12121:001.

This report covers phase two of an investigation directed by Mr. William J. Dircks, Executive Director for Operations of the NRC, into the circumstances surrounding the development of the "Preliminary Report, Seismic Reverification Program" prepared by R. L. Cloud Associates, Inc. (R. L. Cloud), for PG&E. Statements made at a meeting on November 3, 1981 between representatives from PG&E and the NRC led the NRC to believe that no circulation of the results of the Diablo Canyon seismic reverification study by R. L. Cloud had taken place preliminary to the draft report submitted to the NRC on November 18, 1981. In late November and early December 1981, the NRC received information that draft reports of the results of the R. L. Cloud reverification study were circulated within PG&E prior to submittal to the NRC. NRC Region V initiated a special investigation on December

16, 1981, to obtain all pertinent facts related to: (1) the statements made at the November 3, 1981 meeting, and (2) PG&E's reviews and comments on draft reports of the results of the R. L. Cloud study prior to a draft report being submitted to the NRC on November 18, 1981.

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-0864: REVIEW AND EVALUATION OF THE NUCLEAR REGULATORY COMMISSION SAFETY RESEARCH PROGRAM FOR FISCAL YEAR 1983. * ACRS - Advisory Committee on Reactor Safeguards. February 1982. 78pp. 8203030226. 12119:137.

Public Law 95-209 includes a requirement that the Advisory Committee on Reactor Safeguards submit an annual report to Congress on the safety research program of the Nuclear Regulatory Commission. This report presents the results of the ACRS review and evaluation of the NRC safety research program for Fiscal Year 1983. The report contains a number of comments and recommendations.

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-0870: FY 1983 BUDGET ESTIMATES. * Division of Budget. January 1982. 100pp. 8203040436. 12125:153.

Fiscal Year 1983 Budget Justifications to Congress. The budget estimates for salaries and expenses for FY 1983 provide for obligations of $479,500,000 to be funded in total by a new appropriation.

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-0871 V01 N03: SUMMARY INFORMATION REPORT. Data As Of June 30, 1982. (Brown Book) * Office of Resource Management, Director. July 1982.

51pp. 8208180212. 14389:233.

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-0871 V01 N04: SUMMARY INFORMATION REPORT. July 1 - September 30, 1982. (Brown Book). * Office of Resource Management, Director. November 1982. 52pp. 8212220143. 16527:102.

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. 96pp. 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-0872: STUDY OF USING LICENSEE EVENT REPORTS FOR A STATISTICAL ASSESSMENT OF THE EFFECT OF OVERTIME AND SHIFT WORK ON OPERATOR ERROR, FINAL REPT. DISALVO, R.; GERY, A.; PITTMAN, J. Division of Facility Operations. July 1982. 104pp. 8208040281. 14229:195.

A study 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-0873: A BAYESIAN ANALYSIS OF DIESEL GENERATOR FAILURE DATA. VESELY, W. E.; NIYOGI, P. K.; GOLDBERG, F. F.; et al. Division of Risk Analysis. January 1982. 48pp. 8201220034. 11659:140.

A simple Bayesian approach has been developed to evaluate failure rate implications from the number of failures and number of successes in a given number of diesel tests. For the Bayesian approach, the diesel is modeled as having a constant probability of failure per trial which is unknown and whose possible values are describable by a probability distribution. The approach utilizes discrete probability distributions (probability mass functions) for ease of implementation.

As a potential tool for the analyst, a computer code has been written to efficiently calculate the diesel posterior failure rate distributions for any input diesel test data and assumed prior distribution. The code can be used to monitor diesel tests for up-to-date failure rate implications. In addition, a wide variety of sensitivity analysis can be performed using the code.

NUREG-0875: COMMENTS ON THE NRC SAFETY RESEARCH PROGRAM BUDGET FOR FISCAL YEARS 1984 AND 1985. * ACRS - Advisory Committee on Reactor Safeguards. July 1982. 31pp. 8208170312. 14372:024.

Recommendations of the Advisory Committee on Reactor Safeguards are presented to the Commissioners for their consideration for FY 84 and 85 budget for the NRC safety research program.

NUREG-0876: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BYRON STATION, UNITS 1 AND 2. Docket Nos. 50-454 And 50-455. (Commonwealth Edison Company) * Office of Nuclear Reactor Regulation, Director. February 1982. 350pp. 8203050479. 12143:001.

This report provides the results of the NRC staff review of Commonwealth Edison Company's application for licenses to operate the Byron Station, Units 1 and 2. The facility is located near the Rock River in Rockvale Township, Ogle County, Illinois, about three miles south of Byron and 17 miles southwest of Rockford, Illinois. Subject to favorable resolution of the items discussed in the Safety Evaluation Report, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

NUREG-0876 S01: SAFETY EVALUATION REPORT 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. March 1982. 30pp. 8205060186. 12999:001.

Supplement No. 1 to the Safety Evaluation Report of Commonwealth Edison Company's application for licenses to operate the Byron station, Units 1 and 2, located in Rockvale Township, Ogle County, Illinois, 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.

NUREG-0877: FINAL ENVIRONMENTAL STATEMENT RELATED TO LICENSE RENEWAL AND POWER INCREASE FOR THE NATIONAL BUREAU OF STANDARDS REACTOR. Docket No. 50-184. * Division of Licensing. August 1982. 98pp. 8209230590. 14990:172.

This Final Environmental Statement contains an assessment of the environmental impact associated with renewal of Operating License TR-5 for the National Bureau of Standards (NBS) reactor for a period of 20 years at a power level of 20 MW. This reactor is located on the 575-acre NBS site near Gaithersburg in Montgomery County, Maryland, about 20 miles northwest of the center of Washington, D.C. The reactor is a high-flux heavy-water-moderated, cooled and reflected test reactor, which first went critical on December 7, 1967. Though the reactor was originally designed for 20-MW operation, it has been operating for 14 years at a maximum authorized power level of 10 MW. Program demand is now great enough to warrant operation at a power level of 20 MW. No additional major changes to the physical plant are required to operate at 20 MW.

NUREG-0878: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF THE WOLF CREEK GENERATING STATION, UNIT NO. 1. Docket No. 50-482. (Kansas Gas and Electric Company, et al.) * Office of Nuclear Reactor Regulation, Director. January 1982. 100pp. 8201250329. 11675:205.

This draft environmental statement contains the second assessment of the environmental impact associated with the operation of the Wolf

Creek Generating Station, Unit No. 1, pursuant to the National Environmental Policy Act of 1969 (NEPA) and 10CFR Part 51, as amended, of the NRC's regulations. This statement examines: the purpose and need for the Wolf Creek Generating Station; alternatives to the 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 Wolf Creek Generating Station.

Comments should be filed no later than 45 days after the date on which the Environmental Protection Agency notice of availability of this draft environmental statement is published in the "Federal Register".

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-0879: ENVIRONMENTAL ASSESSMENT FOR THE BARNWELL LOW-LEVEL WASTE DISPOSAL FACILITY. * Division of Waste Management. January 1982. 268pp. 8201270386. 11713:135.

This Environmental Assessment was prepared by the U.S. Nuclear Regulatory Commission in response to a request by the South Carolina Department of Health and Environmental Control, Bureau of Radiological Health, for technical assistance in evaluating the impacts of the Barnwell facility. Alternatives in the following areas were considered in the assessment: (1) No Action; (2) Operational Procedures; (3) Financial Guarantees for Compliance, and; (4) Methods of Minimalizing the Potential for Radionuclide Release and Transport over the Long Term. Environmental impacts were evaluated for the following actions: (1) Closing the site and implementing site closure and stabilization plans; (2) Continuing operations under current conditions, and; (3) Continuing operations under altered conditions.

NUREG-0880 FC: SAFETY GOALS FOR NUCLEAR POWER PLANTS: A DISCUSSION PAPER. * Office of Policy Evaluations. February 1982. 45pp. 8203030199. 12110:232.

This report includes a proposed policy statement on safety goals for nuclear power plants published by the Commission for public comment and supporting duscussion paper. Proposed qualitative goals and associated numerical guidelines for nuclear power-plant accident risks are presented. The significance of the goals and guidelines, their bases and rationale, and their proposed mode of implementation are discussed.

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-0881 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WOLF CREEK GENERATING STATION, UNIT 1. Docket No. 50-482. (Kansas Gas and Electric Company, et al.) * Office of Nuclear Reactor Regulation, Director. August 1982. 30pp. 8209130281. 14781:326.

Supplement No. 1 to the Safety Evaluation Report related to operation of the Wolf Creek Generating Station, Unit No. 1 updates the information contained in the Safety Evaluation Report, dated April 1982. This supplement also addresses the ACRS Report issued May 11, 1982.

The Safety Evaluation Report and its supplement pertain to the application for a license to operate the Wolf Creek Generating Station, Unit No. 1 filed by the Kansas Gas and Electric Company on February 19, 1980. The Construction Permit, CPPR-147 was issued on May 17, 1977. The facility is located in Coffey County, Kansas.

NUREG-0882: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE RESEARCH REACTOR AT ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE. Docket No. 50-170. * Office of Nuclear Reactor Regulation, Director. January 1982. 150pp. 8203010031. 12072:071.

This Safety Evaluation Report for the application filed by the Armed Forces Radiobiology Research Institute (AFRRI), Defense Nuclear Agency for a renewal of License R-84 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 by an agency of the U.S. Department of Defense, and is located on the grounds of the National Naval Medical Center, Bethesda, Montgomery County, Maryland. Subject to favorable resolution of one outstanding item discussed in this report, the staff concludes that the facility can continue to be operated by AFRRI without endangering the health and safety of the public.

NUREG-0883: URANIUM MILL APPRAISAL PROGRAM. EVERETT, R. J.; CAIN, C. L. Region 4, Office of Director. August 1982. 33pp. 8208260429. 14589:277.

This report describes the results of special team appraisals at NRC-licensed uranium mills in the period May to November 1981. Since the Three Mile Island accident, NRC management has instituted a program of special team appraisals of radiation protection programs at certain NRC licensed facilities. These appraisals were designed to identify weaknesses and strengths in NRC licensed programs, including those areas not covered by explicit regulatory requirements. The regulatory requirements related to occupational radiation protection and environmental monitoring at uranium mills have been extensively upgraded in the past few years. In addition, there was some NRC staff concern with respect to the effectiveness of NRC licensing and inspection programs. In response to this concern and to changes in mill requirements, the NRC staff recommended that team appraisals be conducted at mills to determine the adequacy of mill programs, the effectiveness of the new requirements, and mill management implementation of programs and requirements. This report describes the appraisal scope and methodology as well as summary findings and conclusions. Significant weaknesses identified during the mill appraisals are discussed as well as recommendations for improvement in uranium mill programs and mill licensing and inspection.

NUREG-0884: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF PERRY NUCLEAR POWER PLANT, UNITS 1 AND 2. Docket Nos. STN 50-440 & STN 50-441. (Cleveland Electric Illuminating Company) * Office of Nuclear Reactor Regulation, Director. March 1982. 150pp. 8203310062. 12462:117.

The information in this statement is the second assessment of the environmental impact associated with the construction and operation of the Perry Nuclear Power Plant, Units 1 and 2, located on Lake Erie in Lake County, about 11 km (7 miles) northeast of Painesville, Ohio. The first assessment was the Final Environmental Statement related to the construction issued in April 1974, prior to the issuance of the construction permits. The present assessment is the result of the NRC staff review of the activities associated with the proposed operation of the station.

NUREG-0884: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF PERRY NUCLEAR POWER PLANT, UNITS 1 AND 2. Docket Nos. 50-440 And 50-441. (Cleveland Electric Illuminating Company) * Office of Nuclear Reactor Regulation, Director. August 1982. 242pp. 8209100217. 14768:015.

The information in this Final Environmental Statement is the second assessment of the environmental impact associated with the construction and operation of the Perry Nuclear Power Plant, Units 1 and 2, located on Lake Erie in Lake County, about 11 km (7 miles) northeast of Painsville, Ohio. The first assessment was the Final Environmental Statement related to the construction of the plant issued in April 1974, prior to issuance of the construction permits (CPPR-148 and CPPR-149). Plant construction for Unit 1 is currently about 83% complete, and Unit 2 about 43% complete. Fuel loading for Unit 1 is currently estimated by the licensee (Cleveland Electric Illuminating Company) for November 1983, with Unit 2 fuel loading scheduled for May 1987. The present assessment is the result of the NRC staff review of the activities associated with the proposed operation of the plant.

NUREG-0886: STEAM GENERATOR TUBE EXPERIENCE. CHENG, C. Y. Division of Licensing. February 1982. 67pp. 8203040167. 12125:248.

This report provides information pertaining to the status of PWR steam generator tube experience and the resolution of unresolved safety issues A-3, A-4, and A-5 regarding steam generator tube integrity. It provides an overview of the types of problems which have occurred in PWR steam generators with particular emphasis on recent operating experience. The report also discusses short and long-term corrective actions being pursued by the industry to resolve these problems, steam generator inspection and repair requirements which have been established to ensure the continued safe operation of PWR steam generators, and occupational radiation exposures associated with the above-listed activities. It should be noted that information included in this report represents the current NRC staff understanding of each issue. This report is intended to be a followup to the similar reports, NUREG-0523 and NUREG-0571, which discuss tube operating experience with the recirculation ("U" tube) type and once-through type steam generators designed by Westinghouse and Combustion Engineering, and Babcock and Wilcox, respectively.

NUREG-0886 ERR: STEAM GENERATOR TUBE EXPERIENCE. * Division of Licensing. March 2, 1982. 4pp. 8203180563. 12343:113.

This report provides information pertaining to the status of PWR steam generator tube experience and the resolution of unresolved safety issues A-3, A-4, and A-5 regarding steam generator tube integrity. It provides an overview of the types of problems which have occurred in PWR steam generators with particular emphasis on recent operating experience. The report also discusses short and long-term corrective actions being pursued by the industry to resolve these problems, steam generator inspection and repair requirements which have been established to ensure the continued safe operation of PWR steam generators, and occupational radiation exposures associated with the above-listed activities. It should be noted that information included in this report represents the current NRC staff understanding of each issue. This report is intended to be a followup to the similar reports, NUREG-0523 and NUREG-0571, which discuss tube operating experience with the recirculation ("U" tube) type and once-through type steam generators designed by Westinghouse and Combustion Engineering, and Babcock and Wilcox, respectively.

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-0887 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PERRY NUCLEAR POWER PLANT, UNITS 1 AND 2. Docket Nos. 50-440 And 50-441. (Cleveland Electric Illuminating Company) * Office of Nuclear Reactor Regulation, Director. August 1982. 43pp. 8208230422. 14518:224.

Supplement No. 1 to the Safety Evaluation Report on the application filed by the Cleveland Electric Illuminating Company on behalf of itself and as agent for 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 441). The facility is located near Lake Erie in Lake County, Ohio. This supplement has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission and reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report.

NUREG-0889: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF SAND ROCK MILL PROJECT. Docket No. 40-8743. (Conoco, Incorporated) * Office of Nuclear Material Safety & Safeguards, Director. March 1982. 306pp. 8204150574. 12690:001.

This Draft Environmental Impact Statement is issued by the U.S. Nuclear Regulatory Commission in response to the request by Conoco, Inc. for the issuance of an NRC Source and Byproduct Material License authorizing operation of the proposed Sand Rock Mill Project. 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-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-0892: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WPPSS NUCLEAR PROJECT NO. 2. Docket No. 50-397. (Washington Public Power Supply System, et al.) * Office of Nuclear Reactor Regulation, Director. March 1982. 500pp. 8204160048. 12717:007.

The Safety Evaluation Report for the application filed by Washington Public Power Supply System for a license to operate the Washington Public Power Supply System Nuclear Project No. 2 located in Richland, Washington, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. Subject to favorable resolution of the items discussed in the Safety Evaluation Report, the staff concludes that the plant can be operated by the

Washington Public Power Supply System without endangering the health and safety of the public.

NUREG-0892 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WPPSS NUCLEAR PROJECT NO. 2. Docket No. 50-397. (Washington Public Power Supply System) * Office of Nuclear Reactor Regulation, Director. August 1982. 243pp. 8208260427. 14592:007.

Supplement No. 1 to the Safety Evaluation Report on the application filed by Washington Public Power Supply System for a license to operate the WPPSS Nuclear Project No. 2, located in Richland, Washington, 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-0892 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WPPSS NUCLEAR PROJECT NO. 2. Docket No. 50-397. (Washington Public Power Supply System) * Office of Nuclear Reactor Regulation, Director. December 1982. 31pp. 8301100010. 16714:327.

Supplement No. 2 to the Safety Evaluation Report on the application filed by Washington Public Power Supply System for a license to operate the WPPS Nuclear Project No. 2, located in Richland, Washington, 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 and Supplement No. 1.

NUREG-0893: THE EFFECTS OF NATURAL PHENOMENA ON THE BATTELLE MEMORIAL INSTITUTE BUILDING JN-18 FACILITIES AT WEST JEFFERSON, OHIO. Docket No. 70-8. * Division of Fuel Cycle & Material Safety. March 1982. 43pp. 8204010537. 12489:208.

An analysis of the effects of natural phenomena on the Battelle Memorial Institute at West Jefferson, Ohio has been prepared by the Office of Nuclear Material Safety and Safeguards. The analysis is in support of the special nuclear material license held by the subject company. It addresses the probable effects of damage to the Battelle Plant by severe weather and earthquake and expresses the consequences of damage as dose to several human receptors. The doses that result from facility damage are multiplied by the occurrence rate for the initiating event yielding the yearly risk.

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-0895: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF SEABROOK STATION, UNITS 1 AND 2. Docket Nos. 50-443 And 50-444. (Public Service Company of New Hampshire, et al.) * Office of Nuclear Reactor Regulation, Director. December 1982. 524pp. 8301100042. 16750:001.

The Final Environmental Statement related to the operation of the Seabrook Station, Units 1 and 2 by Public Service Company of New Hampshire, et al (Docket Nos. 50-443 and 50-444), located in the town of Seabrook, New Hampshire, 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 governments local 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 operating licenses could be granted.

NUREG-0899: GUIDELINES FOR THE PREPARATION OF EMERGENCY OPERATING PROCEDURES. Resolution of Comments on NUREG-0799. * Division of Human Factors Safety. August 1982. 51pp. 8209270234. 15513:286.

The purpose of this document is to identify the elements necessary for utilities to prepare and implement a program of Emergency Operating Procedures (EOPs) for use by control room personnel to assist in mitigating the consequences of a broad range of accidents and multiple equipment failures. This document applies only to the EOPs so designated; it does not address emergency preparedness or emergency planning. It also represents the resolution of comments on NUREG-0799, "Draft Criteria for Preparation of Emergency Operating Procedures."

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-0908: ACCEPTANCE CRITERIA FOR THE EVALUATION OF NUCLEAR POWER REACTOR SECURITY PLANS. * Power Reactor Safeguards Licensing Branch. August 1982. 57pp. 8209270096. 15516:132.

This guidance document contains acceptance criteria to be used in the NRC license review process. It contains specific criteria for use in evaluating the adequacy of nuclear power reactor security programs as detailed in security plans.

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-0910: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE. * Division of Technical Information & Document Control. July 1982. 190pp. 8208180245. 14390:001.

In compliance with statutory requirements set forth in Title 44 U.S. Code, "Public Printing and Documents," and in the applicable regulations cited in Title 41 Code of Federal Regulations, "Public Contracts and Property Management," Chapter 101, Subchapter B, "Archives and Records," the U.S. Nuclear Regulatory Commission submitted to the General Services Administration, National Archives and Records Services, and to the Comptroller General a schedule (commonly referred to as a disposition or retention schedule) proposing the appropriate duration of retention and the final disposition for records created or maintained by the NRC.

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-0912: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE WORCESTER POLYTECHNIC INSTITUTE OPEN POOL TRAINING REACTOR.Docket No. 50-134. * Division of Licensing. December 1982. 70pp. 8301100008. 16714:256.

This Safety Evaluation Report for the application filed by the Worcester Polytechnic Institute for a renewal of operating license number R-61 to continue to operate their 10 KW open-pool training 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 Worcester Polytechnic Institute and is located on the

WPI campus in Worcester, Worcester County, Massachusetts. The staff concludes that the reactor facility can continue to be operated by WPI 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-0914: RADIOLOGICAL CONTAINMENT HANDBOOK. * Chemical Engineering Branch. October 1982. 126pp. 8211160468. 16115:314.

This document is to be used as a reference text. It is meant to be used by working personnel as a guide for using temporary radiological containments.

NUREG-0915: A CRITERION FOR THE ONSET OF QUENCH FOR LOW FLOW REFLOOD. HSU, Y. Y.; YOUNG, M. W. Division of Accident Evaluation. July 1982. 31pp. 8208130477. 14351:086.

This study provides a criterion for the onset of quench for low flow reflood. This criterion, which is a combination of two conditions, was obtained by examining temperature data from tests simulating PWR reflood, such as FLECHT, THTF, PBF, CCTF, and FEBA tests, with void fraction data from CCTF, FEBA, and FLECHT low flood tests. The data show that quenching initiated at a void fraction of 0.95 and that the majority of quench occurred at void fractions near 0.85. The results show that rods can be completely quenched by entrained droplets even if the collapsed liquid level does not advance. A thorough discussion of the analysis which supports this quench criterion is given in the rest of this report.

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-0917: NUCLEAR REGULATORY COMMISSION STAFF COMPUTER PROGRAMS FOR USE WITH METEOROLOGICAL DATA. SNELL,W.G. Division of Systems Integration (post 811005). July 1982. 145pp. 8208260068. 14587:153.

The Nuclear Regulatory Commission (NRC) receives hour-by-hour meteorological data on magnetic tape in a format specified in Regulatory Guide 1.70, Rev. 2, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (September 1975). The purpose of this report was to document the computer programs that are used by the NRC meteorology staff to examine, assess and utilize these hourly values of meteorological data. A description of each of the programs is given along with the input requirements, discussion of output, subroutine flow chart, a description of each subroutine, sample output and a program listing.

NUREG-0918: PREVENTION AND MITIGATION OF STEAM GENERATOR WATER HAMMER EVENTS IN PWR PLANTS. ANDERSON,N.R.; HAN,J.T. Division of Safety Technology. November 1982. 39pp. 8212270181. 16551:162.

Water hammer in nuclear power plants is an unresolved safety issue under study at the NRC (USI A-1). One of the identified safety concerns is steam generator water hammer (SGWH) in pressurized-water reactor (PWR) plants. This report presents a summary of: (1) the causes of SGWH, (2) various fixes employed to prevent or mitigate SGWH, and (3) the nature and status of modifications that have been made at each operating PWR plant. The NRC staff considers that the issue of SGWH in top feedring designs has been technically resolved. This report does not address technical findings relevant to water hammer in preheat type steam generators.

NUREG-0920: US NUCLEAR REGULATORY COMMISSION 1981 ANNUAL REPORT. MAHER,W.J.; DICKSON,J.H. Office of Resource Management, Director. July 1982. 224pp. 8208040222. 14234:001.

This report addresses all NRC activities, policies, and decisions made during the reporting period; complete with illustrations, charts, and treatment of technical material in lay language for consumption by the lay public.

NUREG-0921: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF CATAWBA NUCLEAR STATION,UNITS 1 AND 2.Docket Nos. 50-413 And 50-414.(Duke Power Company) * Office of Nuclear Reactor Regulation, Director. August 1982. 198pp. 8208230415. 14518:026.

This Draft Environmental Statement contains the second assessment of the environmental impact associated with the operation of the Catawaba Nuclear Station, Units 1 and 2, pursuant to the National Environmental Policy Act of 1969 (NEPA) and 10 CFR 51, as amended, of the NRC regulations. This statement examines: the affected environment, environmental consequences and mitigating actions, and environmental and economic benefits and costs. Land use and terrestrial and aquatic-ecological impacts will be small. Operational impacts to historic and archaeological sites will be negligible. The effects of routine operations, energy transmission, and periodic maintenance of rights-of-way and transmission facilities should not jeopardize any populations of endangered or threatened species. No significant impacts are anticipated from normal operational releases of radioactivity. The risk associated with accidental radiation exposure is very low. The net socioeconomic effects of the project will be beneficial. The action called for is the issuance of operating licenses for Catawba Nuclear Station, Units 1 and 2.

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. 228pp. 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. 511pp. 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-0928: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE UNIVERSITY OF VIRGINIA OPEN-POOL RESEARCH REACTOR. Docket No. 50-062. * Division of Licensing. September 1982. 77pp. 8210050377. 15626:030.

This Safety Evaluation Report for the application filed by the University of Virginia for a renewal of Operating License R-66 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 by the University of Virginia and is located on the campus in Charlottesville, Virginia. Based on its technical review, the staff concludes that the reactor facility can continue to be operated by the University without endangering the health and safety of the public or endangering the environment.

NUREG-0931: TECHNICAL SPECIFICATIONS FOR SUSQUEHANNA STEAM ELECTRIC STATION, UNIT No 1. App A to License NPF-14. Docket No 50-387. (Pennsylvania Power and Light) * Office of Nuclear Reactor Regulation, Director. July 1982. 487pp. 8208040344. 14222:001.

Susquehana Steam Electric 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-0932: TECHNICAL SPECIFICATIONS. VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1. Docket No. 50-395 (South Carolina Electric & Gas Company) * Office of Nuclear Reactor Regulation, Director. August 1982. 521pp. 8209020105. 14727:212.

The Virgil C. Summer Nuclear Station, Unit No. 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-0936 V01 N02: NRC REGULATORY AGENDA. Quarterly Report, April-June 1982. * Division of Rules and Records. July 1982. 197pp. 8208230002. 14507:330.

The NRC Regulatory Agenda is a compilation of all rules on which the NRC has proposed or is considering action and all petitions for rulemaking which have been received by the Commission and are pending disposition by the Commission. The Regulatory Agenda is updated and issued each quarter. The Agendas for April and October are published in their entirety in the Federal Register while a notice of availability is published in the Federal Register for the January and July Agendas.

NUREG-0936 V01 N03: NRC REGULATORY AGENDA. Quarterly Report, July-September 17, 1982. * Division of Rules and Records. October 1982. 199pp. 8212010052. 16291:083.

The NRC Regulatory Agenda is a compilation of all rules on which the NRC has proposed or is considering action and all petitions for rulemaking which have been received by the Commission and are pending disposition by the Commission. The Regulatory Agenda is updated and issued each quarter. The agendas for April and October are published in their entirety in the Federal Register while a notice of availability is published in the Federal Register for the January and July Agendas.

NUREG-0937: EVALUATION OF PWR RESPONSE TO MAIN STEAMLINE BREAK WITH CONCURRENT STEAM GENERATOR TUBE RUPTURE AND SMALL BREAK LOCA. LAAKSONEN, J.T.; SHERON, B.W. Division of Systems Integration (post 811005). December 1982. 71pp. 8301190471. 16849:122.

In 1980, the NRC staff raised a potential safety issue involving a coincident steamline break, steam generator tube rupture, and small break loss-of-coolant accident (LOCA). The bases for this concern were that the system response, primarily the maintenance of core cooling, was unanalyzed and the adequacy of the present guidance to operators to respond to combination LOCA's was unknown. This report discusses the staff evaluations performed to assess the system response and the adequacy of the present emergency operator guidelines.

NUREG-0940 V01 N01-2: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED. Quarterly Progress Report, January-June 1982. * Director's Office, Office of Inspection and Enforcement. September 1982. 250pp. 8210210024. 15781:187.

This compilation summarizes significant enforcement actions that have been resolved during two quarterly periods (January - June 1982) and includes copies of letters, notices, and orders sent by the Nuclear Regulatory Commission to the licensee with respect to the enforcement action. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, in the interest of promoting public health and safety as well as common defense and security. The intention is that this publication will be issued on a quarterly basis to include significant enforcement actions resloved during the preceeding quarter.

NUREG-0940 V01 N03: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED. Quarterly Progress Report. July-September 1982. * Enforcement Staff. October 1982. 65pp. 8211030200. 15935:323.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (July - September 1982) and includes copies of letters, notices, and orders sent by the Nuclear Regulatory Commission to the licensee with respect to the enforcement action. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, in the interest of promoting public health and safety as well as common defense security. This publication is issued on a quarterly basis to include significant enforcement actions resolved during the preceding quarter.

NUREG-0942 DRFT: CONDUCTING NEED-FOR-POWER REVIEW FOR NUCLEAR POWER PLANTS. Guidelines To States. NASH, A. D. Office of State Programs, Director. December 1982. 107pp. 8302040472. 17044:084.

The report is intended to describe to state regulatory commissions and other state agencies the standards and criteria used by NRC in conducting need-for-power evaluations for the licensing of nuclear power plants. These are intended as guidelines to states which may wish to perform a need-for-power review that will suffice for adoption by the NRC in its licensing process. Three methodologies which have been used for need-for-power evaluations and which meet NRC standards are included.

NUREG-0945 V01: FINAL ENVIRONMENTAL IMPACT STATEMENT ON 10CFR PART 61: "LICENSING REQUIREMENTS FOR LAND DISPOSAL OF RADIOACTIVE WASTE." Summary And Main Report. * Low Level Waste Licensing Branch. November 1982. 219pp. 8211190327. 16167:052.

The three-volume final environmental impact statement (FEIS) is prepared to guide and support publication of a final regulation, 10 CFR

61, for the land disposal of low-level radioactive waste. The FEIS is prepared in response to public comments received on the draft environmental impact statement (DEIS) on the proposed Part 61 regulation. The DEIS was published in September 1981 as NUREG-0782. Public comments received on the proposed Part 61 regulation separate from the DEIS are also considered in the FEIS. The FEIS is not a rewritten version of the DEIS, which contains an exhaustive and detailed analysis of alternatives, but rather references the DEIS and presents the final decision bases and conclusions (cost and impacts) which are reflected in the Part 61 requirements. Four cases are specifically considered in the FEIS representing the following: past disposal practice, existing disposal practice, Part 61 requirements, and an upper bound example.

NUREG-0945 V02: FINAL ENVIRONMENTAL IMPACT STATEMENT ON 10CFR PART 61: "LICENSING REQUIREMENTS FOR LAND DISPOSAL OF RADIOACTIVE WASTE." Appendices A-B. * Low Level Waste Licensing Branch. November 1982. 631pp. 8211190317. 16173:001.

The three-volume final environmental impact statement (FEIS) is prepared to guide and support publication of a final regulation, 10 CFR Part 61, for the land disposal of low-level radioactive waste. The FEIS is prepared in response to public comments received on the draft environmental impact statement (DEIS) on the proposed Part 61 regulation. The DEIS was published in September 1981 as NUREG-0782. Public comments received on the proposed Part 61 regulation separate from the DEIS, are also considered in the FEIS. The FEIS is not a rewritten version of the DEIS, which contains an exhaustive and detailed analysis of alternatives, but rather references the DEIS and presents the final decision bases and conclusions (costs and impacts) which are reflected in the Part 61 requirements. Four cases are specifically considered in the FEIS representing the following: past disposal practice, existing disposal practice, Part 61 requirements, and an upper bound example.

NUREG-0945 V03: FINAL ENVIRONMENTAL IMPACT STATEMENT ON 10CFR PART 61: "LICENSING REQUIREMENTS FOR LAND DISPOSAL OF RADIOACTIVE WASTE." Appendices C-F. * Low Level Waste Licensing Branch. November 1982. 433pp. 8211190323. 16168:011.

The three-volume final environmental impact statement (FEIS) is prepared to guide and support publication of a final regulation, 10 CFR Part 61, for the land disposal of low-level radioactive waste. The FEIS is prepared in response to public comments received on the draft environmental impact statement (DEIS) on the proposed Part 61 regulation. The DEIS was published in September 1981 as NUREG-0782. Public comments received on the proposed Part 61 regulation separate from the DEIS are also considered in the FEIS. The FEIS is not a rewritten version of the DEIS, which contains an exhaustive and detailed analysis of alternatives, but rather references the DEIS and presents the final decision bases and conclusions (costs and impacts) which are reflected in the Part 61 requirements. Four cases are specifically considered in the FEIS representing the following: past disposal practice, existing disposal practice, Part 61 requirements, and upper bound example.

NUREG-0952: TECHNICAL SPECIFICATIONS FOR SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 3. Docket No. 50-362. (Southern California Edison Company) BRINKMAN, D. Office of Nuclear Reactor Regulation, Director.

November 1982. 454pp. 8212140482. 16431:340.

The San Onofre Nuclear Generating Station, Unit 3 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-0023 V01: EIGHTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. * Office of Nuclear Regulatory Research, Director. March 1982. 500pp. 8204020017. 12505:287.

This is a compilation of papers which were presented at the Eighth Water Reactor Safety Research Information meeting. It consists of four volumes.

NUREG/CP-0023 V02: EIGHTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. * Office of Nuclear Regulatory Research, Director. March 1982. 500pp. 8204020012. 12507:275.

This is a compilation of papers which were presented at the Eighth Water Reactor Safety Research Information meeting. It consists of four volumes.

NUREG/CP-0023 V03: EIGHTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. * Office of Nuclear Regulatory Research, Director. March 1982. 500pp. 8204020010. 12509:259.

This is a compilation of papers which were presented at the Eighth Water Reactor Safety Research Information meeting. It consists of four volumes.

NUREG/CP-0023 V04: EIGHTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. * Office of Nuclear Regulatory Research, Director. March 1982. 400pp. 8204020068. 12504:001.

This is a compilation of papers which were presented at the Eighth Water Reactor Safety Research Information meeting. It consists of four volumes.

NUREG/CP-0024 V01: NINTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. * Office of Nuclear Regulatory Research, Director. March 1982. 800pp. 8204220546. 12828:001.

This is a compilation of papers which were presented at the Ninth Water Reactor Safety Research Information Meeting. It consists of three volumes.

NUREG/CP-0024 V02: NINTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. * Office of Nuclear Regulatory Research, Director. March 1982. 300pp. 8204260450. 12856:001.

This is a compilation of papers which were presented at the Ninth Water Reactor Safety Rearch Information Meeting. It consists of three volumes.

NUREG/CP-0024 V03: NINTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. * Office of Nuclear Regulatory Research, Director. March 1982. 350pp. 8204260453. 12859:001.

This is a compilation of papers which were presented at the Ninth Water Reactor Safety Research Information Meeting. It consists of three volumes.

NUREG/CP-0025: PROCEEDINGS OF A WORKSHOP ON ENVIRONMENTAL ASSESSMENT. WATSON, E. C. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1982. 159pp. 8208180217. PNL-4235. 14390:191.

The workshop was convened to consider the technical basis and methodology for regulatory environmental assessment in 1981. The findings and recommendations of the participants addressed several research needs common to aquatic, terrestrial and atmospheric sciences applicable to environmental assessment.

Model validation - The view often expressed was that there simply is no basis in fact for having a high degree of confidence in estimates derived from existing models.

Source term characterization - The physical state and chemical characteristics of the source term are needed to estimate transformations within the plume, depletion of the plume due to dry deposition, and rainout. Some particle size distribution information applicable to normal operating conditions is available but little or no information exists for releases during accidents.

Development of screening techniques - Reliable means of identifying and separating trivial matters from important problems. Techniques using reasonably conservative models to identify effects, pathways, nuclides or pollutants warranting high priority research. This would be closely coupled and dependent upon de minimis environmental effects to be established by the regulatory functions.

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-0028 V01: PROCEEDINGS OF THE SYMPOSIUM ON LOW-LEVEL WASTE DISPOSAL: SITE SUITABILITY REQUIREMENTS. YALCINTAS, M. G. Oak Ridge National Laboratory. JACOBS, D. G. Evaluation Research Corp. September 1982. 253pp. 8209280321. CONF-811218. 15547:350.

This document is a compilation of the papers presented at the Low-Level Waste Symposium on Site Suitability Requirements, the question and answer periods following individual papers, and the panel discussions at the end of each day.

The goal of the symposium was to provide a forum for experts in a variety of fields representing a variety of organizations. These experts were given the opportunity to comment on the reasonableness and appropriateness of those portions of the proposed 10 CFR Part 61 that deal with site suitability. The following questions were to be addressed by the speakers and considered during discussion periods:

(1) Can sites be found that meet the proposed criteria? (2) Would such sites be safe and effective sites? (3) Do the criteria eliminate safe and effective sites? (4) Can it be shown that a given site meets the proposed criteria? (5) Should the proposed criteria be revised? (6) Should other criteria on site suitability be added?

NUREG/CP-0028 V02: PROCEEDINGS OF THE SYMPOSIUM ON LOW-LEVEL WASTE DISPOSAL: SITE CHARACTERIZATION AND MONITORING. GALCINTOS, G. Oak Ridge National Laboratory. December 1982. 485pp. 8301100001. CONF-820674. 16713:001.

This document is a compilation of the papers presented at the low-level Waste Symposium on Site Characterization and Monitoring, the question and answer periods following individual papers, and the panel discussions of the end of each session. The various session topics covered were: geology, geomechanics, ground water, surface water ecology, air quality, and meteorology and climatology.

NUREG/CP-0029 V01: PROCEEDINGS OF THE FOURTH ASTM-EURATOM SYMPOSIUM ON REACTOR DOSIMETRY. March 22-26, 1982, Washington, D. C. KAM, F. B. K. Oak Ridge National Laboratory. August 1982. 583pp. 8209230007. CONF-820321/V1. 14985:001.

These proceedings contain the papers that were submitted for the Fourth ASTM-EURATOM Symposium on Reactor Dosimetry. This series of biennial international symposia brings together specialists from many countries to provide a forum for the exchange of new and critical information concerning the techniques and applications of neutron and gamma dosimetry in materials irradiation studies. These Symposia serve as the main reporting base for work associated with the improvement, standardization, and maintenance of dosimetry, damage correlation, and the associated reactor analysis procedures and data used for predicting the integrated effects of neutron exposure on fuels and materials for light water reactor (LWR), fast breeder reactor (FBR), and magnetic fusion reactor (MFR) nuclear power systems. The ultimate goal is to obtain international standardization of dosimetry methods with quantified uncertainty limits. The theme of the present symposium was radiation metrology techniques, data bases, and standardization. Application and requirements for radiation metrology of irradiated fuels and materials in fission and fusion technology were emphasized. Topics involving light water reactors, fast breeder reactors, and fusion reactors were covered.

NUREG/CP-0029 V02: PROCEEDINGS OF THE FOURTH ASTM-EURATOM SYMPOSIUM ON REACTOR DOSIMETRY. March 22-26, 1982, Washington, D.C. KAM, F.B.K. Oak Ridge National Laboratory. August 1982. 616pp. 8209230011. CONF-820321/V2. 14983:001.

These proceedings contain the papers that were submitted for the Fourth ASTM-EURATOM Symposium on Reactor Dosimetry. This series of biennial international symposia brings together specialists from many countries to provide a forum for the exchange of new and critical information concerning the techniques and applications of neutron and gamma dosimetry in materials irradiation studies. These Symposia serve as the main reporting base for work associated with the improvement, standardization, and maintenance of dosimetry, damage correlation, and the associated reactor analysis procedures and data used for predicting the integrated effects of neutron exposure on fuels and materials for light water reactor (LWR), fast breeder reactor (FBR), and magnetic fusion reactor (MFR) nuclear power systems. The ultimate goal is to obtain international standardization of dosimetry methods with quantified uncertainty limits. The theme of the present symposium was radiation metrology techniques, data bases, and standardization. Application and requirements for radiation metrology of irradiated fuels and materials in fission and fusion technology were emphasized. Topics involving light water reactors, fast breeder reactors, and fusion reactors were covered.

NUREG/CP-0030: SYMPOSIUM ON UNSATURATED FLOW AND TRANSPORT MODELING. ARNOLD, E.M.; GEE, G.W.; NELSON, R.W. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1982. 337pp. 8210150553. PNL-SA-10325. 15725:156.

These proceedings report the technical papers presented and subsequent discussions held at the Symposium on Unsaturated Flow and Transport Modeling, held March 23-24, 1982 in Seattle, Washington. The symposium was focused on the near-surface disposal of low-level radioactive wastes in the unsaturated zone. The symposium was an integral part of a larger study on the state-of-the-art of modeling flow and transport in the unsaturated zone being performed by Battelle Pacific Northwest Laboratory under interagency agreement with the U.S. Nuclear Regulatory Commission.

NUREG/CP-0031 V01: PROCEEDINGS OF THE CSNI SPECIALIST MEETING ON OPERATOR TRAINING AND QUALIFICATIONS. * Division of Technical Information & Document Control. June 1982. 344pp. 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. 337pp. 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/CP-0032: WORKSHOP ON METEOROLOGICAL ASPECTS OF EMERGENCY RESPONSE PLANS FOR NUCLEAR POWER PLANTS. SETHURAMAN, S.; TICHLER, J.; PATRINOS, A.; et al. Brookhaven National Laboratory. August 1982. 75pp. 8208260406. BNL-NUREG-51552. 14590:031.

The Workshop on Meteorological Aspects of Emergency Response Plans for Nuclear Power Plants was held December 1-3, 1981, at SRI International in Menlo Park, California. The purpose of the workshop was to collect and integrate the comments of the user community on the Nuclear Regulatory Commission's (NRC's) atmospheric dispersion modeling requirements in support of nuclear power plant radiological emergency response plans. The user community was represented by utilities, consultants, and state and federal government agencies. The workshop was organized into five short introductory technical sessions, followed by working sessions. The technical sessions provided an overview of the state of the art in health physics; release characteristics; transport, diffusion, and deposition; and operational aspects. The three working groups addressed the themes of health physics and meteorology, release characteristics and meteorology, and dispersion and deposition. This report summarizes the activities and results of both parts of the workshop.

NUREG/CP-0033 V01: PROCEEDINGS OF THE WORKSHOP ON CONTAINMENT INTEGRITY. June 7-9, 1982. SEBRELL, W. A. Sandia Laboratories. November 1982. 368pp. 8212220513. SAND82-1659. 16524:001.

The United States Nuclear Regulatory Commission initiated a research program to study the issue of Containment Integrity. The current phase of this task has focused on the structural response of containment buildings. There remain many questions regarding the integrity in the other components of the containment building. Because of these questions, it was decided a meeting should be held to address the broader issues of containment integrity. As a result, a workshop on containment integrity was held at the Quality Inn, Pentagon City in Alexandria, Virginia, on June 7-9, 1982. The primary objectives of the workshop included: a. Develop a perspective on current Probabilistic Risk Analysis and Severe Accident Sequence Analysis related studies, b. Presentation of research activities and plans by various organizations, c. Characterization of loadings beyond design conditions, d. Delineate the impact of operational integrity, e. Present analytical and experimental results effecting the ultimate capacity of containment buildings, and f. Bring individuals together in related disciplines to promote mutual concerns and establish a basis for future information exchange. The Proceedings is divided into two volumes. Volume I contains the address by Commissioner Ahearne and papers and discussion for Sessions I, II, and Session III. Volume II contains a presentation by Dr. J. Stevenson and papers from Sessions IV and V. Because of time limitation at the end of Sessions IV and VI, the question and answer periods were very short. The coordinators of the workshop regret that time to bring all questions to a meaningful conclusion was not available.

NUREG/CP-0033 V02: PROCEEDINGS OF THE WORKSHOP ON CONTAINMENT INTEGRITY. June 7-9, 1982. SEBRELL, W. A. Sandia Laboratories. November 1982. 370pp. 8212220172. SAND82-1659. 16528:019.

The United States Nuclear Regulatory Commission initiated a research program to study the issue of Containment Integrity. The current phase of this task has focused on the structural response of containment buildings. There remain many questions regarding the integrity in the other components of the containment building. Because of these questions, it was decided a meeting should be held to address the broader issues of containment integrity. As a result, a workshop on containment integrity was held at the Quality Inn, Pentagon City in Alexandria, Virginia, on June 7-9, 1982. The primary objectives of the workshop included: a. Develop a perspective on current Probabilistic Risk Analysis and Severe Accident Sequence Analysis related studies, b. Presentation of research activities and plans by various organizations, c. Characterization of loading beyond design conditions, d. Delineate the impact of operational integrity, e. Present analytical and experimental results effecting the ultimate capacity of containment buildings, and f. Bring individuals together in related disciplines to promote mutual concerns and establish a basis for future information exchange. The Proceedings is divided into two volumes. Volume I contains the address by Commissioner Ahearne and papers and discussion for Sessions I, II, and Session III. Volume II contains a presentation by Dr J. Stevenson and papers from Sessions IV and V. Becasue of time limitation at the end of Sessions IV and VI, the question and answer periods were very short. The coordinators of the workshop regret that time to bring all questions to a meaningful conclusion was not available.

NUREG/CP-0034 V01: PROCEEDINGS OF THE TOPICAL MEETING ON ADVANCES IN REACTOR PHYSICS AND CORE THERMAL HYDRAULICS. * NRC - No Detailed Affiliation Given. August 1982. 550pp. 8208260401. 14590:177.

Technical papers presented at the ANS Topical Meeting on Advances in Reactor Physics and Core Thermal Hydraulics, September 22-24, 1982, at Kiamesha Lake, N.Y. are included in these Proceedings. Reactor physics, core thermal hydraulics, and the interactions between core physics and thermal hydraulics are covered both for thermal reactors and for fast breeders. There are sessions on current challenges in these areas, on measurement and analysis of fast reactor physics parameters, on coupled core physics and thermal hydraulics analysis, on in-core fuel management, on nodal and homogenization methods in reactor physics, on core thermal hydraulic and nuclear instrumentation, and on validation of fast reactor thermal hydraulic methods. In addition there are sessions on reactor theory, on measurement and analysis of thermal reactor physics parameters, on validation of thermal reactor thermal hydraulics methods, and on development and utilization of differential and integral nuclear data. This report consists of two volumes.

NUREG/CP-0034 V02: PROCEEDINGS OF THE TOPICAL MEETING ON ADVANCES IN REACTOR PHYSICS AND CORE THERMAL HYDRAULICS. * NRC - No Detailed Affiliation Given. August 1982. 639pp. 8208260420. 14592:250.

Technical papers presented at the ANS Topical Meeting on Advances in Reactor Physics and Core Thermal Hydraulics, September 22-24, 1982, at Kiamesha Lake, N.Y. are included in these Proceedings. Reactor physics, core thermal hydraulics, and the interactions between core physics and thermal hydraulics are covered both for thermal reactors and for fast breeders. There are sessions on current challenges in these areas, on measurement and analysis of fast reactor physics parameters, on coupled core physics and thermal hydraulics analysis, on in-core fuel management, on nodal and homogenization methods in reactor physics, on core thermal hydraulic and nuclear instrumentation, and on validation of fast reactor thermal hydraulic methods. In addition there are sessions on reactor theory, on measurement and analysis of thermal reactor physics parameters, on validation of thermal reactor thermal hydraulics methods, and on development and utilization of differential and integral nuclear data. This report consists of two volumes.

NUREG/CP-0036: PROCEEDINGS OF THE WORKSHOP ON NUCLEAR POWER PLANT AGING. Bethesda, MD. August 4-5, 1982. BADER, B.E.; HANCHEY, L.A. Sandia Laboratories. December 1982. 289pp. 8301120094. SAND82-2264C. 16783:103.

A Workshop on Nuclear Power Plant Aging was held on August 4-5, 1982, in Bethesda, Maryland, sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, and hosted by Sandia National Laboratories, Albuquerque, New Mexico.

The proceedings were divided into four Technical Sessions. Twelve papers were presented in Technical Sessions I and II. The large majority have been reproduced exactly as they were received. A few have been lightly edited and retyped. Technical Session III consisted of oral presentations. As indicated, four presentations have been abstracted from the transcript of the proceedings and in two cases abstracts, as submitted by the authors, are included. A summary of Technical Session IV (Panel Discussion) is presented, also based on the transcript.

NUREG/CP-0039: PROCEEDINGS OF THE HEALTH PHYSICS SOCIETY 1981 SUMMER SCHOOL (AT UNIV OF KY):SELECTED TOPICS IN REACTOR HEALTH PHYSICS. CHRISTENSON, R.; SAYEG, J. A.; SIMMONS, G. H.; et al. Kentucky, Univ. of. December 1982. 697pp. 8301190441. 16851:289.

A collection of the papers presented by participants at the 1981 Summer School on SELECTED TOPICS IN REACTOR HEALTH PHYSICS held at the University of Kentucky, Lexington, and sponsored by the Health Physics Society. The papers cover a range of topics of continuing interest such as health physics data management systems, dosimetry problems, quality assurance for radiation monitoring, instrument systems, measurements and statistics, contamination control, rad waste treatment and ALARA program design.

NUREG/CR-0169 V04: LOFT EXPERIMENTAL MEASUREMENTS UNCERTAINTY ANALYSES:LIQUID LEVEL TRANSDUCERS. MEACHUM, T. R. EG&G, Inc. October 1982. 17pp. 8211110102. EGG-2037. 16041:341.

The uncertainties of the measurements from the liquid level transducers (LLTs) installed in the Loss-of-Fluid Test (LOFT) reactor system have been computed and documented herein. The LLT is a conductivity-sensitive device designed to detect the presence or absence of liquid. Four types of LLTs are installed at various locations in the reactor core, downcomer, lower plenum, and upper plenum to provide information pertaining to the liquid level during LOFT transient experiments. This analysis determined that the in-core, downcomer, lower plenum, and upper plenum LLTs have 2 uncertainties of 2.78, 4.5, 8.33, and 6.25% of range, respectively.

NUREG/CR-0169 V11: LOFT EXPERIMENTAL MEASUREMENTS UNCERTAINTY ANALYSES, VOLUME XI,FREE FIELD PRESSURE TRANSDUCER. GOODRICH, L. D.; LASSAHN, G. D. EG&G, Inc. July 1982. 19pp. 8208260486. EGG-2037. 14567:336.

Fast pressure transients during the subcooled decompression phase of experiments performed in the Loss-of-Fluid Test (LOFT) system are measured using free-field pressure transducers (FFPTs). This uncertainty analysis estimated the uncertainty for these measurements to be the root-sum-square combination of 0.13 MPa and 1% of reading. Although the FFPTs provide useful pressure measurements during other phases of LOFT experiments, the uncertainty for these measurements may be somewhat larger and is not included in this analysis.

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-0169 V20: LOFT EXPERIMENTAL MEASUREMENTS UNCERTAINTY ANALYSES. VOLUME XX. FLUID VELOCITY MEASUREMENT USING PULSED NEUTRON ACTIVATION. LASSAHN, G. D.; TAYLOR, D. J. EG&G, Inc. September 1982. 32pp. 8210150560. EGG-2037. 15747:296.

Analyses of uncertainty components inherent in pulsed-neutron-activation (PNA) measurements in general and the Loss-of-Fluid Test (LOFT) system in particular are given. Due to the LOFT system's unique conditions, previously-used techniques were modified to make the velocity measurement. These methods render a useful, cost-effective measurement with an estimated uncertainty of 11% of reading.

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-0200 V01-03: SCALE:A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. BUCHOLZ, J. A. Oak Ridge National Laboratory. January 1982. 7,200pp. 8204210747. ORNL/NUREG/CSD-. 12788:001.

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-0416 R01: NONSAP-C:A NONLINEAR STRESS ANALYSIS PROGRAM FOR CONCRETE CONTAINMENTS UNDER STATIC, DYNAMIC, AND LONG-TERM LOADINGS. ANDERSON, C. A. ; SMITH, P. D. ; CARRUTHERS, L. M. Los Alamos Scientific Laboratory. March 1982. 139pp. 8203190049. LA-7496-MS R01. 12352:193.

This report describes the NONSAP-C finite element code and its application to the nonlinear structural analysis of three-dimensional concrete containments under static, dynamic, and long-term loadings. Features of this code that allow for easy application to realistic concrete structural problems are discussed, along with the various material models used to represent plain and reinforced concrete, for both time-dependent and time-independent behavior. Applications of the code to analysis of conventional reinforced concrete structures and to the structural analysis of prestressed concrete reactor vessels (PCRVs) and PCRV models are illustrated. Comparisons of the code predictions with previous numerical solutions to these problems or to experimental data are made. Input instructions for the NONSAP-C code are described in the report.

NUREG/CR-0612: INVESTIGATIONS OF THE PERFORMANCE OF SOLIDIFIED HIGH-LEVEL NUCLEAR WASTE FORMS. CHEUNG, H. Lawrence Livermore Laboratory. March 1982. 198pp. 8204290453. UCRL-52700. 12896:207.

Studies of solidified high-level waste during the period from 1976 to 1978, when work was terminated because of shifting of national emphasis onto spent fuel disposal, are presented in this report. We have given a definition of the problem of management, i.e., handling, generation, and disposal of solidified high-level waste derived from operation of commercial light-water reactors by describing the components of the waste manegement system: the waste form, the containers, storage and transportation appurtenances, handling equipment, the repository surface and underground facilities, the repository site; and the operations. We developed a systems analysis methodology to assess the hazards of waste management. We compiled data on accident probabilities, waste form characteristics, and geological and hydrological properties of potential repository sites. We generated a wide range of management scenarios. We also performed limited sensitivity and uncertainty analyses. On the basis of available information, our preliminary investigations showed that transportation and interim storage are of most concern. We have also identified areas needing further study: transportation data base, thermal and seismic aspects of interim storage, human factors, geochemical transport of radionuclides, and ground water composition, among others. In addition to the technical solution of the problems, we have also give brief consideration to historical and socioeconomic aspects.

NUREG/CR-0633: STEAM EXPLOSION TRIGGERING PHENOMENA PART 2: CORIUM-A AND CORIUM-E SIMULANTS AND OXIDES OF IRON AND COBALT STUDIED WITH A FLOODABLE ARC MELTING APPARATUS. NELSON, L. S.; PLANNER, H. N.; BUSTON, L. D. Sandia Laboratories. March 1982. 141pp. 8204290447. SAND79-0260. 12897:042.

Laboratory scale experiments on the thermal interaction of light water reactor core materials with water, initially reported in Part I of this report (Nelson and Buxton, 1978), have been continued with emphasis on molten oxidic systems. The experiments were performed with a floodable arc melter using high-speed photography, flash x-ray imaging and lithium niobate pressure transducers as active diagnostic techniques during the interaction, and debris analysis afterward. Molten samples (10 to 35 grams) of four-component Corium-A, and Corium-E simulants, and of iron oxides and cobalt oxides were flooded with approximately 1.5 liters of water. With only a few exceptions, natural triggering of pressure-producing interactions did not occur with these oxidic mateials. However, when short duration pressure transients were introduced into the water, vigorous explosions could be initiated, producing measured peak pressures as high as 5.7 MPa. Associated with these explosions was extensive fragmentation of the melt with weight-average particle sizes as small as 16 um. Most of the explosive events which were triggered were multi-stage in nature. Nine parameters were studied: trigger magnitude, direction of trigger application, and time delay for application of the initiating pressure transients; melt composition; ambient pressure in the chamber; water subcooling; water composition; melt quantity; and heating time during preparation of melt. Initiation of explosions was observed to be most sensitive to the first six parameters.

NUREG/CR-0665: TRAC-PIA. AN ADVANCED BEST-ESTIMATE COMPUTER PROGRAM FOR PWR LOCA ANALYSIS. * Los Alamos Scientific Laboratory. March 1982. 379pp. 8205060027. LA-7777-MS. 13004:001.

The Transient Reactor Analysis Code (TRAC) is being developed at the Los Alamos Scientific Laboratory to provide an advanced "best estimate predictive capability for the analysis of postulated accidents in light water reactors." TRAC-P1A provides this analysis capability for pressurized water reactors and for a wide variety of thermal-hydraulic experimental facilities. It features a three-dimensional treatment of the pressure vessel and associated internals; two-phase nonequilibrium hydrodynamics models; flow-regime-dependent constitutive equation treatment; reflood tracking capability for both bottom flood and falling film quench fronts; and consistent treatment of entire accident sequences including the generation of consistent initial conditions.

NUREG/CR-0729: PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM-THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT FOR TEST 171. CLEMONS, V. D.; FLANDERS, R. M.; CRADDICK, W. G. Oak Ridge National Laboratory. March 1982. 50pp. 8205060033. ORNL/NUREG/TM-2. 13006:160.

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) test 171. The objective of the program is to investigate the thermal-hydraulic phenomenon governing the energy transfer and transport processes that occur during a loss-of-coolant accident (LOCA) in a PWR system. Test 171 was an isothermal test conducted to gain baseline information about the hydrodynamic behavior of the facility with the pressurizer surge line located upstream of the main heat exchangers. The primary purpose of this report is to make the reduced instrument responses during test 171 available. The responses are presented in graphical form in engineering units and have been analyzed only to the extent necessary to assure reasonableness and consistency.

NUREG/CR-0730: PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM-THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT FOR TEST 160. CLEMONS, V. D.; CRADDICK, W. G.; WHITE, M. D. Oak Ridge National Laboratory. March 1982. 54pp. 8205060199. ORNL/NUREG/TM-3. 12999:070.

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) test 160. The objective of the program is to investigate the thermal-hydraulic phenomenon governing the energy transfer and transport processes that occur during a loss-of-coolant accident (LOCA) in a PWR system. Test 160 was conducted to determine (through comparison with tests 151 and 155) the effect of altered initial core power levels on fuel rod simulator surface temperature and surface flux behavior in the THTF electric core. The primary purpose of this report is to make the reduced instrument responses during test 160 available. The responses are presented in graphical form in engineering units and have been analyzed only to the extent necessary to assure reasonableness and consistency.

NUREG/CR-0731: PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM-THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT FOR TEST 161. CLEMONS, V. D.; CRADDICK, W. G.; WHITE, M. D. Oak Ridge National Laboratory. March 1982. 56pp. 8205030640. ORNL/NUREG/TM-3. 12949:203.

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) test 161. The objective of the program is to investigate the thermal-hydraulic phenomenon governing the energy transfer and transport processes that occur during a loss-of-coolant accident (LOCA) in a PWR system. Test 161 was conducted to determine (through comparisons with tests 153 and 155) the effect of bundle outlet fluid temperature on the behavior of fuel rod simulator surface temperatures and surface fluxes in the THTF electric core. The primary purpose of this report is to make the reduced instrument responses during test 161 available. The responses are presented in graphical form in engineering units and have been analyzed only to the extent necessary to assure reasonableness and consistency.

NUREG/CR-0732: PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM-THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT FOR TEST 162. CLEMONS, V. D. ; CRADDICK, W. G. ; WHITE, M. D. Oak Ridge National Laboratory. March 1982. 53pp. 8205060132. ORNL/NUREG/TM-3. 12999:173.

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) test 162. The objective of the program is to investigate the thermal-hydraulic phenomenon governing the energy transfer and transport processes that occur during a loss-of-coolant accident (LOCA) in a PWR system. Test 162 was a duplicate of test 152 and was conducted to verify repeatability of system response. The primary purpose of this report is to make the reduced instrument responses during test 162 available. The responses are presented in graphical form in engineering units and have been analyzed only to the extent necessary to assure reasonableness and consistency.

NUREG/CR-0733: PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM-THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT FOR TEST 163. CLEMONS, V. D. ; CRADDICK, W. G. ; FLANDERS, R. M. Oak Ridge National Laboratory. March 1982. 50pp. 8205030663. ORNL/NUREG/TM-3. 12949:150.

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) test 163. The objective of the program is to investigate the thermal-hydraulic phenomenon governing the energy transfer and transport processes that occur during a loss-of-coolant accident (LOCA) in a PWR system. Test 163 was conducted to determine the effect of moving the pressurizer surge line from downstream to upstream of the main heat exchangers. The primary purpose of this report is to make the reduced instrument responses during test 163 available. The responses are presented in graphical form in engineering units and have been analyzed only to the extent necessary to assure reasonableness and consistency.

NUREG/CR-0734: PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM-THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT FOR TEST 164R. CLEMONS, V. D. ; CRADDICK, W. G. ; FLANDERS, R. M. Oak Ridge National Laboratory. March 1982. 47pp. 8205040063. ORNL/NUREG/TM-3. 12972:019.

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) test 164R. The objective of the program is to investigate the thermal-hydraulic phenomenon governing the energy transfer and transport processes that occur during a loss-of-coolant accident (LOCA) in a PWR system. Test 164R was conducted to determine (through comparison with test 163) the sensitivity of fuel rod simulator surface temperatures and surface fluxes to alterations in electric core power and outlet temperature when the pressurizer surge line is located upstream of the main heat exchangers. The primary purpose of this report is to make the reduced instrument responses during test 164R available. The responses are presented in graphical form in engineering units and have been analyzed only to the extent necessary to assure reasonableness and consistency.

NUREG/CR-0735: PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM-THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT FOR TEST 165. CLEMONS, V. D.; CRADDICK, W. G.; WHITE, M. D. Oak Ridge National Laboratory. March 1982. 53pp. 8205060129. ORNL/NUREG/TM-3. 12999:122.

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) test 165. The objective of the program is to investigate the thermal-hydraulic phenomenon governing the energy transfer and transport processes that occur during a loss-of-coolant accident (LOCA) in a PWR system. Test 165 was conducted to determine the effect of an intact hot leg on the behavior of fuel rod simulator surface temperatures and surface fluxes in the THTF electric core. The primary purpose of this report is to make the reduced instrument responses during test 165 available. The responses are presented in graphical form in engineering units and have been analyzed only to the extent necessary to assure reasonableness and consistency.

NUREG/CR-0787: A STUDY OF THE FIXED SITE NEUTRALIZATION MODEL (FSNM). ENGI, D.; HARLAN, C. P. Sandia Laboratories. March 1982. 77pp. 8111100097. SAND79-0273. 10593:266.

The purpose of this report is to describe in detail an example of input to the Fixed Site Neutralization Model (FSNM). This input was synthesized by NRC staff to represent the hypothetical facility which forms the basis of a board game - the Guard Tactics Simulation (GTS) - which was developed by NRC/NMSS. In addition to the facility representation, the input also reflects the security force response strategy and the adversary scenarios which were conceptualized by NRC staff in order to test FSNM.

NUREG/CR-0833: FIRE PROTECTION RESEARCH PROGRAM CORNER EFFECTS TESTS. KLAMERUS, L. J. Sandia Laboratories. March 1982. 72pp. 8111100089. SAND79-0966. 10593:339.

Under the direction of the Nuclear Regulatory Commission, Sandia Laboratories has been conducting confirmatory research in fire protection for nuclear power plants. During all previous full scale fire tests at Sandia Laboratories involving fires, both electrically and exposure initiated, an open area in a nuclear power plant was simulated. The question was often asked, "How much contribution to fire severity does a reradiating ceiling and wall make?" This report presents the results of several tests which address this question. By quantifying the effects of corner reradiation (i.e., a ceiling joining a wall) at different distances from a horizontal array of cable trays ignited by an exposure fire, it was found for the cables tested that fire damage, as measured by the extent of cable insulation degradation, varied approximately as the inverse of the square of the distance separating the cables from the corner. As experienced in previous Sandia fire tests, the difference in fire resistance between IEEE-383 qualified cable and unqualified cable was apparent in these corner-configuration tests.

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-1030 V03: SEDIMENT AND RADIONUCLIDE TRANSPORT IN RIVERS. Phase 3-Field Sampling Program For Cattaraugus and Buttermilk Creek, New York. ECKER, R. M.; WALTERS, W. H.; ONISHI, Y. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1982. 234pp. 8209010455. 14716:113.

A field sampling program was conducted on Cattaraugus and Buttermilk Creeks, New York during April 1979 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. Bed sediment, suspended sediment and water samples were collected during unsteady flow conditions over a 45 mile reach of stream channel. Radiological analysis of these samples included gamma ray spectrometry analysis, and radiochemical separation and analysis of Sr-90, Pu-238, Pu-239, Pu-240, Am-241 and Cm-244. Tritium analysis was also performed on water samples. Based on the evaluation of radionuclide levels in Cattaraugus and Buttermilk Creeks, the Nuclear Fuel Services facility at West Valley, New York, may be the source of Cs-137, Sr-90, Cs-134, Co-60, Pu-238, P-239-240, Am-241, Cm-244 and tritium found in the bed sediment, suspended sediment and water of Buttermilk and Cattaraugus Creeks. This field sampling effort was the last of a three phase program to collect hydrologic and radiologic data at different flow conditions.

NUREG/CR-1030 V04: SEDIMENT AND RADIONUCLIDE TRANSPORT IN RIVERS: Summary Report; Field Sampling Program For Cattaraugus and Buttermilk Creeks, New York. WALTERS, W. H.; ECKER, R. M.; ONISHI, Y. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1982. 130pp. 8301100038. PNL-3117. 16747:205.

A three-phase field sampling program was conducted on the Buttermilk, Cattaraugus Creek system 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 radionuclides transport model, SERATRA. Phase 1 of the sampling program was conducted during November and December 1977; Phase 2 during September 1978; and Phase 3 during April 1979. Bed sediment, suspended sediment, and water samples were collected over a 45-mile reach fo the creek system. Bed sediment samples were also collected at the mouth of Cattaraugus Creek in Lake Erie. A fourth sampling trip was conducted during May 1980 to obtain supplementary channel geometry data and flood plain sediment samples. Radiological analysis of these samples included gamma ray spectrometry analysis, and radiochemical separation and analysis of Sr-90, Pu-238, Pu-239, 240, Am-241 and Cm-244. Tritium analysis was also performed on water samples. Based on the evaluation of radionuclide levels in Cattaraugus and Buttermilk Creeks, the Nuclear Fuel Services facility at West Valley, New York, may be the source of Cs-137, Sr-90, Cs-134, Co-60, Pu-238, Pu-239, 240, Am-241, Cm-244 and tritium found in the bed sediment, suspended sediment and water of Buttermilk and Cattaraugus Creeks.

NUREG/CR-1098: A MATHEMATICAL MODEL FOR SIMULATION OF THE FATE OF COPPER IN A MARINE ENVIRONMENT. ORLOB, G. T.; HROVAT, D.; WAKEMAN, T.; et al. Laurence Livermore Laboratory. December 1982. 106pp. 8301190455. UCRL-52728. 16853:266.

A methematical model for the simulation of the fate of copper in a marine environment was developed. The model, which describes the kinetics of copper transformation from ionic copper to complexes with dissolved organic matter and sorption on suspended sediment, is imbedded in a two-dimensional finite element model which is capable of simulating advection and diffusion processes in natural receiving waters. Kinetic rate and equilibrium constants for the model were developed idependently in laboratory experiments. A test simulation was performed under realistic conditions of slug discharge of ionic copper with the cooling water from a nuclear power station situated on the California coast. Results show that the model performed correctly under the conditions assumed. Future research and developement is directed toward improving description of copper kinetics under varying environmental conditions and exploring the sensitivity of the model.

NUREG/CR-1111: DEVELOPMENT OF A FABRICATION PROCEDURE FOR THE MRBT FUEL SIMULATOR BASED ON THE USE OF COLD-PRESSED BORON NITRIDE PREFORMS. MCCULLOCK, R. W.; JACOBS, P. T.; CLARK, D. L. Oak Ridge National Laboratory. March 1982. 149pp. 8006270065. ORNL/NUREG/TM-3. 02795:168.

The Multirod Burst Test (MRBT) Program requires prompt availability of high-quality fuel simulators (FSs) to meet its program objectives (studies of Zircaloy deformation resulting from postulated PWR LOCA conditions). The FS electrically simulates the thermal behavior of the nuclear fuel. The primary FS performance requirement is uniform axial clad temperature distribution (maximum allowable deviation of 1.5%) during transient conditions. This is determined by transient infrared examination of the FS ramped to 500 degrees C (932 degrees F) at a rate of 50 degrees C (90 degrees F) per second. (The industrially procured simulators rarely met this requirement, yet many were used for less critical applications.) This report delineates the development of a new fabrication technology for the FS by the ORNL Fuel Rod Simulator Technology Development Program; this technology alleviated the difficulties encountered with simulators procured from industrial sources.

NUREG/CR-1120 V08: SEISMIC SAFETY MARGINS RESEARCH PROGRAM. Progress Report No. 12. BOHN, M. P. ; BERNREUTER, D. L. ; CHUANG, L. E. ; et al. Lawrence Livermore Laboratory. February 1982. 100pp. 8203040038. 12120:098.

This document is a progress report on the Seismic Safety Margins Research Program (SSMRP) covering the period July 1, 1981 through September 30, 1981. The report gives a general description of the program, together with financial summaries and individual project details. Each project is summarized to show accomplishments, schedules, milestones and completion dates, budget and expenditures, and any concerns that may affect the project.

NUREG/CR-1120 V09: SEISMIC SAFETY MARGINS RESEARCH PROGRAM: Progress Report No. 13. BOHN, M. P. ; BERNREUTER, D. L. ; CHUANG, T. Y. ; et al. Lawrence Livermore Laboratory. August 1982. 89pp. 8209210509. 14949:144.

The Seismic Safety Margins Research Program (SSMRP) is an NRC-funded, multiyear program conducted by Lawrence Livermore National Laboratory (LLNL). Its goal is to develop a complete, fully coupled analysis procedure (including methods and computer codes) for estimating the risk of an earthquake-caused radioactive release from a commercial nuclear power plant. The analysis procedure is based on a state-of-the-art evaluation of the current seismic analysis and design process and explicitly includes the uncertainties inherent in such a process. The results will be used to improve seismic licensing requirements for nuclear power plants.

This document is a progress report on the Seismic Safety Margins Research Program covering the period October 1, 1981, through March 31, 1982. The report gives a general description of the program, together with financial summaries and individual project details. Each project is summarized to show accomplishments, schedules, milestones and completion dates, budget and expenditures, and any concerns that may affect the project.

NUREG/CR-1141: ADVANCED REACTOR SAFETY RESEARCH. Quarterly Report July-September 1979. * Sandia Laboratories. March 1982. 218pp. 8205060056. SAND79-2158. 13007:001.

Sandia Laboratories, Alburquerque, New Mexico is conducting the Advanced Reactor Safety Research Program on behalf of the U.S. Nuclear Regulatory Commission. The overall objective of the program is to provide NRC with a comprehensive data base essential to (a) defining key safety issues, (b) understanding the controlling-accident sequences, (c) verifying the complex computer models used in accident analysis and licensing reviews, and (d) assuring the public that advanced power-reactor systems will not be licensed and placed in commercial service in the United States without appropriate consideration being given to their effects on health and safety.

NUREG/CR-1159: MATHEMATICAL PHANTOMS REPRESENTING CHILDREN OF VARIOUS AGES FOR USE IN ESTIMATES OF INTERNAL DOSE. CRISTY, M. Oak Ridge National Laboratory. March 1982. 115pp. 8204150558. ORNL/NUREG/TM-3. 12691:152.

This report presents development of a series of distinct phantoms representing children of ages 0, 1, 5, 10, and 15 years to overcome the shortcomings of the mathematical phantom developed in the mid-1960's by Fisher and Snyder. To estimate dose in children, Snyder and co-workers employed phantoms that were transformations of the adult phantom.

Equations for major body sections were given explicitly, with the internal organs defined implicitly through similitude transformation equations. The major shortcoming of these derivative phantoms was that the organ sizes and bone marrow distributions were not always realistic. All equations for boundaries of organs are explicitly defined with realistic sizes. In addition, the regional distributions of hematopoietically active bone marrow and inactive fatty marrow have been assigned for each phantom, using the method of Cristy. Because detailed anatomical data for organ shapes and locations are not available for children, the "Similitude Rule" was used to determine the shape and the location of most of the organs. This rule is consistent with drawings depicting developmental trends of organs in the trunk. Known exceptions to the rule were adjusted appropriately. The organ volumes were assigned such that organ masses at the various ages conform closely with the data presented in International Commission on Radiological Protection Publication 23.

NUREG/CR-1190: RISK METHODOLOGY FOR GEOLOGIC DISPOSAL OF RADIOACTIVE WASTE: THE NETWORK FLOW AND TRANSPORT (NWFT) MODEL. CAMPBELL, J. E. ; LANGKOPF, B. S. ; KAESTNER, P. C. ; et al. Sandia Laboratories. March 1982. 77pp. 8205060085. SAND79-1920. 13006:268.

The Network Flow and Transport (NWFT) Model has been developed at Sandia to supplement the groundwater flow and radionuclide transport capability provided by the Sandia Waste Isolation Flow and Transport (SWIFT) model. NWFT requires only a small fraction of the computer time required by SWIFT. However, transport calculations in NWFT are presently limited to decay chains of up to three isotopes which must have the same distribution coefficients. It is anticipated that NWFT will be used jointly with SWIFT. SWIFT will be used to establish the fluid flow field for different depository breachment scenarios. SWIFT will also provide pressure boundary conditions for NWFT and will be used to check selected NWFT radionuclide discharge results. NWFT will be used in sensitivity and risk analysis to examine the effects of variables which alter the radionuclide source rate and migration time.

NUREG/CR-1205 R01: DATA SUMMARIES OF LICENSEE EVENT REPORTS OF PUMPS AT U.S. COMMERICIAL NUCLEAR POWER PLANTS. January 1, 1972 to September 30, 1980. TROJOVSKY, M. EG&G, Inc. January 1982. 422pp. 8201280139. 11730:202.

This report presents data summaries of Licensee Event Reports (LERs) of pumps at U.S. Commercial (light water reactor) nuclear power plants from January 1, 1972, through September 30, 1980. LERs are written reports filed with the NRC whenever certain failures or incidents occur concerning nuclear plant safety systems. The LERs are sorted according to plant, system, and human factors. The selected pump failures reported in the LERs are used to estimate gross standby and operating failure rates, in per-hour and per-demand units. An explanation and summary tables of all results are provided. In addition to the quantitative failure rate information, there is also considerable qualitative information tabulated to allow the user to make additional pump failure rate calculations or inferences. This revised report updates and supersedes the original January 1980 printing of NUREG/CR-1205.

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-1254: FEASIBILITY OF BOTTOM-COOLED DEBRIS BED EXPERIMENTS IN THE ANNULAR CORE RESEARCH REACTOR (ACRR). RIVARD, J. B. ; GRONAGER, J. E. ; LIPINSKI, R. J. ; et al. Sandia Laboratories. March 1982. 43pp. 8204290463. SAND79-0876. 12896:167.

This report documents a preliminary step in the investigation of the feasibility of bottom-cooled debris-bed experiments in the Sandia Annular Core Research Reactor (ACRR). The objective of such experiments is to address the issue of in-vessel retention of debris following a core-disruptive accident in a sodium-cooled nuclear reactor--an issue of obvious safety significance. The study concludes that bottom-cooled experiments yielding significant information applicable to in-vessel retention are feasible in the ACRR, that the supporting technology is available or can be reasonably acquired, and that the experiments can be performed in a manner free from hazard to the public. Three experiment approaches at increasing levels of difficulty are described for achieving the program goals.

NUREG/CR-1288: FISSION PRODUCT SOURCE TERMS FOR THE LWR LOSS-OF-COOLANT ACCIDENT. LORENZ, R. A. ; COLLINS, J. L. ; MALINAUSKAS, A. Oak Ridge National Laboratory. March 1982. 34pp. 8205030648. ORNL/NUREG/TM-3. 12949:001.

Models for cesium and iodine release from light-water reactor (LWR) fuel rods failed in steam were formulated based on experimental fission product release data from several types of failed LWR fuel rods. The models were applied to a pressurized water reactor (PWR) undergoing a hypothetical loss-of-coolant accident (LOCA) temperature transient. Calculated total iodine and cesium releases from the fuel rods were 0.053 and 0.025% of the total reactor inventories of these elements, respectively, with most of the release occurring at the time of rupture. These values are approximately two orders of magnitude less than releases used in WASH-1400, the Reactor Safety Study.

NUREG/CR-1303: UNCERTAINTIES IN THE CALCULATION OF LONG-TERM COLLECTIVE DOSE AND HEALTH EFFECTS - A PRELIMINARY ASSESSMENT. KOCHER, D. C.; RYAN, M. T.; LEGGETT, R. W.; et al. Oak Ridge National Laboratory. March 1982. 73pp. 8204300022. ORNL/NUREG/TM-3. 12920:351.

The purpose of this report is to present the results of a preliminary study of potential uncertainties involved in the calculation of long-term population dose and resulting human health effects following releases of long-lived radionuclides to the biosphere. Consideration of these uncertainties is limited in this study to a period of 10,000 years following a release which is assumed to occur in the near future. In addition, the only health effect considered is the induction of fatal cancers from exposure to the released activity.

NUREG/CR-1306: QUARTERLY PROGRESS REPORT ON BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM FOR OCTOBER-DECEMBER 1979. BOHANAN, R. E.; FELDE, D. K.; HYMAN, C. R.; et al. Oak Ridge National Laboratory. March 1982. 24pp. 8205060119. ORNL/NUREG/TM-3. 13002:317.

The Blowdown Heat Transfer Program has been asked by the Nuclear Regulatory Commission to conduct a series of bundle uncovery and recovery tests with the Thermal-Hydraulic Test Facility (THTF) to gain data on the thermal-hydraulic phenomena that might occur in a reactor core during a small-break loss-of-coolant accident. A single fuel rod simulator was tested in the Forced Convection Test Facility under conditions more severe than those anticipated for the uncovery and recovery tests to assess rod survivability. Results were positive, and two of the series of uncovery and recovery tests were conducted in the THTF. Development and assessment of analytical tools needed for planning and analyzing bundle 3 testing are continuing. An isothermal blowdown in the THTF was conducted. Spurious trips of the bundle safety circutry have prevented completion of rod characterization testing. Design and procurement for the in-bundle densitometer are continuing.

NUREG/CR-1363 R01: DATA SUMMARIES OF LICENSEE EVENT REPORTS OF VALVES AT U.S. COMMERCIAL NUCLEAR POWER PLANTS FROM JANUARY 1, 1976 TO DECEMBER 31, 1980. MILLER, C. F.; HUBBLE, W. H.; TROJOVSKY, M.; et al. EG&G, Inc. October 1982. 474pp. 8211160489. EGG-EA-5816. 16110:001.

This report presents data summaries of Licensee Event Reports (LERs) of valves at U.S. commercial (light water reactor) nuclear power plants from January 1, 1976, through December 31, 1980. LERs are written reports filed with the NRC whenever certain failures or incidents occur concerning nuclear plant safety systems. The LERs are sorted according to plant, type of event, human factors, and valve type. The valve failures or incidents reported in the LERs were used to estimate gross standby and operating failure rates, in per-hour and per-demand units. The report includes a variety of different statistics calculated to highlight or show important failure modes or other failure information. In addition to the quantitative failure rate information, there is also considerable qualitative information tabulated to allow the user to make additional valve failure rate calculations or inferences. This revised report updates and supersedes the original three-volume June 1980 printing of NUREG/CR-1363.

NUREG/CR-1369 R01: PROCEDURES EVALUATION CHECKLIST FOR MAINTENANCE, TEST AND CALIBRATION PROCEDURES USED IN NUCLEAR POWER PLANTS. BRUNE, R. L.;

WEINSTEIN, M. HPT, Inc. * Sandia Laboratories. September 1982. 25pp. 8210060004. SAND80-7054. 15636:308.

This report describes a checklist to be used by the United States Nuclear Regulatory Commission (NRC) inspectors during their evaluation of maintenance, test and calibration procedures. The objective of the checklist is to aid inspectors in identifying procedural characteristics that can lead to human performance deficiencies. A companion document, Development of a Checklist for Evaluating Maintenance, Test, and Calibration Procedures Used in Nuclear Power Plants, NUREG/CR-1368, SAND80-7053, describes how the checklist was developed.

Revision 1 of the checklist, presented herein, is the result of a one-year field test by NRC inspectors in all five NRC regions. It incorporates improvements that were suggested by inspectors based on their experience with the checklist in performing evaluations of licensee procedures.

NUREG/CR-1409: SUMMARY OF THE ZION/INDIAN POINT STUDY. MURFIN, W. B. Sandia Laboratories. March 1982. 70pp. 8204290505. SAND80-0617. 12895:280.

Results of a study by Sandia National Laboratories (SNL), Los Alamos National Scientific Laboratory (LASL), and Battelle Columbus Laboratories (BCL) for the identification of reactor core-melt accident mitigation measures at the Zion and Indian Point plants are summarized. Mitigation strategies have been identified that show promise of providing large reductions in consequences for specific accident sequences. However, without an overall risk analysis, it is not clear to what extent a given mitigation scheme reduces overall risk. The study evaluated filtered-vented containment systems, steam explosions, hydrogen burning, hydrogen control measures, melt/concrete and melt/MgO interactions, and meltdown phenomenology. Steam explosions have been determined to be unlikely to present a threat to containment at Zion and Indian Point. It has been determined that a one-time burn of the quantities of hydrogen expected in a meltdown accident with modest initial over-pressures could threaten containment integrity. Penetration of the basemat in a meltdown accident could not be confidently established; if penetration occurs, it will probably be after 3 to 4 days. A core retention device at these plants might give an additional delay of a few hours to a day.

NUREG/CR-1418: EXPERIMENTS WITH A VORTEX SHEDDING FLOWMETER IN TWO-PHASE AIR-WATER FLOW. TURNAGE, K. G. Oak Ridge National Laboratory. March 1982. 45pp. 8204290616. ORNL/NUREG/TM-3. 12898:146.

Experiments performed with a strain gauge-type vortex shedding flowmeter in two-phase vertical upflow and downflow in the Air-Water Test Facility at the Oak Ridge National Laboratory are described. Digital signals analysis techniques were used to evaluate the utility of the test meter for measuring two-phase flow. The studies indicate that vortex shedding can be used to produce clear, modulated signals in a generally homogeneous stream with relatively small quantities of a dispersed phase. At intermediate void fractions, however, the test meter produced meaningless or intermittent signals except when the flow velocity was very high.

NUREG/CR-1449: LMFBR AEROSOL RELEASE AND TRANSPORT PROGRAM. Quarterly Progress Report For October-December 1979. KRESS, T. S.; TOBIAS, M. L.

Oak Ridge National Laboratory. March 1982. 44pp. 8204150559. ORNL/NUREG/TM-3. 12708:006.

This report summarizes progress for the Liquid-Metal Fast Breeder Reactor (LMFBR) Aerosol Release and Transport (ART) Program sponsored by the Division of Reactor Safety Research of the Nuclear Regulatory Commission for the period October-December 1979. Topics discussed include (1) recent capacitor discharge vaporization (CDV) underwater tests conducted in the Fuel Aerosol Simulant Test (FAST) Facility to evaluate the disassembly process, including bubble dynamics and UO(2) vapor condensation and transport; (2) tests in the CRI-III vessel to evaluate UO(2) temperatures during melting and CDV discharge; (3) underwater tests in the CRI-III vessel to determine the effect of reduced xenon pressure on bubble line; (4) a single-component U(3)O(8) aerosol experiment using the plasma torch and two-component (U(3)O(8) and Na(2)O(x)) mixed-aerosol experiments in the Nuclear Safety Pilot Plant (NSPP); (5) calibration experiments and comparisons using the LASL-Stober spiral centrifugal; (6) predictive calculations of some sodium-burning experiments in CRI-II, and (7) a study of the relative importance of ordinary and thermal diffusion effects in UO(2) vapor bubbles generated in FAST experiments.

NUREG/CR-1476: PWR BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM-THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT FOR TEST 177. CLEMONS, V. D.; FLANDERS, R. M.; CRADDICK, W. G. Oak Ridge National Laboratory. March 1982. 50pp. 8205060039. ORNL/NUREG/TM-2. 13006:209.

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) test 177, which is part of the ORNL Pressurized-Water Reactor (PWR) Blowdown Heat Transfer Separate-Effects Program. The objective of the program is to investigate the thermal-hydraulic phenomenon governing the energy transfer and transport processes that occur during a loss-of-coolant accident in a PWR system.

The primary purpose of this report is to make the reduced instrument responses during test 177 available. The responses are presented in graphical form in engineering units and have been analyzed only to the extent necessary to assure reasonableness and consistency.

NUREG/CR-1497 ERR: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1978. * Brookhaven National Laboratory. January 22, 1982. 10pp. 8203040124. 12119:319.

Releases of radioactive mateials in airborne and liquid effluents from commercial light water reactors during 1978 have been compiled and reported. Data on solid waste shipments as well as selected operating information have been included. This report supplements earlier annual reports issued by the former Atomic Energy Commission and the Nuclear Regulatory Commission. The 1978 release data are compared with previous years releases in tabular form. Data covering specific radionuclides are summarized.

NUREG/CR-1534: PWR FLECHT-SEASET STEAM GENERATOR SEPARATE EFFECTS TASK DATA ANALYSIS AND EVALUATION REPORT. NRC/EPRI/Westinghouse Report No. 9. HOWARD, R. C.; HOCHREITER, L. E. Westinghouse Electric Corp. February 1982. 285pp. 8203040065. NP-1461. 12123:013.

This report presents the evaluation of the data from the Steam Generator Separate Effects Task of the Full-Length Emergency Cooling Heat Transfer Separate Effects and Systems Effects Test Program (FLECHT

SEASET). In this task a series of heat transfer tests were run on a model steam generator operating under simulated loss-of-coolant conditions. The model steam generator was made up of 32 full-length U-tubes instrumented with thermocouples to measure secondary fluid, tube wall, and primary steam temperatures. The separate effects tests measured steam generator bundle heat transfer with known boundary conditions to provide better understanding of the steam generator behavior in the systems effects tests. The test results are presented in NRC/EPRI/Westinghouse Report No. 4. This report describes the analysis of the data and an analytical model that adequately predicts the test data.

NUREG/CR-1546 V02: MANAGEMENT OF RADIOACTIVE WASTE GASES FROM THE NUCLEAR FUEL CYCLE. Volume II. Evaluation of Storage/Disposal Alternatives. PROUT, W. E.; DURANT, W. S.; EVANS, A. G.; et al. E. I. du Pont de Nemours & Co., Inc. January 1982. 104pp. 8201290022. DPST-81-1. 11740:030.

A basis for making decisions pertaining to the management of gaseous radioactive wastes (Kr-85, C-14, and I-129) from the nuclear fuel cycles, has been developed. Phase two, documented herein, focuses primarily on the final disposition of the waste forms and development of performance criteria. The most promising waste forms and repository options are described and then compared with general performance criteria being considered for inclusion in an EPA guideline related to storage of high-level waste. Long-range storage schemes for each of the waste gases are then recommended. With no long-term storage facility available in the near future, an above-ground interim, engineered storage facility for krypton-85, carbon-14, and iodine-129 is described in detail. A general methodology is set forth for assessing the risk of long-term options such as storage in a mined repository or in deep ocean sediments. Information required for such studies and further research for its development is described. A preliminary identification and comparison of the various alternatives for collection and storage/disposal of the resulting waste forms has been documented as Volume I.

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-1598: EVALUATION OF CATCH-PER-UNIT-EFFORT INDICES USED IN AQUATIC MONITORING PROGRAMS AT NUCLEAR POWER PLANT SITES. MCKENZIE, D. H. ; SKALSKI, J. R. ; SIMMONS, M. A. ; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1982. 113pp. 8208180237. PNL-3369. 14391:072.

Catch-per-unit-effort (CPUE) indices have been used as a fisheries management tool and in monitoring programs at nuclear power plants. An examination of CPUE techniques was conducted with the purpose of developing guidelines for evaluating monitoring programs and interpreting the resulting information. Data bases were selected for analysis from power plant monitoring programs which: (1) had an extensive data base; (2) made use of several fish sampling methods; and (3) had monitoring designs which incorporated replicate samples. At riverine sites, the emphasis centered on applied biological and field sampling; quantitative aspects were evaluated with graphics and coefficients of variation. At the remaining sites, emphasis was placed on statistics and a posteriori sampling designs and hypotheses. Data analyzed from selected programs do not provide evidence to support basic assumptions of CPUE to detect changes of reasonable magnitude, and capability to assess power plant-induced change through CPUE measurement. The findngs indicate that CPUE indices cannot be relied on as the sole bases for assessing population changes. Future approaches should be based on a realistic framework that integrates qualitative and quantitative components and recognizes the shortcomings of CPUE as a monitoring tool.

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-1628: QUARTERLY PROGRESS REPORT ON BLOWDOWN HEAT TRANSFER SEPARATE EFFECTS PROGRAM FOR APRIL-JUNE 1980. COOK, D. H. ; MAILEN, G. S. ; FLANDERS, R. M. ; et al. Oak Ridge National Laboratory. March 1982. 23pp. 8204300026. ORNL/NUREG/TM-4. 12921:057.

During this quarter, two transient film boiling tests were run - one in upflow, the other in downflow. The purpose of these tests was to provide experimental heat transfer data that could be used in an assessment of several film boiling correlations used in current thermal-hydraulic computer codes. The tests were designed to provide accurate posttest calculations of heat fluxes, surface temperatures, and local fluid conditions. The Upflow Film Boiling Test 3.03.6AR was run on May 21, 1980, in the THTF. This test resulted in single-phase fluid flow at the test section inlet (subcooled) during the time the bundle was in high-flow film boiling. Furthermore, no flow reversal occurred during the test. Preliminary posttest analysis indicated that the calculation of mass flows at the inlet to the test section should result in numbers with relatively small uncertainties.

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. 118pp. 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. D. Sandia Laboratories. CYBULSKIS, P. Battelle Memorial Institute, Columbus Laboratories. June 1982. 240pp. 8206240043. 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 V02: RISK ASSESSMENT METHODOLOGY DEVELOPMENT FOR WASTE ISOLATION IN GEOLOGIC MEDIA. Technical Review of Documents NUREG/CR-1262, NUREG/CR-1376, NUREG/CR-1377, NUREG/CR-1397 & NUREG/CR-1603. STEVENS, C. A.; FULLWOOD, R. R.; BASIN, S. L. Science Applications, Inc. February 1982. 123p. 8203090207. SAI-262-81-PA. 12176:217.

This project is an independent 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 five technical reports. Two of the documents pertain to geosphere simulation and three of the documents pertain to statistical methods including the application of Latin Hypercube Sampling.

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. 102pp. 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-1672 V04: RISK ASSESSMENT METHODOLOGY DEVELOPMENT FOR WASTE ISOLATION IN GEOLOGICAL MEDIA.Technical Review Of NUREG/CR-1636 Vol. 4,NUREG/CR-2344 And NUREG/CR-2343. STEVENS,C.A.; FULLWOOD,R.R.; AMIRIJAFARI,B.; et al. Science Applications, Inc. December 1982. 69pp. 8301100049. SAI-324-82-PA. 16751:165.

This project is an ongoing independent technical review of products prepared for the NRC by Sandia National Laboratories on the Risk Assessment Methodology Development for Waste Isolation in Geologic Media program. This report contains a review of three documents. They are NUREG/CR-1634, Vol. 4, concerned with the effects of variable hydrology on waste migration; NUREG/CR-2324, a user's manual for SWIFT; and NUREG/CR-2343, a user's manual for DNET.

In general, these reports exhibit high technical quality that characterizes the SNL work. They are tersely written with little condescension to the non-expert reader for understanding the physical situation being modeled. Indeed, the emphasis is on the mathematical procedures rather than the repository physics, leaving the adequacy of the results presented in many computer plots, pretty much to the interpretation of the reader. Other general comments have been presented previously, such as the data conservatisms, need for data that cannot be measured without disturbing the geometry, and the overall plan for use of the many codes developed in the program.

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-1687: ANALYTICAL MULTIOBJECTIVE DECISION METHODS FOR POWER PLANT SITING: A Review Of Theory And Applications. HOBBS, B. F. Brookhaven National Laboratory. February 1982. 125pp. 8203010390. BNL-NUREG-51204. 12075:001.

The objective of this report is to examine analytical multiple objective decision making techniques in the context of the power plant siting. Methods are compared on four general criteria: (1) theoretical validity, (2) flexibility, (3) results compared to other methods, and (4) ease of use. Emphasis here is on the theoretical merits and actual results of the techniques as discussed in the power plant siting, management science, and psychology literature.

NUREG/CR-1710 V01: A COMPUTERIZED PROCESS CONTROL SYSTEM FOR THE ORR-PSF IRRADIATION EXPERIMENT PART 1: OVERALL VIEW OF THE CONTROL SYSTEM. MERRIMAN, S. H. Oak Ridge National Laboratory. August 1982. 77pp. 8209270340. ORNL/NUREG/TM-4. 15512:185.

A dedicated process control computer has been implemented for regulating the metallurgical Pressure Vessel Wall Benchmark Facility (PSF) at the Oak Ridge Research Reactor. The purpose of the PSF is to provide reliable standards and methods by which to judge the radiation damage to reactor pressure vessel specimens. Benchmark data gathered from the PSF will be used to improve and standardize procedures for assessing the remaining safe operating lifetime of aging reactors. The computer sys controls the pressure vessel specimen environment in the presence of gamma heating so that in-vessel conditions are simulated. Instrumented irradiation capsules, in which the specimens are housed, contain temperature sensors and electrical heaters. The computer system regulates the amount of power delivered to the electrical heaters based on the temperature distribution within the capsules. Time-temperature profiles are recorded along with reactor conditions

NUREG/CR-1744 V01: STRUCTURED ASSESSMENT APPROACH (SAA) INPUT PACKAGE. Volume 1: Data-Gathering Handbook (Physical Security). WAHLER, P. S. Lawrence Livermore Laboratory. March 1982. 358pp. 8204160051. UCRL-53007 V01. 12716:017.

A description of the data-gathering process for the Structured Assessment Approach (SAA) Input Package is presented in this volume.

The Data-Gathering Handbook is divided into two phases, namely, Phase 1: Data-Collection and Phase 2: Data-Recording. In the data-collection phase, a sequence of questions in the handbook elicits the required information. The data-recording phase rearranges the data that have been collected into a format suitable for entering in a Tektronix 4050 Series Computer.

The appendix of this volume demonstrates the use of the SAA Data-Gathering Handbook in a hypothetical nuclear facility.

NUREG/CR-1744 V02: STRUCTURED ASSESSMENT APPROACH (SAA) INPUT PACKAGE. Volume 2: Data-Gathering Forms (Physical Security). WAHLER, P. S. Lawrence Livermore Laboratory. March 1982. 120pp. 8204210646. UCRL-53007 V02. 12799:180.

A complete set of Structured Assessment Approach (SAA) Input Package Data-Gathering Forms is given in the appendix of this volume.

The Data-Gathering Forms are divided into two sections, namely, Section I: Data-Collection and Section II: Data-Recording. In the Data-Collection Section, the forms offer the analyst a sequence of questions which elicit the information required from a facility. The forms in the Data-Recording Section offer a suitable format for arranging the collected data, which is to be entered into the Tektronix 4050 Series Computer.

NUREG/CR-1744 V03: STRUCTURED ASSESSMENT APPROACH (SAA) INPUT PACKAGE. Volume 3: User's Manual (Physical Security). ORVIS, W. J. Lawrence Livermore Laboratory. March 1982. 117pp. 8204150580. UCRL-53007 V03. 12687:009.

The operation and use of the Structured Assessment Approach (SAA) Input Package programming written for a Tektronix 4050 Series Computer is described. The programming consists of the Facility Description Program (described in this volume) and its continuation, the Accounting System Program (planned), plus several service routines. These programs generate the input files that are used by the SAA codes in a mainframe computer, such as the CDC 7600 at the Lawrence Livermore National Laboratory.

NUREG/CR-1756 V01: TECHNOLOGY, SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR RESEARCH AND TEST REACTORS. Main Report. KONZEK, G. J.; LUDWICK, J. D.; KENNEDY, W. E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1982. 360pp. 8203190442. 12369:032.

Safety and Cost Information is developed for the conceptual decommissioning of two representative licensed nuclear research and test reactors. Three decommissioning alternatives are studied to obtain comparisons between costs (in 1981 dollars), occupational radiation doses, potential radiation dose to the public, and other safety impacts. The alternatives considered are: DECON (immediate decontamination), SAFSTOR (safe storage followed by deferred decontamination), and ENTOMB (entombment). The study results are presented in two volumes. Volume 1 (Main Report) contains the results in summary form. Volume 2 (Appendices) contains the detailed data that support the results given in Volume 1, including unit-component data.

NUREG/CR-1756 V02: TECHNOLOGY, SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR RESEARCH AND TEST REACTORS. KONZEK, G. J.; LUDWICK, J. D.; KENNEDY, W. E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1982. 685pp. 8203190457. 12367:066.

Safety and Cost Information is developed for the conceptual decommissioning of two representative licensed nuclear research and test reactors. Three decommissioning alternatives are studied to obtain comparisons between costs (in 1981 dollars), occupational radiation doses, potential radition dose to the public, and other safety impacts. The alternatives considered are: DECON (immediate decontamination), SAFSTOR (safe storage followed by deferred decontamination), and ENTOMB (entombment). The study results are presented in two volumes. Volume 1 (Main Report) contains the results in summary form. Volume 2 (Appendices) contains the detailed data that support the results given in Volume 1, including unit-component data.

NUREG/CR-1760: A SURVEY OF PROPOSED FUNCTIONAL REQUIREMENTS FOR A DISTURBANCE ANALYSIS AND SURVEILLANCE SYSTEM. SIDES, W. H.; OH, C. B.; KNIGHT, P. F. Oak Ridge National Laboratory. March 1982. 30pp. 8204150556. ORNL/NUREG/TM-3. 12690:303.

A program to enhance the capabilities of operators of nuclear power plants is being pursued by the U.S. Nuclear Regulatory Commission

(NRC). The program includes improvements in plant monitoring, diagnostic and corrective action aids, operator-process communication, and operator training. Concerning diagnostic aids, a system to provide surveillance and diagnosis of plant disturbances is being considered. The goal of a DASS is to monitor the plant for the approach or occurrence of such disturbances and to assist the operator in returning the plant to normal operation or in mitigating the consequences of a failure condition or misoperation. This report is the result of a survey of functional requirements currently proposed for a DASS. Suggestions for the scope of a DASS range from the relatively simple and straightforward to the far-reaching and complex. No judgment as to adequacy or necessity of any requirement is made by the authors; inclusion of a requirement in this report does not imply concurrence that the requirement should be pursued in the development of a DASS. While not exhaustive, the survey is intended to indicate the scope and breadth of existing suggestions regarding DASS functions.

NUREG/CR-1786: MAINTAINABILITY ANALYSIS PROCEDURE (MAP). ENGI, D. Sandia Laboratories. March 1982. 70pp. 8204020020. SAND80-2497. 12512:245.

The initial development of a dynamic Monte Carlo modeling procedure that incorporates conditionally linked time-dependent component failure and repair phenomena into the analysis of reactor safety or reactor safeguards systems is described. This procedure, the Maintainability Analysis Procedure (MAP), consists of a network and a procedure for analyzing the network. The maintainability network represents not only the topology of the safety or safeguards system but also a time-varying sequence of events, activities, and decisions that reflect the user-specified characteristics of the included phenomena. Examples of maintainability analyses using MAP in application to a simple reactor sub-system are presented and compared for increasingly complex event sequences.

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. 13001:109.

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-1830: USER'S MANUAL FOR STRIPE: A COMPUTER CODE FOR SIMULATING STRIPED BASS YOUNG-OF-THE-YEAR POPULATION IN THE HUDSON RIVER. ERASLAN, A. H. ; SHARP, R. D. ; VAN WINKLE, W. Oak Ridge National Laboratory. January 1982. 146pp. 8202050245. ORNL/NUREG/TM-2. 11831:249.

The structure and operational features of the main program and subroutines of the STRIPE computer code are described. All the necessary information and instructions are presented for implementing the computer code in simulating the daily variations and the longitudinal distributions of the various life stages of the young-of-the-year striped bass population in the Hudson River. Complete samples of input data and output results are given for 1973 conditions.

NUREG/CR-1846 ADD C: BWR REFILL-REFLOOD PROGRAM TASK 4.4 - CCFL/REFILL SYSTEM EFFECTS TESTS (30 F SECTOR) EXPERIMENTAL TASK PLAN. Addendum C - 30 F SSTF CCFL/Refill, BWR/6 System Response Test Plan. SCHUMACHER, D. G. General Electric Co. January 1982. 32pp. 8201290107. EPRI NP-1525. 11750:125.

Addendum C to the 30 Degree Sector Test Facility (30 SSTF) Experimental Task Plan defines the objectives, specific test conditions, and summarized test operating procedures for the transient loss of coolant accident (LOCA) simulation tests performed in the 30 Degree SSTF under the BWR Refill-Reflood Program.

NUREG/CR-1846 ADD D: BWR REFILL-REFLOOD PROGRAM TASK 4.4-CCFL/REFILL SYSTEM EFFECTS TESTS (30 F SECTOR) EXPERIMENTAL TASK PLAN. Addendum D-SSTF CCFL/Refill with ECCS Variation Test Plan (BWR/4 ECCS

Geometry). SCHUMACHER, D. G. General Electric Co. January 1982. 34pp. 8201220269. GEAP-24893-4. 11659:299.

Addendum D to the 30 degree Sector Experimental Task Plan provides definition of objectives, specific test conditions, and summarized test operating procedures for separate effect and system tests performed in the 30 degree Steam Sector Test Facility (30 degree SSTF) with Lower Pressure Coolant Injection (LPCI) location, and Low Pressure Core Spray (LPCS) geometry representative of BWR/4 configuration.

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-1853 V03: DISTRIBUTION COEFFICIENTS FOR RADIONUCLIDES IN AQUATIC ENVIRONMENTS. Comparison Of Dialysis And Constant Shaking Experiments For A Sediment-Water System From Cattaraugus Creek, New York. SIBLEY, T. H. Washington Univ. March 1982. 40pp. 8204010540. 12489:259.

This report presents the results of constant shaking and dialysis experiments that were conducted with sediment-water systems from Cattaraugus Creek to determine distribution coefficients for (59)Fe, (60)Co, (85)Sr, (137)Cs, (207)Bi, (237)Pu, and (241)Am. Results are discussed in terms of adsorption time, effects of sediment concentration and differences between dialysis and constant shaking experiments. Slow adsorption processes increase the calculated K(d) values during the experiment and K(d) values in the dialysis experiments are slightly lower than in constant shaking experiments. Increased sediment concentrations decrease the K(d) values of (137)Cs and (241)Am but appear to have an opposite effect for (60)Co.

NUREG/CR-1853 V04: DISTRIBUTION COEFFICIENTS FOR RADIONUCLIDES IN AQUATIC ENVIRONMENTS. Effect Of pH On Adsorption. SIBLEY, T. H.; SANCHEZ, A. L.; WURTZ, E. A.; et al. Washington, Univ. of. February 1982. 56pp. 8203010024. 12074:001.

Distribution coefficients are often used to estimate the partitioning of radionuclides between soluble and particulate phases. Although distribution coefficients are often considered to describe equilibrium distributions, the values obtained can be affected significantly by experimental conditions such as pH, sediment concentration, the concentration of complexing ligands and competing ions, and time of adsorption. We studied the effect of pH (pH 4 to pH 10) on the adsorption of (57)Co, (85)Sr, (106)Ru, (137)Cs, (237)Pu, (241)Am, and (244)Cm in laboratory sediment-water systems. Natural sediments and water were collected from a variety of environments and the adsorption of several isotopes was found to be site specific. Adsorption of most radionuclides is strongly pH dependent and increased adsorption coincides with the formation of metal hydrolysis products.

In the pH range of natural waters, (57)Co and (106) Ru were most affected by changes in pH.

NUREG/CR-1853 V05: DISTRIBUTION COEFFICIENTS FOR RADIONUCLIDES IN AQUATIC ENVIRONMENTS: Effect of Sediment Concentration on Distribution Coefficients. SANCHEZ, A. L. ; SIBLEY, T. H. ; WURTZ, E. A. ; et al. Washington, Univ. of. March 1982. 48pp. 8204020093. 12492:278.

Distribution coefficients are often used to estimate the partitioning of radionuclides between soluble and particulate phases. Although distribution coefficients are often considered to describe equilibrium distributions the values obtained can be affected significantly by experimental conditions such as pH, sediment concentration, the concentration of complexing ligands and competing ions, and time of adsorption. We studied the effect of sediment concentration of the adsorption of (57)Co, (85)Sr, (106)Ru, (137)Cs, (237)Pu, (241)Am, and (244)Cm in laboratory sediment-water systems using natural sediments and water from a variety of environments. For most of these radionuclides in sediment-water systems studied, the Kd values decreased as the sediment concentration was increased. Two exceptions, (137)Cs and (57)Co in the Hudson River samples, were noted. The results suggest that the adsorption pattern for these radionuclides is site specific. It is therefore important to have adsorption data for the specific sediment-water system being considered in order to model the hydrologic transport of these radionuclides in this particular environment. More experiments, especially at the lower, environmentally important sediment concentrations, are needed to fully evaluate the relative importance of sediment concentration on the partitioning of radionuclides in natural waters.

NUREG/CR-1864 V01: A STUDY OF NONEQUILIBRIUM FLASHING OF WATER IN A CONVERGING-DIVERGING NOZZLE. Volume I-Experimental. ABUAF, N. ; ZIMMER, G. A. ; WU, B. J. C. Brookhaven National Laboratory. March 1982. 144pp. 8204290492. BNL-NUREG-51317. 12902:171.

A steady water loop with well controlled flow and thermodynamic conditions was designed, built, and made operational for the measurement of net vapor generation rates under nonequilibrium conditions. The test section consists of a converging-diverging nozzle with 49 pressure taps and two observation windows at the exit. Pressure distributions, photographic observations, diametrical averaged centerline void fraction distributions, detailed transverse distributions of the chordal averaged void fractions at 27 axial locations, and area averaged void fraction distributions along the nozzle were recorded under various flashing conditions. The effects of the various parameters such as inlet pressure (140 $<$ P(in) $<$ 766 kPa), inset temperature (100 degrees $<$ T(in) $<$ 149 degrees C), mass flux (1000 $<$ G(in) $<$ 6720 kg/m(2) s), and back pressure on the pressure and void distributions were investigated and are reported here. Since no information on the phase velocities was recorded during the present experiments, the calculation of vapor generation rates from the available experimental data involved the assumption of a slip model between the two phases.

NUREG/CR-1864 V02: A STUDY OF NONEQUILIBRIUM FLASHING OF WATER IN A CONVERGING-DIVERGING NOZZLE. Volume 2-Modeling. WU, B. J. C. ; ABUAF, N. ; SAHA, P. Brookhaven National Laboratory. March 1982. 160pp. 8204080092. BNL-NUREG-51317. 12614:214.

A steady water loop with well controlled flow and thermodynamic conditions was designed, built, and made operational for the measurement of net vapor generation rates under nonequilibrium conditions. The test section consists of a converging-diverging nozzle with 49 pressure taps and two observation windows at the exit. Pressure distributions, photographic observations, diametrical averaged centerline void fraction distributions, detailed transverse distributions of the chordal averaged void fractions at 27 axial locations, and area averaged void fraction distributions along the nozzle were recorded under various flashing conditions. The effects of the various parameters such as inlet pressure (140 less than P(in) less than 766 kPa), inlet temperature (100 degrees less than T(in) less than 149 degrees C), mass flux (1000 less than G(in) less than 6720 kg/m(2) s), and back pressure on the pressure and void distributions were investigated and are reported here. Since no information on the phase velocities was recorded during the present experiments, the calculation of vapor generation rates from the available experimental data involved the assumption of a slip model between the two phases.

NUREG/CR-1869: DISTRIBUTION COEFFICIENTS FOR RADIONUCLIDES IN AQUATIC ENVIRONMENTS: Final Summary Report. SCHELL, W. R. ; SIBLEY, T. H. Washington, Univ. of. March 1982. 30pp. 8204020075. 12493:126.

As part of this research project, 15 topical reports were prepared previously for distribution by the Nuclear Regulatory Commission. This final summary report provides a synopsis of each of the earlier reports including relevant tables of K(d) values and the most significant recommendations and conclusions. Details of the methodology, specific experiments, background literature, and results are found in the reports for each radionuclide. This report serves as a guide to the information in the individual topical reports.

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-1891: RELIABILITY ANALYSIS OF CONTAINMENT STRENGTH. Sequoyah And McGuire Ice Condenser Containments. GREIMANN, L. ; FANOUS, F. ; SABRI, A. Iowa State Univ. August 1982. 268pp. 8209100222. IS-4753. 14768:258.

The Sequoyah and McGuire ice condenser containment vessels were designed to withstand pressures in the range of 12 to 15 psi. Since pressures of the order of 28 psi were recorded during the Three Mile Island incident, a need exists to more accurately define the strength of these vessels. A best estimate and uncertainty assessment of the strength of the containments was performed by applying the second moment reliability method. Material and geometric properties were supplied by the plant owners. A uniform static internal pressure was assumed. Gross deformation was taken as the failure criterion. Both approximate and finite element analyses were performed on the axisymmetric containment structure and the penetrations. The predicted strength for the Sequoyah vessel is 60 psi with a standard deviation of 8 psi. For McGuire, the mean and standard deviation are 84 psi and 12 psi, respectively. In an Addendum, results by others are summarized and compared and a preliminary dynamic analysis is presented.

NUREG/CR-1895: AN INVESTIGATION OF THE DEGREE OF EQUILIBRIUM OF THE LONG-LIVED URANIUM-238 DECAY CHAIN MEMBERS IN AIRBORNE AND BULK URANIUM ORE DUSTS. JACKSON, P. O. ; THOMAS, C. W. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1982. 55pp. 8209100234. PNL-3696. 14767:319.

The degree of disequilibrium among (238)U decay chain members in some airborne dusts and typical ores has been established by precise radiochemical analyses. The information is necessary to evaluate the lung dose model currently used for estimating the effect of the inhalation of uranium ore dust. The particle size distributions of airborne decay chain components in dusts at one uranium mill have been investigated. Statistically significant disequilibria were observed for (239)Th, (226)Ra, and (210)Pb in both airborne dusts and composite ore samples. With the exception of ore from one mill in the United States, most of the daughter concentrations in powdered ore composites were within 10% of (238)U. In airborne dusts, the concentration of (226)Ra was typically below (238)U; the minimum (226)Ra concentration observed for airborne ore dusts was 56% of equilibrium. A statistically significant particle size dependence was observed for (226)Ra/(238)U ratios in several airborne dusts collected at a uranium mill.

NUREG/CR-1912: CHARACTERIZATION AND EVALUATION OF A TURBINE DISC SECTION CONTAINING A SERVICE CRACK. GOLDBERG, A. ; ALBERTSON, S. L. ; FEDRICK, I. A. Lawrence Livermore Laboratory. January 1982. 40pp. 8201260323. UCID-18699. 11692:289.

Cracking had occurred in the keyway of a low-pressure turbine disc. We evaluated a section of the disc to determine whether the tensile and Charpy V-notch (CVN) toughness properties in the hub region corresponded to those originally specified for this disc. We also characterized the microstructure of the steel and analyzed the bore surface.

Room-temperature tensile properties and FATT(50) were found to be essentially unchanged from the specified values. The steel was relatively clean. However, excessive segregation was present and was reflected by variations in grain size, microhardness, sulfur distribution, and inclusion content. Using x-ray analysis, we identified the presence of an MoS(2) deposit and an FeCr(2)O(4) corrosion film along the bore. We propose that excessive segregation (inclusions, alloying elements, embrittling constituents) contributes to excessive scatter in our CVN results.

Segregation and MoS(2) lubricant may both contribute to the cracking problem.

NUREG/CR-1935: QUANTITATIVE FAULT TREE ANALYSIS USING THE SET EVALUATION PROGRAM (SEP). OLMAN,M.D. Sandia Laboratories. November 1982. 50pp. 8212270165. SAND80-2712. 16555:267.

This report describes the use of a program to aid in the analysis of fault trees. The user has the option of executing one or more of a number of procedures which have been developed to perform the quantitative analysis of a fault tree.

NUREG/CR-1956: A VARYING ELASTICITY MODEL OF ELECTRICITY DEMAND WITH GIVEN APPLIANCE SATURATION. CHERN,W.S.; JUST,R.E.; CHANG,H.S. Oak Ridge National Laboratory. July 1982. 64pp. 8209270089. ORNL/NUREG/TM-4. 15516:256.

This report presents the third version of the State-Level Electricity Demand (SLED) Model developed at the Oak Ridge National Laboratory for the Nuclear Regulatory Commission. Specific improvements over previous versions of the SLED model are as follows: (1) A theoretical framework for estimating electric appliance choices and utilization of those appliances at the aggregate level is developed. These refinements enable the model to capture the detailed underlying behavior of electricity consumers and to deal with the effect of market penetration of energy saving technologies on electricity demand. (2) The linkage between average price and marginal price is instituted. Thus the model as estimated can be interpreted using either average or marginal price. The marginal price elasticities are derived from the average price elasticities and presented in the report. (3) Important determinants of price elasticities have been identified and the elasticities of demand are specified to be variable, rather than constant, among states in a region as well as over time. The formulation of variable elasticities permits the estimation of demand coefficients for a wide range of circumstances, such as in utility service areas.

NUREG/CR-1968: SWIFT SELF-TEACHING CURRICULUM. FINLEY,N.C. Sandia Laboratories. March 1982. 170pp. 8205130254. SAND81-0410. 13089:079.

This document contains a series of sample problems and solutions for the Sandia Waste-Isolation Flow and Transport (SWIFT) model developed at Sandia National Laboratories for the Risk Methodology for Geologic Disposal of Radioactive Waste Project (FIN A-1192). With this document and the SWIFT User's Manual, the student may familiarize himself with the code, its capabilities and limitations. When the student has completed this curriculum, he or she should be able to prepare data input for SWIFT and have some insights into interpretation of the model output. 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 (FIN A-1158).

NUREG/CR-1989: ANALYSIS OF HYPOTHETICAL SEVERE CORE DAMAGE ACCIDENTS FOR ZION PRESSURIZED WATER REACTOR Docket Nos. 50-295 And 50-304. HASKIN,F.E.; DARBY,J.L.; MURFIN,W.B. Sandia Laboratories. December 1982. 192pp. 8301120115. SAND81-0504. 16784:222.

This report describes analyses of the response of a Pressurized

Water Reactor at the Zion Plant to hypothetical core meltdown sequences. The analyses consider the progression of core meltdown, containment response, and consequences to the public for many specific accident sequences within the categories of Loss of Coolant Accidents (LOCAs), transient-initiated accidents, and containment-bypass accidents. The report does not deal with the probability of the accidents occurring. Strategies for accident management and mitigation of consequences are suggested. Uncertainties in the calculated plant responses are described.

NUREG/CR-2000 V01 N1: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of January 1982. * Oak Ridge National Laboratory. March 1982. 33pp. 8204010531. ORNL/NSIC-200. 12489:165.

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 assigned by the NSIC staff when the summaries are prepared for computer entry.

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. 105pp. 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-2000 V01 N6: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of June 1982. * Oak Ridge National Laboratory. July 1982. 104pp. 8207290396. ORNL/NSIC-200. 14193:218.

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 N8: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of August 1982. * Oak Ridge National Laboratory. September 1982. 180pp. 8210150577. ORNL/NSIC-200. 15723:113.
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 N9: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of September 1982. * Oak Ridge National Laboratory. October 1982. 97pp. 8211160485. ORNL/NSIC-200. 16114:219.
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 V01N11: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of November 1982. * Oak Ridge National Laboratory. December 1982. 127pp. 8301100052. ORNL/NSIC-200. 16751:235.
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. 147pp. 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-2029: SODIUM CONCRETE ABLATION MODEL. SUO-ANTTILA, A. Sandia Laboratories. January 1982. 56pp. 8201270409. SAND81-0415. 11718:291.

A model has been developed to explain and predict the behavior of the interaction between sodium and basaltic concrete. A fundamental assumption of the model is that transport of the primary reactants, e.g., water and sodium, occurs in the vapor phase within the concrete pores and is a diffusion process. A simultaneous solution to the continuity, diffusion, energy, momentum and chemical kinetics equations are obtained to yield penetration, thermal energy and gas evolution rates.

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-2060: EXPERIMENTAL DATA REPORT FOR FLASHING TRANSIENTS. SAM, R. G.; CROWLEY, C. J. Creare, Inc. February 1982. 58pp. 8204280560. CREARE TN-330. 12876:226.

This report presents experimental results of the blowdown and refill of 1/5, 1/15, and 1/30-scale model pressurized water reactor vessels. Data from 116 tests investigating the effects of initial vessel pressure, ECC injection rate, ECC injection temperature, initial vessel fluid temperature, initial vessel fluid mass, containment pressure, reverse core steam flow, cold leg break size, hot leg break size, downcomer gap size, and plenum volume are described. These data provide an extensive data base for code assessment. Key phenomena studied include phase separation, condensation, heat transfer and momentum exchange.

NUREG/CR-2071: COMPARISON OF THERMAL-HYDRAULIC RESPONSE IN THE SEMISCALE, THTF AND FLECHT FACILITIES. HANSON, R. G. EG&G, Inc. January 1982. 37pp. 8201210139. EGG-2093. 11646:317.

The Semiscale Program performed numerous simulations of large break loss-of-coolant accidents. Since several modifications to the

Semiscale Mod-1 system were made during the conversion to the Mod-3 facility, comparisons were performed to evaluate the effect of various configuration differences on system thermal-hydraulic response. Mod-3 experimental results were also compared to test data from other facilities (the Thermal-Hydraulic Test Facility and the FLECHT-SET facility) to determine if the data were repeatable. The importance of comparing data from various facilities lies in identifying the effects that differences in scale and facility configuration have on system response to transient events. The analysis identified disparities in some test results due to configuration differences, especially core length. Core thermal-hydraulic response from blowdown and core thermal-hydraulic response from blowdown and reflood separate effects tests were shown to be repeatable to the extent that the experiments were performed with similar boundary conditions in the different systems.

NUREG/CR-2080: A REVIEW OF H2 DETECTION IN LIGHT WATER REACTOR CONTAINMENTS. NEIDEL, E. C.; GOVER, J. E.; CASTLE, J. G. Sandia Laboratories. February 1982. 63pp. 8203190084. SAND81-0326. 12366:111.

Hydrogen detection systems are being installed in existing LWRs and are planned for new plants. This study outlines the potential uses of hydrogen concentration measurements in nuclear power plant containment buildings including the control of hydrogen mitigation systems. The types of hydrogen sensors that are available commercially are reviewed and our principal concerns about the limitations of these sensors for nuclear power plant applications are identified. We suggest sensor design modifications that have potential for improving the performance of existing hydrogen sensor systems. We describe the sensing methods and sensor technology that utilities are employing for hydrogen measurement. An alternative in-containment measurement system is also discussed.

NUREG/CR-2081: RISK METHODOLOGY FOR GEOLOGIC DISPOSAL OF RADIOACTIVE WASTE: THE NWFT/DVM COMPUTER CODE USERS MANUAL. CAMPBELL, J. E.; LONGSINE, D. E.; CRANWELL, R. M. Sandia Laboratories. January 1982. 100pp. 8202030131. SAND81-0886. 11810:001.

The Network Flow and Transport/Distribution Velocity Method (NWFT/DVM) computer code has been developed to simulate ground water and contaminant transport. It is used in conjunction with the Sandia Waste Isolation Flow and Transport (SWIFT) model. SWIFT simulation of a depository breachment scenario provides the boundary condition for the ground water flow model in NWFT/DVM. NWFT/DVM can then recreate the scenario velocity field as a network of one-dimensional segments through which radio-nuclides are transported. An important application of NWFT/DVM is that repeated runs necessitated by varying input parameters for a scenario can be accomplished with reasonable computer costs. The Distributed Velocity Method (DVM) is incorporated to provide flexibility and efficiency in solving the radionuclide transport problem. It allows for the transport of decay chains of any length, with isotopes having different retardations, and with source rates being leach- or solubility-limited.

NUREG/CR-2088: A LITERATURE REVIEW OF OCCUPATIONAL LICENSING APPLIED TO INDUSTRIAL RADIOGRAPHY. TURNAGE, J. J. Central Florida, Univ. of,. February 1982. 35pp. 8203030007. 12110:305.

This report summarizes the literature on the licensing of individuals in various occupations to determine whether licensing of individual radiographers might improve safety performance and reduce radiation overexposures.

The review discusses (1) previous NRC considerations of licensing radiographers, (2) the literature on occupational licensing, (3) studies of factors that relate to good safety records, and (4) experiences from the NRC's reactor operator licensing program. Findings are applied to the possible licensing of industrial radiographers.

The review concludes that licensing may be a means to assure adequate safety training for radiographers, but it is unlikely that licensing will improve a radiographer's motivation to perform safely. Improved regulation of management may be the most effective means of assuring safety.

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-2127 V03: REACTOR SAFETY RESEARCH PROGRAMS. Quarterly Report, July-September 1981. EDLER, S.K. Battelle Memorial Institute, Pacific Northwest Laboratory. January 1982. 70pp. 8203040006. PNL-3810-3. 12122:181.

This document summarizes the work performed by Pacific Northwest Laboratory (PNL) from July 1 through September 30, 1981, for the Division of Accident Evaluation, U.S. Nuclear Regulatory Commission (NRC). Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasiblility of determining the strength of structural graphite, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE and probabilistic fracture mechanics, and assessing the integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. Experimental data and analytical models are being provided to aid in decision-making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Core thermal models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Fuel assemblies and analytical support are being provided for experimental programs at other facilities. These programs include loss-of-coolant accident (LOCA) simulation tests at the NRU reactor, Chalk River, Canada; fuel rod deformation, severe fuel damage, and postaccident coolability tests for the ESSOR reactor Super Sara Test Program, Ispra, Italy; the instrumented fuel assembly irradiation program at Halden, Norway; and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory (INEL), Idaho Falls, Idaho. These programs will provide data for computer modeling of reactor systems and fuel performance during various abnormal operating conditions.

NUREG/CR-2127 V04: REACTOR SAFETY RESEARCH PROGRAMS. Quarterly Report, October-December 1981. EDLER, S. K. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1982. 87pp. 8204020027. PNL-3810-4. 12492:191.

This document summarizes the work performed by Pacific Northwest Laboratory (PNL) from October 1 through December 31, 1981, for the Division of Accident Evaluation, U.S. Nuclear Regulatory Commission (NRC). Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining the strength of structural graphite, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. Experimental data and analytical models are being provided to aid in decision-making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Core thermal models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Fuel assemblies and analytical support are being provided for experimental programs at other facilities. These programs include loss-of-coolant accident (LOCA) simulation tests at the NRU reactor, Chalk River, Canada; fuel rod deformation, severe fuel damage, and postaccident coolability tests for the ESSOR reactor Super Sara Test Program, Ispra, Italy; the instrumented fuel assembly irradiation program at Halden, Norway; and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory (INEL), Idaho Falls, Idaho. These programs will provide data for computer modeling of reactor system and fuel performance during various abnormal operating conditions.

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-2134: BWR REFILL-REFLOOD PROGRAM, TASK 4.7-CONSTITUTIVE CORRELATIONS FOR SHEAR AND HEAT TRANSFER FOR THE BWR VERSION OF TRAC. ANDERSON, J. G. M.; CHU, K. K. General Electric Co. November 1982. 96pp. 8212160762. EPRI NP-1582. 16464:190.

TRAC (Transient Reactor Analysis Code) is a computer code for best estimate analysis of the thermal hydraulic conditions in a reactor system. The constitutive correlations for shear and heat transfer in the boiling water reactor (BWR) version of TRAC are described.

A new model, that accounts for the effect of phase and velocity profiles, has been developed for the interfacial shear and a new set of constitutive correlations are derived. Improvements have been made to the heat transfer in the area of subcooled boiling, boiling transition, and thermal radiation.

NUREG/CR-2141 V03: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM. Quarterly Progress Report For July-September 1981. WHITMAN, G. D. ; BRYAN, R. H. Oak Ridge National Laboratory. March 1982. 138pp. 8204290623. ORNL/TM-8145. 12898:198.

THhe Heavy-Section Steel Technology 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 five tasks: (1) program administration and procurement, (2) fracture mechanics analyses and investigations, (3) investigations of irradiated materials, (4) thermal-shock investigations, and (5) pressure vessel investigations. Three-dimensional fracture mechanics codes were applied to benchmark and vessel problems. Subcontractors investigated dynamic fracture analysis and transitional toughness behavior. Investigations of properties of irradiated material include statistical analysis of Charpy data and continued irradiation of specimens. Preparations for thermal-shock experiment TSE-6 continued, and further overcooling accident problems were studied. Preparations for the low-upper-shelf intermediate vessel test V-8A are continuing. Further work on design and analyses related to the pressurized thermal-shock test program were performed.

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-2144. A COMPENDIUM OF COMPUTER CODES FOR LIGHT WATER REACTOR ANALYSIS. BIENARZ, P. P. ; PRASSINOS, P. G. ; ENGI, D. Sandia Laboratories. March 1982. 250pp. 8204020065. SAND81-1142. 12511:261.

A compilation of computer codes that can be used to analyze the effects of postulated accidents and failure phenomena upon light water reactor systems is presented. From a safeguards perspective, these codes can be used in the determination of the consequences of acts of sabotage, and therefore, they play a key role in ranking vital areas. Each code is described in terms of its function, its limitations, its use and comparison of the ouput from the code with experimental observations. A list of applicable reference documents related to each code is also presented. Primary emphasis is placed on nuclear power plant loss-of-coolant accident analyses, nuclear power plant transient analyses, and miscellaneous (nontransient) analyses. A tabulation of available codes and their purpose facilitates use of the report in identifying codes applicable to a particular problem.

NUREG/CR-2162: MINICOMPUTER CAPABILITIES RELATED TO METEOROLOGICAL ASPECTS OF EMERGENCY RESPONSE. RAMSDELL, J. V. ; ATHEY, G. F. ; BALLINGER, M. Y. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1982. 87pp. 8203030247. PNL-3773. 12119:044.

The purpose of this report is to provide the NRC staff involved in reviewing licensee emergency response plans with background information on the capabilities of minicomputer systems that are related to the collection and dissemination of meteorological information. The treatment of meteorological information by organizations with existing emergency response capabilities is described, and the capabilities, reliability, and availability of minicomputers and minicomputer systems are discussed.

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 V03: PHYSICS OF REACTOR SAFETY. Quarterly Report, July - September 1981. * Argonne National Laboratory. January 1982. 20pp. 8201270405. ANL-81-29 V03. 11719:290.

This quarterly progress report summarizes work done during the months of July-September 1981 in Applied Physics and Components Technology Divisions for the Division of Reactor Safety Research. 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 accidents under natural convection conditions. An executive summary is provided including a statement of the findings and recommendations of the report.

NUREG/CR-2181 V04: PHYSICS OF REACTOR SAFETY. Quarterly Report, October-December 1981. * Argonne National Laboratory. April 1982. 28pp. 8205030657. ANL-81-29 V04. 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 t.e Reactor Safety Appraisals Section. Work on reactor core thermal-hy. ulics 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-2182 V02: STATION BLACKOUT AT BROWNS FERRY UNIT ONE-IODINE AND NOBLE GAS DISTRIBUTION AND RELEASE. WICHNER, R.P.; WEBER, C.F.; LORENZ, R.A.; et al. Oak Ridge National Laboratory. September 1982. 261pp. 8210210034. ORNL/NUREG/TM-4. 15773:001.

This is the second volume of a report describing the predicted response of Unit 1 of the Browns Ferry Nuclear Plant to a postulated Station Blackout, defined as a loss of offsite power combined with a failure of all onsite emergency diesel-generators to start and load. The Station Blackout is assumed to persist beyond the point of battery exhaustion and the completely powerless state leads to core uncovery, meltdown, reactor vessel failure, and failure of the primary containment by overtemperature-induced degradation of the electrical penetration assembly seals. The sequence of events is described in Volume 1; the material in this volume deals with the analysis of fission product noble gas and iodine transport during the accident. Factors which affect the fission product movements through the series of containment design barriers are reviewed. For a reactive material such as iodine, proper assessment of the rate of movement requires determination of the chemical changes along the pathway which alter the physical properties such as vapor pressure and solubility and thereby affect the transport rate. A methodology for accomplishing this is demonstrated in this report.

NUREG/CR-2183: DISPERSED FILM FLOW BOILING OF HIGH PRESSURE WATER IN A ROD BUNDLE. MORRIS, D.G.; MULLINS, C.B.; YODER, G.L. Oak Ridge National Laboratory. September 1982. 88pp. 8210150567. ORNL/TM-7864. 15716:246.

The following six dispersed flow film boiling correlations were assessed using data from the third ORNL transient film boiling experiment (Test 3.08.6C) conducted in the THTF: a. Dougall-Rohsenow,

b. Dougall-Rohsenow (with Prandtl number evaluated at the wall temperature, as used in RELAP4-MOD7), c. Groeneveld 5.9, d. Groeneveld 5.7, e. Groeneveld-Delorme, f. Condie-Bengston IV. The correlations were evaluated with 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 indiviual correlations often overpredict the heat transfer coefficients, while the Groeneveld 5.7, Groeneveld 5.9, and Condie-Bengston IV correlations tend to be in good agreement with the data. The Groeneveld-Delorme correlation underpredicts the data. Based on correlation comparison results, a best-estimate heat transfer logic was formulated for the dispersed flow film boiling regime.

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-2185: SALE-3D: A SIMPLIFIED ALE COMPUTER PROGRAM FOR CALCULATING THREE-DIMENSIONAL FLUID FLOW. AMSDEN, A. A.; RUPPEL, H. M. Los Alamos Scientific Laboratory. January 1982. 147pp. 8201200621. LA-8905-MS. 11623:125.

This report presents a simplified numerical fluid-dynamics computing technique for calculating time-dependent flows in three dimensions. An implicit treatment of the pressure equation permits calculation of flows far subsonic without stringent constraints on the time step. In addition, the grid vertices may be moved with the fluid in Lagrangian fashion or held fixed in an Eulerian manner, or moved in some prescribed manner to give a continuous rezoning capability. This report describes the combination of Implicit Continuous-fluid Eulerian (ICE) and Arbitrary Lagrangian-Eulerian (ALE) to form the ICEd-ALE technique in the framework of the Simplified-ALE (SALE-3D) computer program, for which a general flow diagram and complete FORTRAN listing are included. Sample problems show how to modify the code for a variety of applications. SALE-3D is patterned as closely as possible on the previously reported two-dimensional SALE program.

NUREG/CR-2186: QUANTITATIVE SOFTWARE RELIABILITY ANALYSIS OF COMPUTER CODES RELEVANT TO NUCLEAR SAFETY. MUELLER, C. J. Argonne National Laboratory. August 1982. 99pp. 8209270453. ANL-81-84. 15530:179.

This report presents the results of the first year of ongoing research programs to determine the probability of failure characteristics of computer codes relevant to nuclear safety. An introduction to both qualitative and quantitative aspects of nuclear software is given. A mathematical framework is presented which will enable the a priori prediction of the probability of failure characteristics of a code given the proper specification of its properties. The framework consists of four parts: 1) a classification system for software errors and code failures; 2) probabilistic modeling for selected reliability characteristics; 3) multivariate regression analyses to establish predictive relationships among reliability characteristics and generic code property and development parameters; and 4) the associated information base. Preliminary data of the type needed to support the modeling and the predictions of this program are described. Illustrations of the use of the modeling are given but the results so obtained, as well as all "results" of code failure probabilities presented herein, are based on data which at this point are preliminary, incomplete, and possibly non-representative of codes relevant to nuclear safety.

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-2192 V01 N3: EVALUATION OF ISOTOPE MIGRATION - LAND BURIAL. Water Chemistry At Commercially Operated Low-Level Radioactive Waste Disposal Sites. Quarterly Progress Report, October-December 1981.

PIETRZAK, R. F. ; DAYAL, R. Brookhaven National Laboratory. August 1982. 45pp. 8209270125. BNL-NUREG-51409. 15516:091.

In this report we present field measurements of trench water properties conducted during the October 1981 sampling trip to Maxey Flats. In-line measurements of specific conductance, dissolved oxygen, Eh, pH, sulfide electrode response and temperature are reported. Appreciable changes in the specific conductance, relative to past results, were observed but since water is periodically removed and replenished by rainwater infiltration, fluctuations in the ionic strength of the solution are expected. Dissolved oxygen levels and pH remained relatively constant with respect to previous measurements. The oxidation potential, Eh, continues to evolve towards a more reducing condition. Sulfide concentrations were too low to be quantified. Radionuclide sorption isotherms were evaluated for two Barnwell trench waters and sediment from the Hawthorne formation. Laboratory experiments to study the chemical changes and the coprecipitation of radionuclides by ferric hydroxide formed during the oxidation of Maxey Flats and West Valley trench waters were made.

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-2193 V01 N3: PROPERTIES OF RADIOACTIVE WASTES AND WASTE CONTAINERS. Quarterly Progress Report, October-December 1981. BECKER, W. W. ; HAYDE, P. R. ; HOPE, M. P. ; et al. Brookhaven National Laboratory. August 1982. 88pp. 8209210476. BNL-NUREG-51410. 14949:228.

A study correlating the leachability of (137)Cs from small-scale to large-scale cement forms was performed. The waste forms consisted of organic ion exchange resins incorporated in Portland I cement, with a waste-to-cement ratio of 0.6 and a water-to-cement ratio of 0.4 (as free water) and boric acid waste (12% solution), incorporated in Portland III cement, with a waste-to-cement ratio of 0.7. (137)Cs was added to both waste types prior to solidification. The samples' dimensions varied from 1 in. x 1 in. to 22 in. x 22 in. (diameter x height) in size. Leach data extending over a period of 260 days were obtained. A method based on semi-infinite plane source diffusion model was applied to analyze the leach data. An effective bulk diffusion coefficient was calculated from the leach data for both types of solidified waste. A derived mathematical expression allows prediction of the amount of (137)Cs leached from the forms as a function of leaching time and waste form dimensions. A reasonably good agreement between the experimental and calculated data is obtained.

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-2199: THE EVALUATION OF MATHEW/ADPIC AS A REAL-TIME DISPERSION MODEL. LEWELLEN,W.S.; SYKES,R.I.; OLIVER,D. Aeronautical Research Associates of Princeton. March 1982. 143pp. 8204070121. ARAP REPT 442. 12592:214.

The key computer models used to describe the wind field and the subsequent atmospheric dispersion in the ARAC (Atmospheric Release Advisory Capability) System are evaluated to determine their potential effectiveness. An evaluation approach is developed which consists of a review of the scientific and mathematical foundations of the models, an analysis of the sensitivity of the models to different inputs, and a review of previous verification tests. The accuracy of the model is measured in terms of a concentration pattern test which is specifically developed in this report to provide a quantitative measure of how well a predicted spatial distribution of ground-level, time-integrated concentration agrees with real field observations. The principal conclusion is that the available evidence does not convince us that results using MATHEW/ADPIC will provide a high level of confidence for decisions regarding emergency dispersal in the atmosphere. Justification for the extra cost and slower response of MATHEW/ADPIC, vis-a-vis a simple Gaussian puff model such as MESODIF, is quite questionable. Further tests are required to firmly establish any real advantages.

NUREG/CR-2200: RADIATION EXPOSURE OF TRANSPORTATION WORKERS HANDLING LARGE QUANTITIES OF RADIOACTIVE PACKAGES. SMITH, B. P.; DAPPE, W. J.; LANTZ, M. W.; et al. Reynolds Electrical & Engineering Co., Inc. January 1982. 33pp. 8201290028. 11750:091.

The study was designed to investigate radiation exposures to transportation workers that handle large quantities of radioactive packages. One of the objectives was to determine the correlation between the doses received and the total Transportation Index (TI) for both cargo handlers and drivers. The data and observations suggest that for facilities that handle several hundred TI per week, there is a high probability that some of their workers will receive radiation exposures exceeding the recommendations for a nonradiation worker.

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. 13794:269.

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 V03: ADVANCED TWO-PHASE FLOW INSTRUMENTATION PROGRAM. Quarterly Progress Rept. July-September 1981. HARDY, J. E.; MILLER, G. N.; ROGERS, S. C.; et al. Oak Ridge National Laboratory. January 1982. 17pp. 8202050244. ORNL/TM-8162. 11831:154.

The performance of the Westinghouse Reactor Vessel Level Indicating System (RVLIS) in the S-UT-3 test (a communicative break in the cold leg of Semiscale) was analyzed. The Westinghouse RVLIS gave similar indications to Semiscale Test Facility instrumentation measuring the same phenomena [differential pressure (dP)] over equal spans. The Westinghouse measurement is apparently conservative when compared with the two-phase froth level. These dP measurements appear to be nonconservative estimates of level, however, when the measurement system spans the upper core support plate. Level measurement errors of up to 150 cm (60 in.) were observed during S-UT-3. Westinghouse claims that these differences are caused by differences between Semiscale and Westinghouse Reactors. A recommendation for resolving these differences is made.

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-2212: AN EVALUATION OF GROUND PENETRATING RADAR FOR ASSESSMENT OF LOW LEVEL NUCLEAR WASTE DISPOSAL SITES. HORTON, K. A.; MOREY, R. M.; BEERS, R. H.; et al. Geo-Centers, Inc. February 1982. 130pp. 8203300306. GC-TR-81-171. 12459:319.

Ground penetrating radar (GPR) has been used to remotely and non-destructively survey a number of low-level waste disposal sites of interest to the Nuclear Regulatory Commission (NRC). The measurements have been used to provide information to the NRC regarding the feasibility of measuring the underlying structure of the earth and targets located within the earth with GPR. In support of the survey program, work has also been carried out in several allied areas including the development of a combination of models of the radar system, geologic structure and composition, and targets that allow predictive calculations to be carried out; the application of pattern recognition and data processing techniques to the acquired data to enhance, distinguish, and analyze features detected by GPR; and the correlation of GPR data with data acquired by other geophysical techniques such as electrical resistivity.

NUREG/CR-2216: INSTITUTIONAL ISSUES ASSOCIATED WITH SPENT FUEL STORAGE UNDER DOMESTIC AND INTERNATIONAL AUSPICES. O'BRIEN, J. N. Brookhaven National Laboratory. January 1982. 63pp. 8201130452. BNL-NUREG-51413. 11577:003.

This report examines the various institutional issues associated with the establishment of away-from-reactor (AFR) spent fuel storage. Technical factors contributing to the problem of spent fuel congestion are briefly reviewed and differing projections of capacity shortfalls are discussed.

Chapter II analyzes the institutional considerations pertinent to the establishment of a domestically constructed and operated AFR facility. This chapter discusses facility characteristics, federal agency jurisdictions, the state/federal interface, and financial arrangements.

Chapter III discusses the problems associated with establishment of an AFR storage facility under international auspices. This chapter discusses international auspices, appropriate compensation for energy content, and sovereign immunity. A comprehensive description of the organization and structure of international organizations is presented. Recommendations are made for the organization and structure of an international AFR.

NUREG/CR-2217: DETECTION OF SPECIAL NUCLEAR MATERIALS AT PORTAL MONITORS AND LOCATION AND RECOVERY OF CONTRABAND SPECIAL NUCLEAR MATERIALS:LEGAL AND TECHNICAL PROBLEMS. O'BRIEN,J.N. Brookhaven National Laboratory. January 1982. 52pp. 8201130287. BNL-NUREG-51414. 11570:036.

This report examines the issues of how reliably special nuclear materials (SNM) can be detected during attempts to steal it and how recovery techniques initiated because of a confirmed theft may affect civil liberties. Chapter II addresses the technical abilities and limitations of detecting SNM under both controlled and uncontrolled conditions. The concepts on "spiking" and shielding are examined. Chapter III discusses the legal requirements and technical limits on detecting small quantities of SNM during smuggling attempts. Assessments are made concerning the type of detectors most desirable and which forms of SNM could logically be spiked to enhance their detectability. Administration and legal restrictions on portal searches and emergency site responses to SNM losses are comprehensively examined. Chapter IV examines the activity of searching for and recovering contraband SNM. Methods for searching, sources of difficulty, and estimates of sensitivity are made. (All data are unclassified.) The legal implications of area and perimeter searches are examined with particular regard to problems of search and seizure law.

NUREG/CR-2220 V01: IMPACT OF ENTRAINMENT AND IMPINGEMENT ON FISH POPULATIONS IN THE HUDSON RIVER ESTUARY.Entrainment Impact Estimates For Six Fish Populations Inhabiting The Hudson River Estuary. BOREMAN,J.; BARNTHOUSE,L.W.; VAUGHAN,D.S.; et al. Commerce,Dept.of, National Marine Fisheries Service. January 1982. 300pp. 8202040124. ORNL/NUREG/TM-3. 11816:358.

This volume is concerned with the estimation of the direct (or annual) entrainment impact of power plants on populations of striped bass, white perch, Alosa spp. (blueback herring and alewife), American shad, Atlantic tomcod, and bay anchovy in the Hudson River estuary. Entrainment impact results from the killing of fish eggs, larvae, and young juveniles that are contained in the cooling water cycled through a power plant. An Empirical Transport Model (ETM) is presented as the means of estimating a conditional entrainment mortality rate (defined as the fraction of a year class which could be killed due to entrainment in the absence of any other source of mortality). The estimation of several parameters required by the ETM are presented: physical input parameters (e.g., power-plant withdrawal flow rates); the longitudinal distribution of ichthyoplankton in time and space; the duration of susceptibility of the vulnerable organisms; the W-factors, which express the ratio of densities of organisms in power plant intakes to densities of organisms in the river; the entrainment mortality factors (f-factors), which express the probability that an organism will be killed if it is entrained.

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 V03: HIGH TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH. Quarterly Progress Report. July 1 - September 30, 1981. BALL, S. J.; CLEVELAND, J. C.; CONKLIN, J. C.; et al. Oak Ridge National Laboratory. January 1982. 17pp. 8202050242. ORNL/TM-8128. 11831:128.

Development work continued on the accident dynamics simulation codes ORTAP, BLAST, and ORECA for the Fort St. Vrain (FSV) reactor.

New steam line and main steam bypass system models were developed and incorporated into ORTAP. An initial simulation of the FSV prestressed concrete reactor vessel and liner cooling system was developed and tested for use in the severe accident sequence analysis task.

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-2222: THEORY AND PRACTICE OF GENERAL ADJUSTMENT AND MODEL FITTING PROCEDURES. STALLMAN, F. W. Oak Ridge National Laboratory. January 1982. 36pp. 8201270102. ORNL/TM-7896. 11703:154.

This paper presents a general method for nonlinear model fitting and adjustment. It is based on the least squares principle and allows the simultaneous determination of model parameters and the adjustment of input data in order to obtain a set of adjusted data and parameters which are consistent with the model equations. The derivation of the mathematical formulas for the least squares estimates and their variances uses the Lagrange multiplier method. Several applications from the area of neutron dosimetry and testing of irradiated materials are discussed.

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-2229 V02: BWR LARGE BREAK SIMULATION TESTS-BWR BLOWDOWN/EMERGENCY CORE COOLING PROGRAM. Volume 2-Appendices I-N. LEE, L. S.; SOZZI, G. L.; ALLISON, S. A. General Electric Co. July 1982. 552pp. 8207290410. EPRI NP-1783. 14126:001.

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-size electrically heated bundle. Fluid delivery systems were included to simulate coolant injections.

Tests conducted under this program include large break (design basis accidents), 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 report covers the large break results. Volume 1 contains the summary, discussion, analysis and conclusions. Volume 2 contains the data reports.

NUREG/CR-2230: BWR SMALL BREAK SIMULATION TESTS WITH AND WITHOUT DEGRADED ECC SYSTEMS. BWR BLOWDOWN/EMERGENCY CORE COOLING PROGRAM. HWANG, W. S. General Electric Co. January 1982. 100pp. 8201290017. EPRI NP-1782. 11749:013.

Two simulation tests of the boiling water reactor (BWR) small break loss-of-coolant accidents (LOCA's) were conducted in the two loop test apparatus (TLTA). The first test investigated the small break with nondegraded emergency core coolant (ECC) systems and the second test studied the same small break but with degraded ECC systems in which the high pressure core spray (HPCS) was assumed unavailalble.

This report discusses the test simulation, apparatus, instrumentation, measurements, and test results of these two tests. These tests provide a basis for evaluating the models and assumptions used in evaluating the BWR response. The results indicate that the phenomena that govern the system response for these small break simulations were similar to the phenomena governing the system response for large breaks. The importance of CCFL at the bundle (core region) inlet in holding up inventory in the core region and aiding in reflood was demonstrated.

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-2233: NEUTRON DOSIMETER PERFORMANCE AND ASSOCIATED CALIBRATIONS AT NUCLEAR POWER PLANTS. SCHWARTZ, R. B. Commerce, Dept. of, National Bureau of Standards. ENDRES, G. W. R. ; CUMMINGS, F. M. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1982. 60pp. 8207290409. 14125:252.

This report addresses problems associated with the calibration and use of personnel neutron dosimeters and monitoring instruments. Four particular items are addressed: 1) The threshold response of NTA film. NTA film is not recommended for use at reactors. 2) A discussion of dosimeter and remmeter calibrations performed using a D(2)O-moderated (252)Cf source. Use of the moderated Cf source is recommended for calibrating dosimeters and instruments used at reactors. 3) The edge effect created by placing the neutron-sensitive elements of albedo dosimeters close to the phantom edge. It is recommended that dosimeters be no closer than 7 cm effective distance from the edge of the phantom on which they are irradiated. 4) The response of various personnel neutron dosimeters inside the containment at nuclear power plants. It is recommended that dosimeters which demonstrate adequate sensitivity be used and be corrected for variations due to neutron energy spectral differences. Dosimeters that were found to be adequate were TLD-albedo dosimeters and polycarbonate track-etch dosimeters which utilized (n,alpha) radiators. NTA films, CR-9 films and polycarbonate films which did not use radiators are inadequate for personnel neutron dosimetry at nuclear power plants.

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-2238 V18 N2: ADVANCED REACTOR SAFETY RESEARCH PROGRAM QUARTERLY REPORT. April-June 1981. * Sandia Laboratories. September 1982. 259pp. 8209270471. SAND81-1529. 15530:295.

A serious effort to characterize the aerosol source term associated with melt/concrete interactions is underway. A critical question in any such investigation is how well must the aerosol source term be known. Clearly, uncertainties must be resolved well enough so that they no longer affect calculations of aerosol behavior in reactor containments or the consequences of aerosol release. An effort to define how well the aerosol source term must be known is now being pursued by examining the effects of source term description on the predicted behavior of aerosols. Early indications are that the aerosol will have to be characterized in terms of mass-weighted size distribution, number-weighted size distribution, shape factor and concentration in addition to chemical composition.

NUREG/CR-2239: TECHNICAL GUIDANCE FOR SITING CRITERIA DEVELOPMENT. ALDRICH, D. C. ; SPRUNG, J. L. ; ALPERT, D. J. ; et al. Sandia Laboratories. December 1982. 462pp. 8301120023. SAND81-1549. 16782:001.

Technical guidance to support the formulation and comparison of possible siting criteria for nuclear power plants has been developed for the Nuclear Regulatory Commission by Sandia National Laboratories. Information has been developed in four areas: (1) consequences of hypothetical severe nuclear power plant accidents, (2) characteristics of population distributions about current reactor sites, (3) site availability within the continental United States, and (4) socioeconomic impacts of reactor siting.

The impact on consequences of source term magnitude, meteorology, population distribution and emergency response have been analyzed. Population distributions about current sites were analyzed to identify statistical characteristics, time trends, and regional differences. A site availability data bank was constructed for the continental United States. The data bank contains information about population densities, seismicity, topography, water availability, and land use restrictions. Finally, the socioeconomic impacts of rural industrialization projects, energy boomtowns, and nuclear power plants were examined to determine their nature, magnitude, and dependence on site demography and remoteness.

NUREG/CR-2241: TECHNOLOGY AND COST OF TERMINATION SURVEYS ASSOCIATED WITH DECOMMISSIONING OF NUCLEAR FACILITIES. WITHERSPOON, J. Oak Ridge National Laboratory. February 1982. 198pp. 8203030212. ORNL/HASRD-121. 12110:001.

There are different types of nuclear facilities each having different residual radionuclides, different exposure pathways, and differing biological doses. Various methods of establishing residual levels of radioactivity have been suggested. All of these methods of criteria development have a common endpoint in that some radiation dose (and risk) can be estimated for individuals, or segments of a population associated with a site. The objective of this study is to provide information on the methods and costs of conducting terminal radiological surveys to verify compliance with dose standards for unrestricted use of nuclear sites following decommissioning.

NUREG/CR-2250: MODELING POWER PLANT IMPACTS ON MULTIPOPULATION SYSTEMS: APPLICATION OF LOOP ANALYSIS TO THE HUDSON RIVER WHITE PERCH POPULATION. BARNTHOUSE, L. W. Oak Ridge National Laboratory. January 1982. 55pp. 8201270099. ORNL/TM-7900. 11703:090.

The white perch population of the Hudson River suffers unusually high mortality due to impingement and entrainment at Hudson River power plants. Previous assessments of the magnitude and biological significance of this mortality have treated the white perch as an isolated entity. In this study an attempt was made to interpret power plant mortality from a community perspective. Simple food chain and food web models, together with information on the life histories and vulnerabilities to power plants of fish and macroinvertebrate populations in the Hudson River, were used to assess (a) effects of interactions with other populations on the response of the white perch population to power plant mortality, and (b) indirect effects on other populations of mortality imposed on white perch. The simplest model studied was a three-compartment food chain consisting of white perch, its invertebrate prey, and the organic detritus pool.

NUREG/CR-2255: EXPERT ESTIMATION OF HUMAN ERROR PROBABILITIES IN NUCLEAR POWER PLANT OPERATIONS: REVIEW OF PROBABILITY ASSESSMENT AND SCALING. STILLWELL, W. G.; SEAVER, D. A.; SCHWARTZ, J. P. Sandia Laboratories. August 1982. 76pp. 8209230021. SAND81-7140. 14977:152.

This report reviews probability assessment and psychological scaling techniques that could be used to estimate human error probabilities (HEPs) in nuclear power plant operations. The techniques rely on expert opinion and can be used to estimate HEPs where data do not exist or are inadequate. These techniques have been used in various other contexts and have been shown to produce reasonably accurate probabilities. Some problems do exist, and limitations are discussed. Additional topics covered include methods for combining estimates from multiple experts, the effects of training on probability estimates, and some idea on structuring the relationship between performance shaping factors and HEPs. Preliminary recommendations are provided along with cautions regarding the costs of implementing the recommendations. Additional research is required before definitive recommendations can be made.

NUREG/CR-2257: INSTABILITY TESTING OF COMPACT AND PIPE SPECIMENS UTILIZING A TEST SYSTEM MADE COMPLIANT BY COMPUTER CONTROL. JOYCE, J. A. U.S. Naval Academy. March 1982. 54pp. 8204070083. 12595:017.

The aim of this paper is to demonstrate that a computer controlled test machine can replace a test machine made compliant by a mechanical spring for tearing instability testing of simple compact and cracked pipe geometries. For both geometries tested herein close agreement was demonstrated between the "computer compliant" and "spring compliant" test systems. The results show that though the computerized system utilized here is slower than the spring machine, this is not a serious drawback for structural materials with low to moderate rate dependence. The "inertia free" response of the computerized system is in fact a positive feature for studying tearing instability arrest and promises to be very useful in further studies in that area.

NUREG/CR-2264 CHARACTERIZATION METHODS FOR ULTRASONIC TEST SYSTEMS. BUSSE, L. J. ; BECKER, F. L. ; BOWEY, R. E. ; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1982. 69pp. 8208180232. PNL-4215. 14391:002.

Methods for the characterization of ultrasonic transducers (search units) and instruments are presented. The instrument system is considered as three separate components consisting of a transducer, a receiver-display, and a pulser. The operation of each component is assessed independently. The methods presented were chosen because they provide the greatest amount of information about component operation and were not chosen based upon such conditions as cost, ease of operation, field implementation, etc. The results of evaluating a number of commercially available ultrasonic test instruments are presented.

NUREG/CR-2268: METABOLIC FATE AND EVALUATION OF INJURY IN RATS AND DOGS FOLLOWING EXPOSURE TO THE HYDROLYSIS PRODUCTS OF URANIUM HEXAFLUORIDE, IMPLICATIONS FOR A BIOASSY PROGRAM RELATED TO POTENTIAL RELEASES OF URANIUM HEXAFLUORIDE. MORROW, P. ; LEACH, L. ; SMITH, F. ; et al. Rochester, Univ. of,. December 1982. 172pp. 8301190426. 16859:263.

This final report summarizes the experimental studies undertaken in rats and dogs in order to ascertain whether or not the NRC bioassay program has adequate biological bases for quantifying and evaluating uranium hexafluoride exposures. Animals were administered the spontaneous hydrolysis products of uranium hexafluoride by inhalation exposures, intratracheal instillations and parenteral injections. Attention was given dose-effect relationships appropriate to the kidney, the unique site of subacute toxicity; to the rates of uranium excretion; and to uranium retention in renal tissue. These criteria were examined in both naive and multiply-exposed animals. Findings of the studies: partly substantiate the ICRP excretion model for hexavalent uranium; generally provide a lower renal injury threshold concentration than implicit in the MPC for natural uranium; indicate distinctions in response based on exposure history, e.g., uranium excretion rate; compare and evaluate various biochemical indices of renal injury; raise uncertainties about prevailing views of "reversible" renal injury, renal "tolerance" and possible hydrogen fluoride synergism with uranium effects; and reveal species differences in several areas, e.g., renal retention of uranium. While the study presents some complicating features to extant bioassay practice, it nevertheless supplies data supportive of the bioassay concept.

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-2281 V04: NUCLEAR REACTOR SAFETY. October 1-December 31, 1981. STEVENSON, M. G. Los Alamos Scientific Laboratory. July 1982. 42pp. 8208120368. LA-9305-PR. 14339:240.

The work that is highlighted here represents accomplishments for the period October 1 - December 31, 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-2282: LARGE-SCALE, TRANSIENT TESTS OF THE INTERACTION OF MOLTEN STEEL WITH CONCRETE. POWERS, D. A.; ARELLANO, F. E. Sandia Laboratories. March 1982. 362pp. 8204150564. SAND81-1753. 12688:001.

A series of large-scale tests of high temperature melt interactions with basaltic, limestone/common sand, and limestone concrete are reported. The tests were conducted to identify qualitatively those phenomena that arise during the interactions that are pertinent to the analysis of core meltdown accidents at nuclear power plants. The tests were done by pouring about 200 kg of mild steel or stainless steel at 1973 K (1700 C) into concrete crucibles. The melts were allowed to cool naturally once they were in the crucibles. Each crucible was exposed to molten steel several times to simulate prolonged periods of interaction. All the concretes behaved in qualitatively similar ways. Gas generation due to the thermal decomposition of hydrates and carbonates in the concrete was the most dramatic feature of the melt interaction with concrete. The steam and carbon dioxide liberated from the concrete vigorously agitated the melt and seemed to provide some resistance to the flow of heat from the melt to the concrete. These gases also reacted with the steel melts to form hydrogen and carbon monoxide. Combustion of the reaction products as well as convective heat losses produced by the gases contributed to the upward heat transfer from the melt.

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-2285: INTERIM TECHNICAL ASSESSMENT OF THE MARCH CODE. RIVARD, J. B. Sandia Laboratories. January 1982. 278pp. 8201270389. SAND81-1672. 11711:163.

The NRC's increasing requirements for capability of LWR meltdown analysis have led to a request for an independent assessment of the MARCH Code. To a large extent, MARCH was found to integrate recent understanding of the many complex phenomenologies, engineered systems, and events into a fast-running code applicable to a wide variety of LWR systems and accident sequences involving core melt. However, uncritical usage of MARCH can lead to seriously erroneous conclusions because numerous limitations in the modeling, coding, and documentation of MARCH do exist. Taken together, these limitations may seriously compromise the validity of some predictions; furthermore, the code limitations and approximations are such that the user cannot a priori assume that conservative results will be obtained in all cases. Hence, it is recommended that NRC initiate a strong effort to improve MARCH capabilities and to standardize its usage. It is also recommended that the assessed version of MARCH be employed in formal safety evaluation only when the results are accompanied by explicit evaluation and discussion of code limitations with regard to the potential of the limitations for invalidating conclusions drawn from code results, and that calculations be made using a documented version of the code with data input appropriate to the plant and accident sequence with a fully accessible set of values for the modeling parameters and options used.

NUREG/CR-2288: COMPUTER SURETY. Computer System Inspection Guidance. * Lawrence Livermore Laboratory. * Teknekron Research, Inc. March 1982. 163pp. 8204020059. UCID-18975. 12492:328.

This document discusses computer surety in NRC licensed nuclear facilities from the perspective of physical protection inspectors. It gives background information and a glossary of computer terms, along with threats and computer vulnerabilities, methods used to harden computer elements, and computer audit controls.

NUREG/CR-2296: A FIRST STUDY OF AEROSOLS PRODUCED BY NEUTRONIC HEATING OF FRESH URANIUM DIOXIDE FUEL UNDER CORE-DISRUPTIVE ACCIDENT CONDITIONS. ELRICK, R. M. Sandia Laboratories. November 1982. 84pp. 8212130283. SAND81-1788. 16418:127.

Using close-in, time-resolved sampling of fuel debris, high-speed photography, and photometric determination of fuel temperatures, the kinetic thermal states of the disrupting UO (2) fuel were defined as it was neutronically heated to vapor by a 5 ms full width at half-maximum pulse of 2380 kj/kg. From these data it was possible to obtain a fuel energy-temperature relation up to the melt temperature and debris kinetics in the form of particle distributions in velocity and size for particles from about 10 um to < 3 nm in size. Several calculations and comparisons were made to establish a fairly high degree of reliability for the sampling and analysis techniques.

NUREG/CR-2297: SECURITY MANAGEMENT TECHNIQUES AND EVALUATIVE CHECKLISTS FOR SECURITY FORCE EFFECTIVENESS. SCHURMAN, D. L.; DATESMAN, G. H.; TRUITT, J. O. 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 V03: LMFBR AEROSOL RELEASE AND TRANSPORT PROGRAM. Quarterly Progress Report for July-September 1981. KRESS, T. S.; TOBIAS, M. L. Oak Ridge National Laboratory. February 1982. 54pp. 8205060131. ORNL/TM-8149. 13003:296.

This report summarizes progress for the Aerosol Release and Transport Program sponsored by the Office of Nuclear Regulatory Research, Division of Accident Evaluation of the Nuclear Regulatory Commission for the period July-September 1981. Topics discussed include (1) preparations for under-sodium tests at the Fast Aerosol Simulant Test Facility, (2) progress in interpretation of Oak Ridge National Laboratory-Sandia Laboratory normalization test results, (3) U(3)O(3) in steam (light-water reactor accident) aerosol experiments conducted in the Nuclear Safety Power Plant, (4) experiments on B(2)O(3) and SiO(2) aerosols at the Containment Research Installation-II Facility, (5) fuel-melting tests in small-scale experimental facilities for the core-melt aerosol program, (6) analytical comparison of simple adiabatic nonlinear and linear analytical models of bubble oscillation phenomena with experimental data.

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(3)O(3) and Fe(2)O(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-2305 V02: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Quarterly Progress Report For Period Ending June 30, 1981. DODD, C. V.; DEEDS, W. E.; MCCLUNG, R. W. Oak Ridge National Laboratory. February 1982. 10pp. 8203190446. ORNL/TM-8161. 12371:083.

Eddy-current methods provide the best in-service inspection of steam generator tubing, but they can produce ambiguity because of the many independent variables that affect the signal. The current development program has used mathematical models and has developed or modified computer programs to design optimum probes, instrumentation, and techniques for multifrequency, multiproperty examinations. Interactive calculations and experimental measurements have used modular eddy-current instrumentation and a minicomputer. These establish the coefficients for the complex equations that define the values of the desired properties (and the attainable accuracy) despite changes in other significant variables. The computer programs for calculating the accuracy with which various properties can be measured indicate that the tubing wall thickness and the defect size can be measured much more accurately than is currently required, even when other properties vary. Our experimental measurements have confirmed these results. We have obtained excellent data at scheduled field inspections of the Ginna and Point Beach steam generator. A field test was recently performed at Babcock and Wilcox, Lynchburg, Virginia, on tubing containing intergranular attack. The tubing was extracted from a Ginna steam generator for metallographic examination. We are continuing to improve the equipment and the data-processing systems for field application.

NUREG/CR-2305 V03: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Quarterly Progress Report For Period Ending September 30, 1981. DODD, C. V.; DEEDS, W. E.; MCCLUNG, R. W. Oak Ridge National Laboratory. July 1982. 7pp. 8207260003. ORNL/TM-8296. 14069:051.

The current development program has used mathematical models and has developed or modified computer programs to design optimum probes, instrumentation, and techniques for multifrequency, multiproperty examinations. Interactive calculations and experimental measurements have been made with the use of modular eddy-current instrumentation and a minicomputer. These establish the coefficients for the complex equations that define the values of the desired properties (and the attainable accuracy) despite changes in other significant variables. The computer programs for calculating the accuracy with which various properties can be measured indicate that the tubing wall thickness and the defect size can be measured more accurately than is currently required, even when other properties vary. Our experimental measurements have confirmed these results. We have analyzed data taken during scheduled field inspections of the Ginna and Point Beach steam generators and have obtained excellent results. We are also performing tests with pancake probe coils to improve detection of circumferential flaws. A field test was recently performed at Babcock and Wilcox, Lynchburg, Virginia, on tubing containing intergranular attack. The tubing was extracted from a Ginna steam generator for metallographic examination. We are continuing to improve the equipment and the data-processing systems for field application.

NUREG/CR-2305 V04: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Annual Progress Report For Period Ending December 31, 1981. DODD, C. V. ; DEEDS, W. E. ; MCCLUNG, R. W. Oak Ridge National Laboratory. August 1982. 23pp. 8209270131. ORNL/TM-8352. 15518:239.

Eddy-current methods provide the best in-service inspection of steam generator tubing, but present techniques can produce ambiguity because of the many independent variables that affect the signals. The current development program has used mathematical models and developed or modified computer programs to design optimum probes, instrumentation, and techniques for multifrequency, multiproperty examinations. To facilitate the extensive laboratory scanning of specimens that are necessary to calibrate the instrumentation for all the possible combinations of positions of flaws, tube supports, and probe coils, we have designed, constructed, and used a computer-controlled automatic positioner. An advanced microcomputer has been designed, constructed, and installed in the instrumentation to control the examination and provide real-time calculations of the desired properties for display and recording during the scanning of the tube.

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. 89pp. 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-2308: DESIGN CRITERIA FOR THE SPACING OF NOZZLES AND REINFORCED OPENINGS IN CYLINDRICAL NUCLEAR PRESSURE VESSELS AND PIPE. MOORE, S. E. ; MERSHON, J. L. Oak Ridge National Laboratory. March 1982. 31pp. 8204150572. ORNL-5809. 12687:155.

The minimum separation distance between any two nozzle penetrations in a nuclear pressure vessel or piping system is governed by design rules given in Section III of the ASME Boiler and Pressure Vessel Code. A broad-based generic study of these rules was conducted to assess their validity and/or develop more appropriate rules as needed. During the course of the study, we determined that the various rules given in the Code are not always consistent with its basic stress criteria and may be either unduly conservative or nonconservative depending on the dimensional details of a given design. A new separation rule was therefore developed and is presented in the form of proposed changes to the ASME Code. The new rule is based on a comprehensive collection of finite-element stress-analysis studies and experimental data that span the range of dimensional parameters permitted for the design of nozzles in cylindrical vessels. These include data for both isolated and closely spaced nozzles. The proposed rule was also examined by comparison with a detailed finite-element analysis study for the placement of an emergency core-cooling system nozzle in a prototypical four-loop pressurized-water-reactor pressure vessel. The proposed rule would permit placing the nozzle in the vessel approximately at its optimum design location, whereas the current rules could not be used without performance of a detailed stress analysis. The comparison study also showed that the proposed rule is conservative, as it should be, but less conservative than the current rules.

NUREG/CR-2308 ERR: DESIGN CRITERIA FOR THE SPACING OF NOZZLES AND REINFORCED OPENINGS IN CYLINDRICAL NUCLEAR PRESSURE VESSELS AND PIPE. * Oak Ridge National Laboratory. March 8, 1982. 2pp. 8204230022. ORNL-5809. 12834:331.

The minimum separation distance between any two nozzle penetrations in a nuclear pressure vessel or piping system is governed by design rules given in Section III of the ASME Boiler and Pressure Vessel Code. A broad-based generic study of these rules was conducted to assess their validity and/or develop more appropriate rules as needed. During the course of the study, we determined that the various rules given in the Code are not always consistent with its basic stress criteria and may be either unduly conservative or nonconservative depending on the dimensional details of a given design. A new separation rule was therefore developed and is presented in the form of proposed changes to the ASME Code. The new rule is based on a comprehensive collection of finite-element stress-analysis studies and experimental data that span the range of dimensional parameters permitted for the design of nozzles in cylindrical vessels. These include data for both isolated and closely spaced nozzles. The proposed rule was also examined by comparison with a detailed finite-element analysis study for the placement of an emergency core-cooling system nozzle in a prototypical four-loop pressurized-water-reactor pressure vessel. The proposed rule would permit placing the nozzle in the vessel approximately at its optimum design location, whereas the current rules could not be used without performance of a detailed stress analysis. The comparison study also showed that the proposed rule is conservative, as it should be, but less conservative than the current rules.

NUREG/CR-2311: THE IMPACT OF IMPINGEMENT ON THE HUDSON RIVER WHITE PERCH POPULATION:Final Report. BARNTHOUSE,L.W.; KIRK,B.L.; VAN WINKLE,W.; et al. Oak Ridge National Laboratory. March 1982. 60pp. 8204070074. ORNL/TM-7975. 12594:320.

This report summarizes a series of analyses of the magnitude and biological significance of the impingement of white perch at the Indian Point Nuclear Generating Station and other Hudson River power plants. Included in these analyses were evaluations of (a) two independent lines of evidence relating to the magnitude of impingement impacts on the Hudson River white perch population, (b) the additional impact caused by entrainment of white perch, (c) data relating to density-dependent growth among young-of-the-year white perch, (d) the feasibility of performing population-level analyses of impingement impacts on the white perch populations of Chesapeake Bay and the Delaware River, and (e) the feasibility of using simple food chain and food web models to evaluate community-level effects of impingement and entrainment.

NUREG/CR-2313 ERR: BOUNDARY-FITTED COORDINATE TRANSFORMATION FOR THERMOHYDRAULIC ANALYSIS IN ARBITRARY THREE-DIMENSIONAL GEOMETRIES. SHA, W. T. Argonne National Laboratory. January 7, 1982. 2pp. 8203040134. ANL-81-54. 12119:315.

A method using boundary-fitted coordinate transformation applicable to an arbitrary three-dimensional geometry is developed. A set of corresponding transformed compressible, single-phase three-dimensional, time-dependent Navier-Stokes and energy equations is derived and presented.

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-elongation properties and on material agiing, regardless of the aging environment. The data obtained, however, may be too limited to show significant dose-rate effects.

NUREG/CR-2315: THE EFFECT OF PRETREATMENT ON THE INITIAL REACTION RATES OF PGX AND H451 GRAPHITES WITH H(2)O AND O(2). ETO, M.; GROWCOCK, F. B. Brookhaven National Laboratory. January 1982. 18pp. 8201130363. BNL-NUREG-51447. 11569:331.

Initial reaction rates were examined for PGX and H451 graphites exposed to different atmospheres prior to oxidation in 2% H(2)O/He at room temperature caused high initial reaction rates of PGX and H451 graphites oxidized in 2% H(2)O/He at 750 degrees C or 800 degrees C, which may be attributed to oxidant adsorption prior to oxidation. High initial reaction rates were not observed when the specimens were exposed to He without H(2)O before oxidation. Exposure of the graphites to inert gas at 750 or 800 degrees C resulted in very high initial reaction rates, especially in the case of PGX graphite; this phenomenon may be attributed to catalysis by metallic impurities which are activated upon reduction by the graphite matrix.

NUREG/CR-2316: EFFECT OF PRESTRESS AND STRESS ON THE STRENGTH AND OXIDATION RATE OF NUCLEAR GRAPHITE. ETO, M.; GROWCOCK, F. B. Brookhaven National Laboratory. January 1982. 22pp. 8201270395. BNL-NUREG-51448. 11714:044.

Effects of prestress or stress during oxidation on the reactivities and strengths of PGX and H451 graphites were examined at oxidation temperatures of 450-600 degrees C in air and 550-750 degrees C in 2% H(2)O/He. Little or no effect of compressive prestress at stress levels of up to 75% of the mean fracture strengths was found. A small effect on the reactivity of H451 graphite was observed in the case of 2% H(2)O/He, i.e. the rate was enhanced by no more than 35% for compressive prestress levels of up to 90% of the fracture strength.

Tensile and compressive stresses during oxidation did not affect the reactivities of these graphites at stress levels of up to 28% of the mean compressive strength and 35% of the mean tensile strength for H451 graphite, and 44% and 56% for PGX graphite, respectively.

NUREG/CR-2317 V01 N1: CONTAINER ASSESSMENT-CORROSION STUDY OF HLW CONTAINER MATERIALS. Quarterly Progress Report, April-June 1981. AHN, T. M.; LEE, B. S.; SOO, P. Brookhaven National Laboratory. March 1982. 16pp. 8204150203. BNL-NUREG-51449. 12687:293.

A research program on container assessment has been initiated to determine the general and localized corrosion mechanisms in high-level waste container materials such as titanium, cooper, and lead. In this quarter, our efforts have been on the initiation of a program establishing specific experiments pertaining to uniform corrosion, crevice corrosion, pitting corrosion, stress corrosion, hydrogen embrittlement, radiation effects, vapor-solution-container wall interaction, and an analytical study. Preliminary tests have been performed for the uniform and crevice corrosion of copper and titanium in brine at 150 degrees C.

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 irradiation of brine and the gases generated were analyzed.

NUREG/CR-2317 V02 N1: CONTAINER ASSESSMENT-CORROSION STUDY OF HLW CONTAINER MATERIALS. January-March 1982. AHN,T.M.; SOO,P. Brookhaven National Laboratory. October 1982. 28pp. 8211110647. BNL-NUREG-51449. 16050:152.

The current program was initiated to determine potential corrosion failure modes in TiCode-12 high level waste container material exposed to prototypic repository conditions. A basic goal was to elucidate the mechanism by which failure occurs and to develop means of extrapolating short term test data to the prediction of very long term container behavior. Work to date in this program has shown that crevice corrosion is a possible failure mode in TiCode-12. This is the first observation that has been made of this type of corrosion in a simulated repository brine environment. Thus, a major part of the BNL effort is directed to finding conditions under which it occurs, determining the precise mechanism of failure, and to evaluating whether the occurrence of long term crevice attack will compromise container integrity. Efforts in this quarter have been concentrated on the crevice corrosion and hydrogen embrittlement of TiCode-12 and CP titanium. Electrochemical work in acidified brines has also been performed to aid in understanding the corrosion of these materials in the crevice environment.

NUREG/CR-2324: USER'S MANUAL FOR THE SANDIA WASTE-ISOLATION FLOW AND TRANSPORT MODEL (SWIFT) RELEASE 4.81. REEVES,M.; CRANWELL,R.M. Sandia Laboratories. January 1982. 154pp. 8201280132. SAND81-2516. 11730:046.

This report describes a three-dimensional finite-difference model (SWIFT) which is used to simulate flow and transport processes in geologic media. The model was developed for use by the Nuclear Regulatory Commission in the analysis of deep geologic nuclear waste-disposal facilities. This document, as indicated by the title, is a user's manual and is intended to facilitate the use of the SWIFT simulator. Mathematical equations, submodels, application notes, and a description of the program itself are given herein. In addition, a complete input data guide is given along with several appendices which are helpful in setting up a data-input deck.

NUREG/CR-2331 V01 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH. Quarterly Progress Report, April 1-June 30, 1981. CERBONE, R. J.; DIAMOND, D. J.; GINSBERG, T.; et al. Brookhaven National Laboratory. January 1982. 148pp. 8201130405. BNL-NUREG-51454. 11572:257.

This progress report describes current activities and technical progress in the programs at Brookhaven National Laboratory sponsored by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research during April-June 1981. The projects reported this quarter are the following: HTGR Safety Evaluation, SSC Code Development, Thermal Hydraulic LWR and LMFBR Safety Experiments, RAMONA-3B Code Modification and Evaluation, LWR Plant Analyzer Development Program, LWR Code Assessment and Application, Stress Corrosion Cracking of PWR Steam Generator Tubing, and Standards for Materials Integrity in LWR's. The previous reports have covered the periods October 1, 1976 through March 31, 1981.

NUREG/CR-2331 V01 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH. Quarterly Progress Report, July 1-September 30, 1981. CERBONE, R. J.; DIAMOND, D. J.; WEEKS, J. R.; et al. Brookhaven National Laboratory. March 1982. 125pp. 8204070100. BNL-NUREG-51454. 12595:328.

The Advanced and Water Reactor Safety Research Programs Quarterly Progress Report will be combined and hereafter entitled, "Safety Research Programs Sponsored by the Office of Nuclear Regulatory Research Quarterly Progress Report." This progress report will continue to describe current activities and technical progress in the programs at Brookhaven National Laboratory sponsored by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. The projects reported this quarter are the following: HTGR Safety Evaluation, SSC Code Development, Thermal Hydraulic LWR and LMFBR Safety Experiments, LWR Plant Analyzer Development Program, LWR Code Assessment and Application, Stress Corrosion Cracking of PWR Steam Generator Tubing, and Standards for Materials Integrity in LWR's. The previous reports have covered the periods October 1, 1976 through June 30, 1981.

NUREG/CR-2331 V01 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH. Quarterly Progress Report, October 1-December 31, 1981. CERBONE, R. J.; DIAMOND, D. J.; GINSBERG, T.; et al. Brookhaven National Laboratory. July 1982. 158pp. 8207260004. BNL-NUREG-51454. 14068:106.

The Advanced and Water Reactor Safety Research Programs Quarterly Progress Reports will be combined and here after entitled, "Safety Research Programs Sponsored by the Office of Nuclear Regulatory Research, Quarterly Progress Report." This progress report will continue to describe current activities and technical progress in the programs at Brookhaven National Laboratory. The projects reported each quarter are the following: HTGR Safety Evaluation, SSC Code Development, Thermal Hydraulic LWR and LMFBR Safety Experiments, RAMONA-3B Code Modification and Evaluation, LWR Plant Analyzer Development Program, LWR Code Assessment and Application, Stress Corrosion Cracking of PWR Steam Generator Tubing, Standards for Materials Integrity in LWRs, Probability Based Load Combinations for Structural Design, Mechanical Piping Benchwork Problems, and Soil Structure Interaction. The previous reports have covered the period October 1, 1976 through September 30, 1981.

NUREG/CR-2331 V02 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH. January-March 1982. KOUTS, H. J. C. ; KATO, W. J. Brookhaven National Laboratory. October 1982. 189pp. 8211110109. BNL-NUREG-51454. 16047:307.

This progress report will describe current activities in the programs sponsored by the Division of Accident Evaluation, Division of Engineering Technology, and Division of Facility Operations of the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. The projects reported are the following: HTGR Safety Evaluation, SSC Development, Validation and Application, Generic Balance of Plant Modeling, Thermal Hydraulic LWR and LMFBR Safety Experiments, RAMONA-3B Code Modification and Evaluation, LWR Plant Analyzer Development, LWR Code Assessment and Application, Stress Corrosion Cracking of PWR Steam Generator Tubing, Standards for Materials Integrity in LWRs, Probability Based Load Combinations for Structural Design, Mechanical Piping Benchwork Problems, Soil Structure Interaction, Human Error Rate Data Analysis, and Criteria on Human Engineering Regulatory Guides. The previous reports have covered the period October 1, 1976 through December 31, 1981.

NUREG/CR-2332: TIME DEPENDENT UNAVAILABILITY OF A CONTINUOUSLY MONITORED COMPONENT. GINSBERG, T. ; DICKEY, J. M. ; HALL, R. E. Brookhaven National Laboratory. January 1982. 43pp. 8201270408. BNL-NUREG-51457. 11714:067.

The equations which describe the time dependent unavailability of a monitored component are developed and solved. It is assumed that the component failure rate satisfies a Weibull distribution and that each repair requires the same amount of time. A repair may restore the component to a condition similar to its condition just before the failure or alternatively to the same condition as a new component. The solutions for the unavailability are presently used in the FRANTIC II code. The accuracy of approximate solutions has been checked.

NUREG/CR-2333 V01: NUCLEAR WASTE MANAGEMENT TECHNICAL SUPPORT IN THE DEVELOPMENT OF NUCLEAR WASTE FORM CRITERIA FOR THE NRC. Task 1:Waste Package Overview. DEYEL, R. ; LEE, B. S. ; WILKE, R. J. ; et al. Brookhaven National Laboratory. February 1982. 422pp. 8203190009. BNL-NUREG-51458. 12355:001.

This report assesses the current state of waste package development for high-level waste, transuranic waste, and spent fuel in the U.S. and abroad. Recent and on-going research on various waste forms, container materials and backfills are reviewed and those which are likely to perform most satisfactorily in the repository environment are identified. Radiation effects on the waste package components have been reviewed and the magnitude of these effects has been identified. Areas requiring further research have been identified. The important variables affecting radionuclide release from the waste package have been described and an evaluation of regulatory criteria for high-level waste and spent fuel is presented. Finally, for spent fuel, high-level, and TRU waste, an attempt is made to describe and identify components which could be used to construct a waste package having potential to meet NRC performance requirements.

NUREG/CR-2333 V02: NUCLEAR WASTE MANAGEMENT TECHNICAL SUPPORT IN THE DEVELOPMENT OF NUCLEAR WASTE FORM CRITERIA FOR THE NRC. Task 2:Alternative TRU Technologies. BIDA, G. ; MACKENZIE, D. R. Brookhaven National Laboratory. February 1982. 222pp. 8203190004.

BNL-NUREG-51458. 12352:326.

Three main areas of transuranic (TRU) waste management are addressed: immobilization processes and waste forms for ultimate geologic disposal of TRU waste; decontamination as a method for TRU waste management; and potential problems associated with gas generation by certain TRU wastes. Waste forms are considered in terms of the regulations and criteria proposed in 10 CFR 60. Evaluation of the waste forms is based principally on ability to meet the release rate criterion of 10(5)/year given in the Performance Objectives of Section 111, but also on the general requirements of Section 133. The two classes of metallic waste which are candidates for decontamination treatment are Zircaloy cladding hulls from light waste reactor fuel elements, and failed facilities and equipment. Decontamination methods are addressed with regard to their ability to remove contamination to a level below the 10 nCi/g TRU limit. Other important factors are the volume reduction achieved, and compatibility of the secondary waste streams with acceptable waste forms. Gas generation by combustible TRU wastes and cast concretes containing TRU isotopes is discussed, and its potential for damage to a geologic repository is considered. Exclusion of combustible TRU waste from repositories is recommended.

NUREG/CR-2333 V03: NUCLEAR WASTE MANAGEMENT TECHNICAL SUPPORT IN THE DEVELOPMENT OF NUCLEAR WASTE FORM CRITERIA FOR THE NRC. Task 3: Waste Inventory Review. MAJUMDAR, D.; INDUSI, J. D.; MAAKTALA, H. K. Brookhaven National Laboratory. February 1982. 1p. 8203190238. BNL-NUREG-51458. 12513:141.

This inventory of high level nuclear waste in the U.S. was developed from existing data, reports, and other sources of information. Existing and projected inventories of high level waste (HLW), transuranic (TRU) waste, spent fuel, and decommissioning and dismantling (D&D) waste will be required in making prudent decisions in the licensing of high level waste repositories.

Numerous references, reports, and data sources were consulted for this work. Discussions with responsible individuals at the major waste storage facilities were held to help resolve and understand discrepancies that appear in the literature. Low level wastes are excluded from this inventory. For decommissioning and dismantling wastes, no detailed projections for commercial power reactors are available.

The inventory is reported in three parts. Part 1 gives the existing and projected inventory of HLW, spent fuel, and TRU wastes. Part 2 is concerned with the characteristics and volume of high activity D&D waste from commercial power reactors. Part 3 provides additional information of D&D wastes from power reactors and particle accelerators.

NUREG/CR-2333 V04: NUCLEAR WASTE MANAGEMENT TECHNICAL SUPPORT IN THE DEVELOPMENT OF NUCLEAR WASTE FORM CRITERIA FOR THE NRC. Task 4: Test Development Review. AHN, T. M.; CZYSCINSKI, K. S.; FRANZ, E. M.; et al. Brookhaven National Laboratory. February 1982. 195pp. 8203180464. BNL-NUREG-51458. 12340:182.

This report reviews DOE's development of methods for testing high-level nuclear waste packages. These methods are for potential use in comparative tests of alternative materials and components, in tests designed to control the quality of the materials used in the waste package and eventually in tests to demonstrate that the waste package will meet the proposed regulation for disposal of high-level nuclear waste (proposed 10CFR60).

This review includes the activities of the Materials Characterization Organization. This organization is dealing successfully with comparative testing, but, in general, has not yet addressed compliance testing.

Testing and characterization of each component are discussed in separate chapters. Testing of the waste form is considered in terms of chemically induced changes (leaching), mechanical effects and radiation effects. Waste container testing involves corrosion investigations, mechanical testing and studies of radiation effects. Testing of geologic backfill is discussed with the emphasis placed on the ability of the backfill to retain radionuclides under different aqueous flow conditions. Finally, the different objectives for whole package testing are reviewed and compared to recent experiments started in DOE laboratories. Recommendations for further research are made and general types of compliance tests are identified.

NUREG/CR-2333 V05: NUCLEAR WASTE MANAGEMENT TECHNICAL SUPPORT IN THE DEVELOPMENT OF NUCLEAR WASTE FORM CRITERIA FOR THE NRC. Task 5:National Waste Package Program. DAVIS,M.S. Brookhaven National Laboratory. February 1982. 58pp. 8203190002. BNL-NUREG-51458. 12353:236.

This report assesses the need for a centrally organized waste package effort and whether the present program meets those needs. It is the conclusion of the BNL staff that while the DOE has in principle organized a national effort to develop high-integrity waste packages for geologic disposal of high-level waste, the DOE effort has not yet produced data indicating that a waste package will comply with NRC's criteria. The BNL staff feels, however, that such a package is achievable either by development of high integrity components which by themselves could comply with 1,000-year containment or by the development of new waste package designs that could comply with the containment and the controlled release criteria in the 10CFR60 performance objectives.

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-2342: XSDRNPM-S BIASING OF MORSE-SGC/S SHIPPING CASK CALCULATIONS. HOFFMAN, T. J. ; TANG, J. S. Oak Ridge National Laboratory. October 1982. 28pp. 8211150682. ORNL/CSD/TM-175. 16076:318.

This report describes implementation of a systematic approach for biasing a Monte Carlo radiation transport calculation. In particular, the adjoint fluxes from a one-dimensional discrete ordinates calculation with the XSDRNPM-S code are used to generate biasing parameters for the multi-group Monte Carlo code, MORSE-SGC/S. Application of this biasing procedure to several deep penetrating spent fuel shipping cask problems is also reported. The results obtained for neutron and gamma-ray transport indicate that relatively inexpensive Monte Carlo calculations are possible for dry and water filled shipping cask problems using these procedures.

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-2345 V02: LWR PRESSURE VESSEL IRRADIATION SURVEILLANCE DOSIMETRY. Quarterly Progress Report. April 1981 - June 1981. GUTHRIE, G. L. ; MCELROY, W. W. Hanford Engineering Development Laboratory. December 1982. 82pp. 8301190421. HEDL-TME 81-34. 16858:031.

The Light Water Reactor Pressure Vessel (LWR-PV) Surveillance Dosimetry Improvement Program was established by NRC to improve, maintain, and standardize neutron dosimetry, damage correlation, and the associated reactor analysis data and procedures that predict the integrated effect of neutron exposure to LWR-PVs. A vigorous research effort to measure and analyze these areas exists worldwide, with strong cooperative links among NRC-supported activities at HEDL, ORNL, NBS, and NRL and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Julich, Germany), and several UK laboratories. These cooperative links are strengthened by the active membership of the scientific staff from participating countries and laboratories in the ASTM E10 Committee on Nuclear Technology and Applications. Several subcommittees of ASTM E10 are responsible for the preparation of LWR-PV surveillance standards.

NUREG/CR-2345 V04: LWR PRESSURE VESSEL IRRADIATION SURVEILLANCE DOSIMETRY. Quarterly Progress Report, October 1981-December 1981. GUTHRIE, G. L. ; MCELROY, W. N. Hanford Engineering Development Laboratory. October 1982. 86pp. 8211030218. HEDL-TME 81-36. 15930:159.

The Light Water Reactor Pressure Vessel (LWR-PV) Surveillance

Dosimetry Improvement Program was established by NRC to improve, maintain, and standardize neutron dosimetry, damage corrrelation and the associated reactor analysis data and procedures that predict the integrated effect of neutron exposure to LWR-PVs. A vigorous research effort to measure and analyze these areas exists worldwide, with strong cooperative links among NRC-supported activities at HEDL, ORNL, NBS, and NRL and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Julich, Germany), and several UK laboratories. These cooperative links are strengthened by the active membership of the scientific staff from participating countries and laboratories in ASTM E10 Committee on Nuclear Technology and Applications. Several subcommittees of ASTM E10 are responsible for the preparation of LWR-PV surveillance.

NUREG/CR-2347: MARGINS TO FAILURE - CATEGORY-I STRUCTURES PROGRAM: BACKGROUND AND EXPERIMENTAL PROGRAM PLAN. ENDEBROCK, E. ; DOVE, R. ; ANDERSON, C. A. Los Alamos Scientific Laboratory. February 1982. 62pp. 8205060075. LA-9030-MS. 13007:221.

The background and justification for the Margins to Failure--Category-I Structures Program are presented. The results of a survey of selected Architect/Engineering firms engaged in the design of nuclear power plant facilities are included. Computer codes and analytical methods that are used, or may be useful, in assessing the margin to failure of Category-I structures are reviewed. An experimental program plan, whose goal is to obtain information about structural behavior at ultimate load, is proposed. This information is needed to make more reliable assessments of margins to failure. The proposed experimental structures are scaled reinforced concrete shear walls and scale model building systems. This report includes discussions of types of loadings, test procedures, data interpretation, and test plans.

NUREG/CR-2350: SENSITIVITY ANALYSIS TECHNIQUES: SELF-TEACHING CURRICULUM. IMAN, R. L. ; CONOVER, W. D. Sandia Laboratories. June 1982. 161pp. 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-2352: REPOSITORY SITE DEFINITION IN BASALT: PASCO BASIN, WASHINGTON. GUZOWSKI, R. V. ; NIMICK, F. B. ; MULLER, A. B. Sandia Laboratories. March 1982. 210pp. 8204010551. SAND81-2088. 12486:134.

In this report the regional setting, geology, hydrology and geochemistry of a potential deep geologic high-level waste repository site in basalt is discussed. Hydrologic properties and other physical parameters are defined for either each of the stratigraphic units or each general rock type. Detailed documentation and description of the parameters are in the Appendices.

This report documents work performed for the U.S. Nuclear Regulatory Commission (NRC) by Sandia National Laboratories in Albuquerque (SNLA) on a repository site definition for a high-level nuclear waste repository in basalt. Repository site definitions are an integral part of the NRC program at SNLA to develop risk assessment methodologies for the geological disposal of high-level waste. Developing these assessment methods requires an understanding of the natural system to be assessed and a data base with which to exercise and demonstrate the assessment methodology.

NUREG/CR-2353 V01: SPECIFICATION AND VERIFICATION OF NUCLEAR POWER PLANT TRAINING SIMULATOR RESPONSE CHARACTERISTICS:PART I. Summary of Current Practices For Nuclear And Non-Nuclear Simulators. HAAS,P.M. Oak Ridge National Laboratory. BELSTERLING,C.; PANDEY,S.; et al. Franklin Institute/Franklin Research Center. January 1982. 100pp. 8201300349. ORNL/TM-7985/P1. 11759:001.

This report reviews the methods and practices of the nuclear industry for specifying and verifying the performance characteristics of simulators used to train nuclear power plant operators. It also reviews the training simulator methods and practices of selected non-nuclear industries (in particular, the civilian and military aircraft industries) and compares them with those of the nuclear industry. In addition, it identifies non-nuclear methods that might be profitably adopted by the nuclear industry and perhaps included in a simulator standard endorsed by the Nuclear Regulatory Commission. Final conclusions and recommendations are discussed in a companion report (NUREG/CR-2353, Vol. 2; ORNL/TM-7986).

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-2354: LOCAL VOLUME-AVERAGED TRANSPORT EQUATIONS FOR MULTIPHASE FLOW IN REGIONS CONTAINING DISTRIBUTED SOLID STRUCTURES. SHA,W.T.;

SOO, S. L. ; CHAO, B. T. Argonne National Laboratory. February 1982. 57pp. 8203180431. ANL-81-69. 12341:112.

Local volume averaged conservation equations for multiphase flow in regions containing a distributed fixed solid structure with heat transfer are treated. In addition to the usual concept of volume porosity and distributed resistance, the anisotropic nature of the porous media is accounted for by formulating the equations in terms of velocities of phases associated with directional surface permeability. The resulting basic equations are in the form of differential integral transport equations with probability integrals based on the configurations and velocities of the interface. Limiting cases of pure stratified flow and highly dispersed mixture flow are noted. Validity and basis of the governing equations used in the COMMIX-2 code at its present stage of development are demonstrated.

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-2360: TRUST: A COMPUTER PROGRAM FOR VARIABLY SATURATED FLOW IN MULTIDIMENSIONAL, DEFORMABLE MEDIA. REISENAUER, A. ; KEY, K. T. ; NARASIMHAN, T. ; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. January 1982. 250pp. 8202020060. PNL-3975. 11793:163.

The computer code, TRUST, provides a versatile tool to solve a wide spectrum of fluid flow problems arising in variably saturated deformable porous media. The governing equations express the conservation of fluid mass in an elemental volume that has a constant volume of solid. Deformation of the skeleton may be nonelastic.

Permeability and compressibility coefficients may be nonlinearly related to effective stress. Relationships between permeability and saturation with pore water pressure in the unsaturated zone may include hysteresis. The code developed by T.N. Narisimhan grew out of the original TRUMP code written by A.L. Edwards. The code uses an integrated finite difference algorithm for numerically solving the governing equation. Marching in time is performed by a mixed explicit-implicit numerical procedure in which the time step is internally controlled. The time step control and related feature in the TRUST code provide an effective control of the potential numerical instabilities that can arise in the course of solving this difficult class of nonlinear boundary value problems. This document brings together the equations, theory, and users manual for the code as well as a sample case with input and output.

NUREG/CR-2360 ERR: TRUST:A COMPUTER PROGRAM FOR VARIABLY SATURATED FLOW IN MULTIDIMENSIONAL,DEFORMABLE MEDIA. * Battelle Memorial Institute, Pacific Northwest Laboratory. March 12, 1982. 2pp. 8204160067. PNL-3975. 12719:224.

The computer code, TRUST, provides a versatile tool to solve a wide spectrum of fluid flow problems arising in variably saturated deformable porous media. The governing equations express the conservation of fluid mass in an elemental volume that has a constant volume of solid. Deformation of the skeleton may be nonelastic.

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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-2364 V01: PROJECTION MODELS FOR HEALTH EFFECTS ASSESSMENT IN POPULATIONS EXPOSED TO RADIOACTIVE AND NONRADIOACTIVE POLLUTANTS.Vol. 1-Introduction To The SPAHR Demographic Model For Health Risk. COLLINS,J.J.; LUNDY,R.T.; GRAHN,D.; et al. Argonne National Laboratory. November 1982. 92pp. 8212060429. ANL-81-59.

16347:124.

This report, Volume 1 of a 5 volume report presents the Simulation Package for the Analysis of Health Risk (SPAHR), a computer software package based upon a demographic model for health risk projections. The model extends several health risk projection models by making realistic assumptions about the population at risk, and thus represents a distinct improvement over previous models. Complete documentation for use of SPAHR is contained in this publication. The demographic model in SPAHR estimates population response to environmental toxic exposures. Latency of response, changing dose level over time, competing risks from other causes of death, and population structure can be incorporated into SPAHR to project health risks. Risks are measured by morbid years, number of deaths, and loss of life expectancy. Comparisons of estimates of excess deaths demonstrate that previous health risk projection models may have underestimated excess deaths by a factor of from 2 to 10, depending on the pollutant and the exposure scenario. The software supporting the use of the demographic model is designed to be user oriented. Complex risk projections are made by responding to a series of prompts generated by the package. The flexibility and ease of use of SPAHR make it an important contribution of existing models and software packages.

NUREG/CR-2364 V02: PROJECTION MODELS FOR HEALTH EFFECTS ASSESSMENT IN POPULATIONS EXPOSED TO RADIOACTIVE AND NONRADIOACTIVE POLLUTANTS. Vol. 2-SPAHR Introductory Guide. COLLINS, J. J.; LUNDY, R. T. Argonne National Laboratory. November 1982. 42pp. 8212060441. ANL-81-59. 16346:026.

This report, Volume 2 of 5 volume report presents the Simulation Package for the Analysis of Health Risk (SPAHR), a computer software package based upon a demographic model for health risk projections. The model extends several health risk projection models by making realistic assumptions about the population at risk, and thus represents a distinct improvement over previous models. Complete documentation for use of SPAHR is contained in this publication. The demographic model in SPAHR estimates population response to environmental toxic exposures. Latency of response, changing dose level over time, competing risks from other causes of death, and population structure can be incorporated into SPAHR to project health risks. Risks are measured by morbid years, number of deaths, and loss of life expectancy. Comparisons of estimates of excess deaths demonstrate that previous health risk projection models may have underestimated excess deaths by a factor of from 2 to 10, depending on the pollutant and the exposure scenario. The software supporting the use of the demographic model is designed to be user oriented. Complex risk projections are made by responding to a series of prompts generated by the package. The flexibility and ease of use of SPAHR make it an important contribution to existing models and software packages.

NUREG/CR-2364 V03: PROJECTION MODELS FOR HEALTH EFFECTS ASSESSMENT IN POPULATIONS EXPOSED TO RADIOACTIVE AND NONRADIOACTIVE POLLUTANTS. Vol. 3-SPAHR Interactive Package Guide. COLLINS, J. J. Argonne National Laboratory. November 1982. 40pp. 8212060447. ANL-81-59. 16346:068.

This report, Volume 3 of a 5 volume report presents the Simulation for the Analysis of Health Risk (SPAHR), a computer software package based upon a demographic model for health risk projections. The model extends several health risk projection models by making realistic assumptions about the population at risk, and thus represents a distinct improvement over previous models. Complete documentation for use of SPAHR is contained in this publication. The demographic model in SPAHR estimates population response to environmental toxic exposures. Latency of response, changing dose level over time, competing risks from other causes of death, and population structure can be incorporated into SPAHR to project health risks. Risks are measured by morbid years, number of deaths, and loss of life expectancy. Comparisons of estimates of excess deaths demonstrate that previous health risk projection models may have underestimated excess deaths by a factor of from 2 to 10, depending on the pollutant and the exposure scenario. The software supporting the use of the demographic model is designed to be user oriented. Complex risk projections are made by responding to a series of prompts generated by the package. The flexibility and ease of use of SPAHR make it an important contribution to existing models and software packages.

NUREG/CR-2364 V04: PROJECTION MODELS FOR HEALTH EFFECTS ASSESSMENT IN POPULATIONS EXPOSED TO RADIOACTIVE AND NONRADIOACTIVE POLLUTANTS. Vol. 4-SPAHR User's Guide. COLLINS, J. J.; LUNDY, R. T. Argonne National Laboratory. November 1982. 81pp. 8212060451. ANL-81-59. 16346:196.

This report, Volume 4 of a 5 volume report presents the Simulation Package for the Analysis of Health Risk (SPAHR), a computer software package based upon a demographic model for health risk projections. The model extends several health risk projection models by making realistic assumptions about the population at risk, and thus represents a distinct improvement over previous models. Complete documentation for use of SPAHR is contained in this publication. The demographic model in SPAHR estimates population response to environmental toxic exposures. Latency of response, changing dose level over time, competing risks from other causes of death, and population structure can be incorporated into SPAHR to project health risks. Risks are measured by morbid years, number of deaths, and loss of life expectancy. Comparisons of estimates of excess deaths demonstrate that previous health risk projection models may have underestimated excess deaths by a factor of from 2 to 10, depending on the pollutant and the exposure scenario. The software supporting the use of the demographic model is designed to be user oriented. Complex risk projections are made by responding to a series of prompts generated by the package. The flexibility and ease of use of SPAHR make it an important contribution to existing models and software packages.

NUREG/CR-2364 V05: PROJECTION MODELS FOR HEALTH EFFECTS ASSESSMENT IN POPULATIONS EXPOSED TO RADIOACTIVE AND NONRADIOACTIVE POLLUTANTS. Vol. 5-SPAHR Programmer's Guide. COLLINS, J. J.; LUNDY, R. T. Argonne National Laboratory. November 1982. 81pp. 8212060455. ANL-81-59. 16346:278.

This report, Volume 5 of 5 volume report presents the Simulation Package for the Analysis of Health Risk (SPAHR), a computer software package based upon a demographic model for health risk projections. The model extends several health risk projection models by making realistic assumptions about the population at risk, and thus represents a distinct improvement over previous models. Complete documentation for use of SPAHR is contained in this publication. The demographic model in SPAHR estimates population response to environmental toxic exposures. Latency of response, changing dose level over time, competing risks from other causes of death, and population structure can be incorporated into SPAHR to project health risks. Risks are measured by morbid years, number of deaths, and loss of life expectancy. Comparisons of estimates of excess deaths demonstrate that previous health risk projection models may have underestimated excess deaths by a factor of from 2 to 10, depending on the pollutant and the exposure scenario. Complex risk projections are made by responding to a series of prompts generated by the package. The flexibility and ease of use of SPAHR make it an important contribution to existing models and software packages.

NUREG/CR-2366 V01: MULTIROD BURST TEST PROGRAM PROGRESS REPORT FOR JANUARY -JUNE 1981. LONGEST,A.W. Oak Ridge National Laboratory. January 1982. 175pp. 8201270393. ORNL/TM-8058. 11710:351.

Major activities in progress in the Multirod Burst Test (MRBT) Program during the report period included (1) performance of the bundle B-4 test and completion of part of its posttest examination; (2) performance of posttest water flow testing, casing, and sectioning of bundle B-5; (3) fabrication of fuel pin simulators for bundle B-6; (4) performance of three special single rod tests to evaluate the effect on deformation of temperature measuring sensors used in Japanese Atomic Energy Research Institute (JAERI) bundle tests; and (5) performance of three single rod tests to aid selection of test conditions for the next bundle test (B-6). Also, an average deformation model was developed to predict MRBT tube deformation from pressure and temperature test data. An ongoing evaluation of the bundle B-5 burst test information has led to several conclusions that appear to have very important implications for the interpretation of past tests and the design of future tests.

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 ±1%.

NUREG/CR-2367: UPDATED BEST-ESTIMATE LOCA RADIATION SIGNATURE. * Sandia Laboratories. THAYER,D.D.; HOUSTON,D.H.; et al. IRT Corp. January 1982. 61pp. 8201130418. SAND81-7159. 11572:058.

Over the past several years, the Qualification Testing Evaluation (QTE) Program has been addressing the problem of radiation source term definition. As part of the QTE Program, IRT Corporation has conducted studies focusing on expected source terms resulting from the hypothesized loss-of-coolant accident (LOCA).

The objective of this study was to review research conducted since the last study and make modifications and improvements in the "best-estimate" LOCA radiation signature. This report contains a review of the fission product source term literature since 1978, changes made in the "best-estimate" source term, comparisons with the previous "best estimate" source term in Ref. 4, and suggestions for future work.

NUREG/CR-2369: POWER HISTORIES FOR FUEL CODES. GILBERT, E. R.; RAUSCH, W. N.; PANISKO, F. E. Battelle Memorial Institute, Pacific Northwest Laboratory. January 1982. 44pp. 8201290015. PNL-4059. 11750:043.

Computations of power history effects on the pre-loss-of-coolant accident conditions of generic pressurized water reactor and boiling water reactor fuel rods were preformed at Pacific Northwest Laboratory using the U.S. Nuclear Regulatory Commission code FRAPCON-2. Comparisons were made between cases where the fuel operated at a high power throughout life and those where the fuel was at a lower power for most of its burnup and ramped to the high power at 10,000 or 20,000 MWd/MTU burnup.

The PWR was calculated to have more cladding creepdown during the lower power cases, which resulted in slightly lower centerline temperature. This result was insensitive to the method used to increase the power during the ramps. The calculations also indicate that the highest fuel centerline temperatures were reached at startup.

The BWR rod, however, demonstrated a substantial dependence on the power history. In this case, the constant high-power rod released considerable more fission gas than the lower power cases, which resulted in temperature differences of up to 350 degrees C. The highest temperature was reached at end-of-life in the constant high-power case.

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-2378: NUCLEAR POWER PLANT OPERATING EXPERIENCE 1980. MAYS, G. T.; HARIED, J. A.; KUKIELKA, C. A.; et al. Oak Ridge National Laboratory. October 1982. 339pp. 8211030499. ORNL/NSIC-191. 15916:062.

This report is the seventh in a series of reports issued annually that summarize the operating experience of nuclear plants in commercial operation in the United States. Power generation statistics, plant outages, reportable occurrences, fuel element performance, and occupational radiation exposure for each plant are presented and discussed, and summary highlights are given. The report includes 1980 data from 67 plants - 24 boiling-water-reactor plants, 42 pressurized-water-reactor plants, and 1 high-temperature gas-cooled reactor plant.

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-2382 ERR: Errata, changing title to A SUMMARY OF DATING METHODS FOR THE DETERMINATION OF THE LAST TIME OF MOVEMENT OF FAULTS. CURTIS, G. H. Curtis, G. H. January 5, 1982. 1p. 8201180277. 11604:042.

This is a guide to methods of dating the past activity of faults with emphasis on methods to dating movement over the past 500,000 years. This period of time is of most concern to nuclear power plant siting regulations which state that a fault is "capable" (i.e. shows movement at or near the ground surface at least once within the past 35,000 years or movement of a recurring nature within the past 500,000 years). The guide presents: Chronographic Methods; isotopic methods - Primeval Radioactive Elements; and isotopic methods - cosmic ray produced Radioactive elements.

NUREG/CR-2383: RADIONUCLIDE DISTRIBUTIONS AND MIGRATION MECHANISMS AT SHALLOW LAND BURIAL SITES. Annual Report of Research Investigations On The Distribution, Migration And Containment of Radionuclides At Maxey Flats, Kentucky. KIRBY, L. J. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1982. 157pp. 8208130480. PNL-4067. 14352:037.

Subsurface waters at Maxey Flats are anoxic systems with high alkalinity and high concentrations of dissolved ferrous iron. Americium and cobalt in these trench waters are made more soluble by the presence of EDTA, while strontium and cesium are unaffected under the same conditions. EDTA is the major organic complexing component in waste trench 27 leachate, but other polar, water-soluble organics are also present. Evidence points to the migration of plutonium between waste trench 27 and inert atmosphere wells as an EDTA complex. Polar organic compounds may influence the migration of Sr(90) and Cs(137). The primary pathway of water entry into the waste burial trenches is through the trench caps, but major increases in water level have occurred in an experimental trench by subsurface flow. The areal distribution of radionuclides at Maxey Flats has been influenced by surface runoff, deposition from the evaporator plume, subsurface flow and the actions of burrowing animals or deep-rooted trees. Vegetal and surface contamination on site and near site are quite low, and only Co(60) exceeds commonly observed fallout levels. Radionuclide concentrations in surface soil at Maxey Flats are comparable to concentrations resulting from normal fallout in other areas of high rainfall.

NUREG/CR-2384: AGE-SPECIFIC INHALATION RADIATION DOSE COMMITMENT FACTORS FOR SELECTED RADIONUCLIDES. STRENGE, D. L.; PELOQUIN, R. A.; BAKER, D. A. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1982. 42pp. 8209230016. PNL-3912. 14984:256.

Inhalation dose commitment factors are presented for selected radionuclides for exposure of individuals in four age groups: infant, child, teen and adult. Radionucludes considered are (35)S, (36)Cl, (45)Ca, (67)Ga, (75)Se, (85)Sr, (109)Cd, (113)Sn, (125)I, (133)Ba, (170)Tm, (169)Yb, (182)Ta, (192)Ir, (198)Au, (201)Tl, (204)Tl, and (236)Pu. The calculational method is based on the human metabolic model of ICRP as defined in Publication 2 (ICRP 1959) and as used in previous age-specific dose factor calculations by Hoenes and Soldat (1977). Dose commitment factors are presented for the following organs of reference: total body, bone, liver, kidney, thyroid, lung and lower large intestine.

NUREG/CR-2385: CSQ CALCULATIONS OF HYDROGEN DETONATIONS IN THE ZION AND SEQUOYAH NUCLEAR PLANTS. BYERS, R. K. Sandia Laboratories. September 1982. 55pp. 8209280333. SAND81-2216. 15543:043.

Sandia National Laboratories is engaged in an extensive program, sponsored by USNRC, involving many safety-related aspects of hydrogen mixtures in reactor containment buildings. The questions addressed in the program include the generation, transport, and removal of hydrogen, as well as combustion of hydrogen mixtures. This report deals with a very limited aspect of Sandia's hydrogen program: the estimation of detonation-caused loads on containment structures. The response of the containment buildings to the loads is not carefully examined here and would require further study.

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: PATH 1 SELF-TEACHING CURRICULUM: EXAMPLE PROBLEMS FOR PATHWAYS-TO-MAN MODEL. HELTON, J. C. ; FINLEY, N. C. Sandia Laboratories. October 1982. 114pp. 8301170022. SAND81-2377. 16818:317.

This report contains a series of sample problems and solutions for the Pathways-to-Man (PATH 1) model developed at Sandia National

Laboratories for the Risk Methodology for Geologic Disposal of Radioactive Waste Project. With this document and the PATH 1 User's Manual (NUREG/CR-1636 Vol. 1), the user may familiarize himself with the computer program, its capabilities and limitations. When the user has completed this curriculum, he or she should be able to prepare data input for PATH 1 and have some insights into interpretation of the model output. 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-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-2395: WIND TUNNEL STUDY OF GAS DISPERSION NEAR A CUBICAL MODEL BUILDING. LI, W. W.; MERONEY, R. N.; PETERKA, J. A. Colorado State Univ. March 1982. 112pp. 8204070116. 12592:101.

The dispersion of effluent plumes emitted from the surface of a cubical model building into its near wake region (less than five building heights downwind) has been examined. The model study was performed in a wind tunnel with a simulated neutrally stratified shear layer. Mean concentration measurements were made on the building surface and within the near wake region of the model building. Measurements of the concentration fluctuation intensity and the peak-to-mean concentration ratio were also conducted in the near wake. The concentration level on the lee face of the model building is greatly reduced by the presence of a nearby sharp building edge. The optimum location for an intake vent on the building, for equal effluent exhaust to vent intake distances, is a position not directly downwind and at a location where the intake cannot "see" the exhaust vent. The log-normal concentration probability model was found appropriate for measurements in the building wake between one and five building heights downwind. The concentration fluctuation intensity was found to be reduced by the presence of the model building from that of a plume released in an obstructed flow. A simple algorithm, based on the relation of the peak-to-mean concentration ratio and the local concentration fluctuation intensity, suggests an upper limit for the peak-to-mean concentration ratios near the ground centerline.

NUREG/CR-2400: TACTICAL IMPROVEMENT PACKAGE. BAEHR, D. G. ; HEIDER, J. A. ; ADAMS, K. G. Sandia Laboratories. July 1982. 82pp. 8208260414. SAND82-0462. 14589:309.

The Tactical Improvement and Security Force Evaluation Program, which demonstrates the feasibility of using the Engagement Simulation System (ESS) at licensee nuclear facilities, requested by the Nuclear Regulatory Commission, is described. Background information on the ESS and observations on its use, based on exercises at four licensee facilities, are provided. The information required by the licensee to utilize the ESS for security officer training is presented in the form of a Tactical Improvement Package (TIP). Two Instructor's Guides (an expanded and an abbreviated version) are included as aides to interested users. A video tape that complements the test is available on loan from the NRC.

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-2404: ANALYZING SAFEGUARDS ALARMS AND RESPONSE DECISIONS. AL-AYAT, R. A. ; JUDD, B. R. ; MCCORD, R. K. Lawrence Livermore Laboratory. September 1982. 115pp. 8210150548. UCRL-53034. 15726:133.

This report describes a quantitative model designed to help the Nuclear Regulatory Commission (NRC) and its licensees evaluate and respond to alarms indicating that special nuclear material (SNM) may be missing. The model is called the Alarm/Response (A/R) Model.

The report demonstrates three principal uses of the A/R Model. The first is determining the most likely cause of an alarm--theft, hoax, or error. The second is evaluating alternative responses to alarms. Possible responses include conducting investigations, initiating measures to recover stolen SNM, and replying to extortion threats from individuals claiming to possess SNM. For each possible alarm, the model identifies the best response, which can be used to develop contingency plans that the licensee and the NRC can carry out. The third use is to assist the NRC in setting performance standards, especially detection time requirements. In this application, the model helps to determine the value of more timely alarms produced by licensee safeguards systems. All three uses are demonstrated with hypothetical examples. A preliminary computer code produced sample results and determined the sensitivity of those results to subjective factors in the example.

NUREG/CR-2405: SUBSYSTEM FRAGILITY. Seismic Safety Margins Research Program (Phase I). KENNEDY, R. P. ; CAMPBELL, R. D. ; HARDY, G. ; et al. Lawrence Livermore Laboratory. February 1982. 230pp. 8203030237.

UCRL-15407. 12112:001.

Seismic fragility levels of safety-related equipment are developed for use in a seismic oriented Probabilistic Risk Assessment (PRA) being conducted as part of the Seismic Safety Margins Research Program (SSMRP). The Zion Nuclear Power Plant is being utilized as a reference plant and fragility descriptions are developed for specific and generic safety-related equipment groups in Zion. Both equipment fragilities and equipment responses are defined in probabilistic terms to be used as input to the SSMRP event tree/fault tree models of the Zion systems.

NUREG/CR-2407: RADON AND AEROSOL RELEASE FROM OPEN PIT URANIUM MINING. THOMAS, V.W.; NIELSON, K.K.; MAUCH, M.L. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1982. 384pp. 8208260075. PNL-4071. 14578:019.

The model open pit uranium mine reported in NUREG/CR-0628 has been redefined based on a 1981 survey of estimated mining parameters to cause net radon releases of 3300 Ci/yr compared to 2000 Ci/yr from the original model. This equals 840 Ci/RRY (201 tonnes U(3)O(8)/Reference Reactor Year (RRY)) compared to 700 Ci/RRY from the original model. After mining, radon releases continue at the elevated rate of 2700 Ci/yr (43 Ci/yr/RRY) compared to the earlier model estimate of 1380 Ci/yr (29 Ci/yr/RRY). The net increase over the background radon emission rate during preparatory open pit mining at Morton Ranch 1704 pit was estimated from a one-year study to be 150 Ci/yr. The release rate projected by Argonne National Laboratory (ANL) from the St. Anthony mine pit, New Mexico, of approximately 50 Ci/yr at its midlife (8.5 years) was a factor of 17 lower than the estimated release from the active pit of the model mine (830 Ci/yr) at its midlife (also 8.5 years). Due to the large variation found in measurements made in the natural environment, large uncertainty estimates were associated with most measurements.

NUREG/CR-2408: CURVE FITTING AND UNCERTAINTY ANALYSIS OF CHARPY IMPACT DATA. STALLMAN, F.W. Oak Ridge National Laboratory. January 1982. 26pp. 8202040091. ORNL/TM-8081. 11818:116.

A method for the statistical evaluation of Charpy impact data is described. A multi-variate linear least squares fit is used in which both dependent and independent variables may be subject to statistical variations. The input data are restricted to the transition region for the determination of RT(NDT) and to the upper-shelf region for the determination of the upper-shelf energy. Fluence, irradiation temperature, chemical composition, and other parameters may be included in the fit in order to process simultaneously large data sets and to estimate the dependency of RT(NDT) and shift of upper-shelf energy on the parameters. Data from HSST experiments at ORNL are used to demonstrate the application of this method.

NUREG/CR-2409: REQUIREMENTS FOR ESTABLISHING DETECTOR SITING CRITERIA IN FIRES INVOLVING ELECTRICAL MATERIALS. BOCCIO, J.L. Sandia Laboratories. September 1982. 55pp. 8209280332. SAND81-7168. 15543:146.

Due to increased public awareness and regulatory actions, significant strides have been made in the capabilities of fire technology as it applies to fire detection systems. However, these advances in detector selection, siting, reliability and approvals tests have not substantially addressed the overall fire protection requirements within nuclear reactors. This report emphasizes some of the basic requirements and considerations needed for establishing siting criteria for early-warning detection of electrical cable fires. Recent research in electrical cable flammability and damageability characteristics are discussed. Also current work in systemizing detector siting criteria is also described. Confirmatory tests linking assessment of electrical-cable damageability with electrical cable fire detection is stressed.

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 RAG 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-2419: REFERENCE REPOSITORY DEFINITION, SHALE. FINLEY, N. C. Sandia Laboratories. MCKAY, E. D. ; DONATH, F. A. ; et al. CGS, Inc. January 1982. 148pp. 8201130425. SAND81-7165. 11579:089.

This report examines the geologic and hydrologic settings of three regions in North America as the basis for selection of one natural system to serve as a model for a reference repository system in shale (RRS-SH). In this RRS, shale is the candidate host medium for a high-level radioactive waste depository. The three regions studied are: (1) the Illinois Basin in Illinois, (2) the Williston Basin in South Dakota, and (3) the western Canada sedimentary basin in Alberta.

NUREG/CR-2420: REFERENCE REPOSITORY DEFINITION, GRANITE. FINLEY, N. C. Sandia Laboratories. FRYER, K. H. ; FRUTH, L. S. ; et al. CGS, Inc. January 1982. 63pp. 8201130408. SAND81-7166. 11572:197.

This study examines a "typical" granitic pluton prerequisite to the definition and characterization of a Reference Repository System in granite (RR-GR). Sand Springs Range, Churchill County, Nevada, was selected for the study mainly on the basis of the more abundant and detailed geologic and hydrologic data that are available as a result of its selection for the Shoal Event, Vela Uniform Program of the Atomic Energy Commission. Ground-water flow within a granite repository system will be controlled largely by the fracture field within the granitic mass and will be strongly influenced by valley-fill sediments which flank that mass. Examination of the data available for this pluton suggests that a model for the RRS-GR should consider a rather uniform fracture system. This system should be represented by two or more sets of fractures, faults, and/or fracture cleavage. Fractures ranging from closed to open and possibly filled with gouge, as well as dikes of varying thickness and characteristics, need to be represented. Typically, the physical properties required for accurate modeling are known only for intact rock specimens and not for the rock mass. Additional in situ measurements of rock properties would be desirable, but in their absence the mass properties will need to be inferred from available intact specimen data or taken from in situ measurements made elsewhere.

NUREG/CR-2423: MATHEMATICAL SIMULATION OF SEDIMENT AND RADIONUCLIDE TRANSPORT IN ESTUARIES. ONISHI, Y. ; TRENT, D. S. Battelle Memorial

Institute, Pacific Northwest Laboratory. November 1982. 81pp. 8212220509. PNL-4109. 16525:266.

The finite element model FLESCOT (Flow, Energy, Salinity, Sediment and Containment Transport Model) was synthesized under this study to simulate radionuclide transport in estuaries to obtain accurate radionuclide distributions which are affected by these factors: time variance, three-dimensional flow, temperature, salinity, and sediments. Because sediment transport and radionuclide adsorption/desorption depend strongly on sizes or types of sediments, FLESCOT simulates sediment and a sediment-sorbed radionuclide for the total of three sediment-size fractions (or sediment types) of both cohesive (e.g., silt and clay) and noncohesive (e.g., sand) sediments. It also calculates changes of estuarine bed conditions, including bed elevation changes due to sediment erosion/deposition, and three-dimensional distributions of three bed sediment sizes and sediment-sorbed radionuclides within the bed. Although the model was synthesized for radionuclide transport, it is general enough to also handle other contaminants such as heavy metals, pesticides, or toxic chemicals.

The model was applied to the Hudson River estuary using existing field data and data from the Indian Point Nuclear Station. Agreement between predicted and calculated results was reasonable.

NUREG/CR-2425: SEDIMENTS AND RADIONUCLIDE TRANSPORT IN RIVERS:Radionuclide Transport Modeling For Cattaraugas And Buttermilk Creeks, New York. ONISHI,Y.; YABUSAKI,S.B.; KINCAID,C.T.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1982. 398pp. 8301120109. PNL-4111. 16785:054.

SERATRA, a transient, two-dimensional (laterally-averaged) computer model of sediment-contaminant transport in rivers, satisfactorily resolved the distribution of sediment and radionulcide concentrations in the Cattaraugus Creek stream system in New York. By modeling the physical processes of advection, diffusion, erosion, deposition, and bed armoring, SERATRA routed three sediment size fractions, including cohesive soils, to simulate three dynamic flow events. In conjunction with the sediment transport, SERATRA computed radionuclide levels in dissolved suspended sediment, and bed sediment dependent forms for four radionuclides ((137) Cs, (90) Sr, (239), (240) Pu, and (3) H). By accounting for time dependent sediment-radionuclide interaction in the water column and bed, SERATRA is a physically explicit model of radionuclide fate and migration. Sediment and radionuclide concentrations calculated by SERATRA in the Cattaraugus Creek stream system are in reasonable agreement with measured values.

NUREG/CR-2429: DATA BASE FOR RADIATION EVENTS IN THE COMMERCIAL NUCLEAR FUEL CYCLE 1950-1978. BODEAU,D.J.; KULLEN,B.J.; LUNER,C.; et al. Argonne National Laboratory. March 1982. 83pp. 8203120003. ANL/ES-123. 12252:251.

Information reported to the U.S. Nuclear Regulatory Commission by commercial license holders through 1978 on 1634 incidents involving any level of radioactive risk to either workers or the public has been compiled. These incidents have been coded and reduced to a computer format that permits quick access to many details of each event, as well as a short narrative summary. The data base includes all American incidents involving radioactivity at any point in the fuel cycle, except for incidents at reactor site and government facilities. This report provides instruction in the techniques of accessing the data base.

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-2433: NUCLEAR REACTOR COMPONENT POPULATIONS, RELIABILITY DATA BASES, AND THEIR RELATIONSHIP TO FAILURE RATE ESTIMATION AND UNCERTAINTY ANALYSIS. MARTZ, H. F.; BECKMAN, R. J. Los Alamos Scientific Laboratory. March 1982. 29pp. 8204290607. LA-9115-MS. 12895:205.

Probabilistic risk analyses are used to assess the risks inherent in the operation of existing and proposed nuclear power reactors. In performing such risk analyses the failure rates of various components which are used in a variety of reactor systems must be estimated. These failure rate estimates serve as input to fault trees and event trees used in the analyses.

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.

NUREG/CR-2437 V01: LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM.Quarterly Progress Report.January-March 1981. MASSEY,W.E.; TILL,C.E. Argonne National Laboratory. January 1982. 20pp. 8201270400. ANL-81-77 V01. 11705:302.

This progress report summarizes the work performed during January, February, and March 1981 on water-reactor-safety problems. The research and development area covered Transient Fuel Response and Fission-product Release.

NUREG/CR-2437 V02: LIGHT-WATER REACTOR SAFETY RESEARCH PROGRAM.Quarterly Progress Report April-June 1981. MASSEY,W.E.; TILL,C.E. Argonne National Laboratory. February 1982. 50pp. 8203180440. ANL-81-77 V02. 12341:063.

This progress report summarizes the work performed during April, May, and June 1981 on water-reactor-safety problems. The research and development areas covered are Transient Fuel Response and Fission-product Release and Environmentally Assisted Cracking in Light Water Reactors.

NUREG/CR-2437 V03: LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM. Quarterly Progress Report. July-September 1981. MASSEY, W. E.; TILL, C. E. Argonne National Laboratory. March 1982. 106pp. 8204010548. ANL-81-77 V03. 12487:015.

This progress report summarizes work performed during July, August, and September 1981 on water-reactor safety problems. The research and development areas covered are Environmentally Assisted Cracking in Light Water Reactors, Transient Fuel Response and Fission-product Release, and Clad Properties for Code Verification.

NUREG/CR-2437 V04: MATERIALS SCIENCE DIVISION LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM. Quarterly Progress Report, October-December 1981. * Argonne National Laboratory. July 1982. 110pp. 8208130472. ANL-81-77 V04. 14353:078.

This progress report summarizes work performed during October, November, and December 1981 on water-reactor-safety problems. The research and development areas covered are Environmentally Assisted Cracking in Light Water Reactors, Transient Fuel Response and Fission-product Release, and Clad Properties for Code Verification.

NUREG/CR-2438: WET DEPOSITION PROCESSES FOR RADIOIODINES. KELLER, J. H.; HOFFMAN, L. G. Exxon Nuclear Co., Inc. (subs. of Exxon Corp.). VOILLEQUE, P. G. Science Applications, Inc. August 1982. 74pp. 8209020206. ENICO-1111. 14726:207.

The results of a review of wet deposition processes for radioiodines are presented. Experimental measurements have shown that precipitation scavenging can be an important process for removal of radioiodines released to the atmosphere. Measurements of radioiodine species in routine gaseous effluents have shown that the distribution depends upon the type of nuclear facility considered. Because of differences in radioiodine species, observed scavenging rates for radioiodine in fallout are not generally applicable to routine releases from nuclear facilities. Adequate models are available to estimate the air to surface transport by wet processes. At the present time, the need is for more information on the rate constants used in the models. Reactor accident consequence calculations that assume no behavior differences among iodine species indicate that about half the predicted early deaths and illnesses are attributable to radioiodines. The distribution of species released under accident conditions will depend upon the actual sequence of events. However, for many scenarios the wet deposition of radioiodines will have an important bearing on the consequences.

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. 205pp. 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-2444 V01: PWR FLECHT SEASET 21-ROD BUNDLE FLOW BLOCKAGE TASK DATA AND ANALYSIS REPORT, NRC/EPRI/WESTINGHOUSE REPORT NO. 11. LOFTUS, M. J.; HOCHREITER, L. E.; LEE, N.; et al. Westinghouse Electric Corp. September 1982. 560pp. 8210210054. EPRI NP-2014. 15783:001.

This report presents data and limited analysis from the 21-Rod Bundle Flow Blockage Task of the Full-Length Emergency Cooling Heat Transfer Separate Effects and Systems Effects Test Program (FLECHT-SEASET). The tests consisted of forced and gravity reflooding tests utilizing electrical heater rods with a cosine axial power profile to simulate PWR nuclear core fuel rod arrays. Steam cooling and hydraulic characteristics tests were also conducted. These tests were utilized to determine effects of various flow blockage configurations (shapes and distributions) on reflooding behavior, to aid in development/assessment of computational models on predicting reflooding behavior of flow blockage configurations, and to screen flow blockage configurations for future 163-rod flow blockage bundle tests.

NUREG/CR-2444 V02: PWR FLECHT SEASET 21-ROD BUNDLE FLOW BLOCKAGE TASK DATA AND ANALYSIS REPORT, NRC/EPRI/WESTINGHOUSE REPORT NO. 11. LOFTUS, M. J.; HOCHREITER, L. E.; LEE, N.; et al. Westinghouse Electric Corp. September 1982. 906pp. 8210220154. EPRI NP-2014. 15796:128.

This report presents data and limited analysis from the 21-Rod Bundle Flow Blockage Task of the Full-Length Emergency Cooling Heat Transfer Separate Effects and Systems Effects Test Program (FLECHT-SEASET). The tests consisted of forced and gravity reflooding tests utilizing electrical heater rods with a cosine axial power profile to simulate PWR nuclear core fuel rod arrays. Steam cooling and hydraulic characteristics tests were also conducted. These tests were utilized to determine effects of various flow blockage configurations (shapes and distributions) on reflooding behavior, to aid in development/assessment of computational models on predicting reflooding behavior of flow blockage configurations, and to screen flow blockage configurations for future 163-rod flow blockage bundle tests.

NUREG/CR-2453: NATIONAL RELIABILITY EVALUATION PROGRAM (NREP) OPTIONS STUDY. BUSLIK, A. J.; BARI, R. A. Brookhaven National Laboratory. January 1982. 94pp. 8201220035. BNL-NUREG-51485. 11659:197.

Options are identified for implementing the National Reliability Evaluation Program (NREP). Considerations are given to different contingencies related to using the results, availability of a standard methodology, the review process, and the resources required to undertake the program. Descriptions are given of the options as to the end uses of the NREP risk studies and how they, and the scopes of the NREP risk studies, are related to safety benefits. Essential elements of the NREP study are presented, as well as methodologies available for these elements. Some comments are provided on the state-of-the-art of the various methodologies.

NUREG/CR-2454: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A NATURAL CIRCULATION TESTS S-NC-2B AND S-NC-4B. O'CONNELL, T. M. EG&G, Inc. January 1982. 73pp. 8201210134. EGG-2141. 11641:144.

This report presents recorded test data for Tests S-NC-2B, and S-NC-4B of the Semiscale Mod-2A Natural Circulation Test Series. These are part of a series of Semiscale tests that investigates the thermal-hydraulic phenomena resulting from operational transients or small break loss-of-coolant accidents (LOCA) involving the loss of mechanical primary coolant circulation in a pressurized water reactor (PWR). These tests produce experimental data to develop and assess the analytical capability of computer models used to predict the results of small break LOCAs and operational transients involving the loss of primary pumping ability. Test S-NC-2B was a baseline natural circulation test involving single-phase, two-phase, and reflux modes of natural circulation. The test investigated the relationships of primary system mass inventory and core power with the various modes of natural circulation test which investigated the effect of various steam generator secondary side hydraulic conditions on the natural circulation flow rate, with constant core power and primary mass inventory. Test S-NC-4B was a reflux natural circulation test which investigated the effect of steam generator secondary side liquid inventory on the natural circulation flow rate and on other primary system conditions.

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 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-2458: SNUPPS AUXILIARY FEEDWATER SYSTEM RELIABILITY STUDY EVALUATION. ROSCOE, B. J. Sandia Laboratories. January 1982. 45pp. 8203010028. SAND81-2596. 12074:064.

The purpose of this report is to present the results of the review of the Auxiliary Feedwater System Reliability Analysis for SNUPPS.

NUREG/CR-2459 V01: SEMISCALE UNCERTAINTY REPORT:METHODOLOGY. GOLDEN, R. W. EG&G, Inc. September 1982. 57pp. 8211080024. EGG-2142. 15973:302.

Definitions of staistical terms and methods used to derive uncertainty estimates for experimental measurements at the Semiscale Test Facility are presented in this report. Error propagation equations are developed to aid in determining uncertainties for complex calculations. Uncertainty estimates of the data system presented, as well as the other types of errors, which are analyzed in more detail in subsequent volumes that will cover each of the various types of measurements.

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-2461: SUBCOOLED AND LOW QUALITY FILM BOILING OF WATER IN VERTICAL FLOW AT ATMOSPHERIC PRESSURE. FUNG, K. K. Argonne National Laboratory. February 1982. 307pp. 8203190452. ANL-81-78. 12370:083.

Subcooled and low quality film boiling is usually encountered in safety analyses of nuclear reactors. In most of the previous subcooled film boiling studies, cryogenic fluids were used either in a stagnant pool or a forced convective set-up. This data cannot be applied to reactor safety analysis without excessive conservatism or skepticism. In this study, a unique method is used to establish flow film boiling of water in a vertical tube at atmospheric pressure. The data cover a mass flux range of 50-500 kg.m(-2).s(-1) and an inlet subcooling range of 5-70 degrees C. It is found that the heat transfer coefficient depends on the mass flux, inlet subcooling and the axial distance from the point where film boiling first starts. A physical model is developed to predict the wall temperature of a tube during inverted annular film boiling. It considered the thermal boundary layers in the subcooled liquid core and in the superheated vapor film. The predicted wall temperatures and void fractions compare well with the measurements.

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-2465: AN ERROR AND UNCERTAINTY ANALYSIS OF CLASSICAL AND BAYESIAN FAILURE RATE ESTIMATES FROM A POISSON DISTRIBUTION. MARTZ, H. F. Los Alamos Scientific Laboratory. February 1982. 21pp. 8203190046. LA-9132-MS. 12356:345.

An error analysis is presented for the errors and uncertainties that could possibly accompany component failure rate estimates based on a Poisson sampling model within either a classical or a Bayesian estimation framework. Twenty-eight errors are identified based on the use of both objective reliability test data as well as expert opinion in determining the required prior distribution for the failure rate in a Bayesian analysis. The error analysis includes a name, description, possible causes, and likely effects of each error. These twenty-eight errors are further classified into three broad categories: Modeling Errors, Data Uncertainties, and Human Factors. Examples from the nuclear reactor industry are used to illustrate each individual error.

NUREG/CR-2466. STATISTICAL SAMPLING PLANS FOR PRIOR MEASUREMENT VERIFICATION AND DETERMINATION OF THE SNM CONTENT OF INVENTORIES. PIEPEL, G.F. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1982. 100pp. 8204290469. PNL-4057. 12902:041.

Current regulations require that prior information on the Special Nuclear Material (SNM) content of a population of containers be verified and that periodic measurement of the SNM inventory of a facility be performed. This report develops and describes statistical sampling plans for accomplishing these tasks and compares results obtained by sampling to those obtained by the current practice of performing a census (100 percent sampling).

NUREG/CR-2467: LOWER-BOUND KIscc VALUES FOR BOLTING MATERIALS-A LITERATURE STUDY. GOLDBERG, A.; JUHAS, M.C. Lawrence Livermore Laboratory. February 1982. 68pp. 8203090305. UCRL-53035. 12176:023.

An extensive literature survey on stress-corrosion cracking of a variety of steels was made in response to the need by the U.S. Nuclear Regulatory Commission (NRC) to establish lower-bound K(Iscc) values for bolting type materials. Materials evaluated include the heat-treatable plain-carbon and Cr-Mo and Cr-Mo-Ni low alloy steels, 17-4 PH and Custon 455 precipitation-hardening stainless steels, and the 18 Ni maraging steels. An exhaustive survey was made for water, aqueous chloride, and aqueous sulfide environments. We also report limited data on H(2)S and H(2) gases, aqueous H(3)BO(4), and atmospheric environments. The data are presented in the form of K(Iscc) versus yield strength. The proposed NRC lower-bound K(Iscc) curve for the low-alloy steels is consistent with the data reported for the various aqueous environments, but excluding the sulfides. The corresponding lower-bound curve based on reported data for 18 Ni-maraging steels falls below that proposed by the NRC, especially at the high strength levels. The lower-bound curves for the precipitation-hardening stainless steels fall below the lower-bound curve for the maraging steels at the lower strength levels, but they merge at a yield strength of approximately 220 ksi. Reference is also made to crack-growth rate (CGR) and time to failure (t(f)). These two parameters frequently were found to vary differently from the corresponding K(Iscc) values when examined as a function of material or environmental variables. The influence of various material and environmental factors on K(Iscc), CGR, and t(f) are discussed.

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-2471: ANALYSIS OF THE PULSED NEUTRON ACTIVATION TECHNIQUE.
PEREZ-GRIFFO, M.; BLOCK, R.C.; LAHEY, R.T. Rensselaer Polytechnic Inst. February 1982. 200pp. 8203040183. EGG-2160. 12129:001.

The Pulsed Neutron Activation (PNA) Technique for single- and two-phase water flow is analyzed for small and large diameter pipes.

PNA measurements were performed at the RPI Gaerttner Linac Laboratory using the (16)O(n,p)(16)N reaction. Accurate mass-weighted velocities of single- and two-phase flows were obtained by the transit-time method for a vertical 1-in diameter pipe. The potential of (16)N tagging techniques for basic single- and two-phase flow measurements is demonstrated.

Analytical techniques were developed to analyze PNA measurements at LOFT. Neutron and gamma transport calculations by Monte Carlo were done at the tagging and detector positions for the 14-in diameter pipes used at LOFT. Flow structure effects on the interpretation of PNA measurements are evaluated. Dispersion models to describe the transport of the irradiated fluid from the source to detector locations were developed in order to interpret the LOFT L3-7 test PNA data.

NUREG/CR-2472: FINAL REPORT ON SHIPPING CASK SABOTAGE SOURCE TERM INVESTIGATION. SCHMIDT, E. W.; WALTERS, M. A.; TROTT, B. D.; et al. Battelle Memorial Institute, Columbus Laboratories. October 1982. 80pp. 8211080006. BMI-2095. 15974:111.

A need existed to estimate the source term resulting from a sabotage attack on a spent nuclear fuel shipping cask. An experimental program sponsored by the U.S. NRC and conducted at Battelle's Columbus Laboratories was designed to meet that need. In the program a precision shaped charge was fired through a subscale model cask loaded with segments of spent PWR fuel rods and the radioactive material released was analyzed. This report describes these experiments and presents their results.

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-2477: K-FIX (3D,FLX):APPLICATION OF THE K-FIX CODE TO FLUID-STRUCTURE INTERACTION PHENOMENA IN THE HDR GEOMETRY. RIVARD, W. C.; TORREY, M. D. Los Alamos Scientific Laboratory. February 1982. 80pp. 8203190022. LA-9138-MS. 12354:236.

The K-FIX (3D,FLX) code has been applied to calculate fluid-structure interaction phenomena for a light water reactor blowdown experiment. The calculations simulate the German HDR experiment V31.1. A brief description of the code modifications necessary is given as well as comparisons of calculated and experimental data. Complete listings of these modifications are contained in the Appendix.

NUREG/CR-2478 V01: A STUDY OF TRENCH COVERS TO MINIMIZE INFILTRATION AT WASTE DISPOSAL SITES.Task 1 Report - Review of Present Practices And Annotated Bibliography. HERZOG, B.; CARTWRIGHT, K.; JOHNSON, T.; et al. Illinois, State of. March 1982. 254pp. 8204020083. 12493:157.

Failure of current trench covers in the humid eastern portion of the United States to prevent tritium and other radionuclide migration has prompted research into methods of reducing infiltration by modified trench covers. This report includes a discussion of present practices and a review of the relavant scientific literature. The main types of covers suggested are thicker clay covers of the current design, rigid and nonrigid man-made covers, covers employing clay sealants and covers using the wick effect. The two latter concepts are argued to offer the best long-term solution to the problem with the least maintenance. Optimal designs for clay and the wick effect have been proposed by SCS (1978) and Dragonette et al. (1979).

NUREG/CR-2479: FRACTURE DEFORMATION OF THE HIGGANUM DIKE, SOUTH-CENTRAL CONNECTICUT. SAWYER, J.; CARROLL, S. E. Boston College. January 1982.

55pp. 8202010005. 11763:336.

A study was made of fracturing of the Higganum dike in order to interpret the orientation of crustal stresses that have been dominant in south-central Connecticut in recent geologic time and may be responsible for seismic activity concentrated in the vicinity of the town of Moodus. The resultant stress field is oriented with a principal horizontal axis of compression aligned in the north-northwest-south-southeast direction and is consistent with the stress field held responsible by Block and others (1979) for recent overthrusting along the Honey Hill fault zone. The stress field may have caused minor displacement of the Higganum dike along faults cutting older adjacent Paleozoic gneiss, but it is likely that most of these faults originated prior to intrusion of the Higganum dike. The role of the stress field is thus depicted as one that could cause minor slippage along preexisting zones of weakness in the region.

NUREG/CR-2480 V01: EFFECT OF OXIDIZING ENVIRONMENT ON THE STRENGTH AND OXIDATION KINETICS OF HTGR GRAPHITES. ETO,M.; GROWCOCK,F.B. Brookhaven National Laboratory. March 1982. 75pp. 8204150567. BNL-NUREG-51493. 12691:266.

The primary focus of this report is a study of the strength of several HTGR graphites oxidized in various atmospheres. Understanding the role played by the oxidant was a major goal of this investigation; toward this end, a limited study of the kinetics of these oxidation reactions was also undertaken.

The effects of oxidizing atmosphere and temperature on the reactivities and strengths of PGX, H451 and IG-11 were examined. Preliminary measurements of the oxidation kinetics of these graphites in H(2)O-, CO(2)- and O(2)-containing atmospheres indicated that the reactivities of H451 graphite toward O(2) and H(2)O are quite similar to those of IG-11 graphite. The apparent activation energy for oxidation of these in O(2) were estimated to be ~175 kJ/mol while that in H(2)O is probably ~200 kJ/mol. The apparent activation energy of IG-11 graphite oxidized in CO(2) is 255 plus or minus 18 kJ/mol. PGX graphite was found to be quite variable in its reactivity toward H(2)O. A linear dependence with [Fe] was determined, but other intrinsic properties were found to affect its absolute reactivity by as much as a factor of X50. Explanations were advanced for observed anomalies in flow rate and burnoff dependence of the PGX graphite reaction rate in H(2)O. Apparent activation energies for PGX graphite oxidation were estimated to be E(a) (O(2)) ~165 kJ/mol; E(a) (H(2)O/H(2)) = 85 -250 kJ/mol, with a median value of 240 kJ/mol; and E(a) (CO(2)) ~229 plus or minus 12 kJ/mol.

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-2482 V01: REVIEW OF DOE WASTE PACKAGE PROGRAM. Subtask 1.1 - National Waste Package Program. DAVIS, M.S.; SCHEWEITZER, D. Brookhaven National Laboratory. February 1982. 57pp. 8203190006. BNL-NUREG-51494. 12351:187.
This report continues the assessment of the National Waste Package Program. Past reviews of waste forms, container materials and backfill materials are updated, with emphasis on the potential of these materials to meet NRC performance objectives. A start is made in describing information that NRC will need to review the waste package portion of a license application.

DOE waste form programs will emphasize borosilicate glass and SYNROC.

The data that NRC would need to make a positive finding for a waste package in which borosilicate glass is given full or partial credit for containment or controlled release of radionuclides from a borosilicate glass waste form would depend on the rate at which long-lived actinides are leached from the waste form. Almost all the existing data deals with leaching of short-lived fission products. These will be essentially innocuous after 1000 years. Preliminary experiments on waste glass loaded with actinides indicate new and complex problems that have not been studied yet. For example, actinides exist in glass in various oxidation states which may have different leach properties. NRC's information needs to support 1000 year containment by TiCode-12 are summarized.

NUREG/CR-2483: EVALUATION OF SIMULTANEOUS TESTING PROCEDURES FOR NUCLEAR MATERIALS CONTROL AND ACCOUNTING. BOWEN, W.M. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1982. 110pp. 8204150554. PNL-4085. 12707:257.

This report presents the results of a comparative evaluation of four statistical testing procedures for use in the control and accounting of special nuclear materials. Of primary interest is a bivariate procedure that simultaneously tests ID and CID. Descriptions of the four testing procedures are presented with the necessary formulas and special considerations for their implementation. Results of a simulation study indicate the conditions under which each of the tests would provide superior protection against "trickle" diversions.

NUREG/CR-2484: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION. JACKSON, P.L.; CHRISTENSEN, D.; MAUK, F.H. Michigan, Univ. of. January 1982. 100pp. 8202030224. 11806:016.

The geophysical investigations of the Western Ohio and Indiana regions include the maintenance of a precision seismograph network to monitor earthquake activity. Data generated by this network, supplemented with other information, are used to analyze regional seismicity and to interpret the local geologic and seismotechtonic structure. Four array stations in Indiana were added to the nine stations near Anna, Ohio, and the Ohio stations were upgraded by replacing seismometers and installing more stable electronic systems. A new digital computer with analog-to-digital convertors was obtained for direct digital recording and other digital analysis. Ten small earthquakes in the Anna region were recorded, six in a very tight cluster near the village of Anna. Only one of the earthquakes was felt. P-wave studies show an azimuthally dependent difference in residuals between stations within the Western Ohio array--a difference which can only be caused by significant variation in local structure. Aftershocks were recorded, public responses obtained and classified, and intensity contours drawn for the 5.1 magnitude earthquake on July 27, 1980, in Sharpsburg, Kentucky.

NUREG/CR-2485: ULTRASONIC BEAM SPREAD MEASUREMENTS IN THICK PRESSURE VESSEL TYPE STEEL. COOK,K.V.; MCCLUNG,R.W.; LATIMER,P.J.; et al. Oak Ridge National Laboratory. March 1982. 51pp. 8204150573. ORNL/TM-8159. 12708:051.

Ultrasonic beam spread measurements were made in pressure vessel type steel using the techniques outlined by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. In addition, a modified technique was developed that enabled beam spread measurements to be made in both the lateral and forward (analogous to the vertical beam spread addressed in the Code) direction. These data were analyzed under a variety of different conditions and the results of that analysis indicated that in many cases there exists a linear relationship between the beam width at half maximum amplitude and the beam path in metal. The results of this study have led to definite recommendations for scanning overlap and flaw sizing techniques as well as possibly providing a standard of reference for monitoring critical changes in the inspection systems between consecutive inspections.

NUREG/CR-2486: FINAL RESULTS OF THE HYDROGEN IGNITER EXPERIMENTAL PROGRAM. LOWRY,W.E.; BOWMAN,B.R.; DAVIS,B.W. Lawrence Livermore Laboratory. February 1982. 60pp. 8203040161. UCRL-53036. 12126:228.

Thermal igniters proposed by the Tennessee Valley Authority for intentional ignition of hydrogen in nuclear reactor containments have been tested in mixtures of air, hydrogen, and steam. The igniters, conventional diesel engine glow plugs, were tested in a 10.6 ft.(3) pressure vessel with dry hydrogen concentrations from 4% to 29% hydrogen, and in steam fractions of up to 50%. Dry tests indicated complete combustion between 8% and 9% H(2), and no combustion for concentrations below 5%. Steam tests were done with hydrogen volume fractions of 8%, 10%, and 12%. Steam concentrations of up to 30% consistently resulted in ignition. Most of the 40% steam fraction tests resulted in combustion. In a few isolated cases the 50% steam fraction tests indicated a pressure rise. Circulation of the mixture improved combustion in both the dry and the steam tests, most notably at low H(2) concentrations. An analysis of the high steam fraction test data showed a high probability for the presence of small, suspended water droplets in the test mixture. The suppressive influence of this condensation-generated fog on combustion is evaluated.

NUREG/CR-2487: EVALUATION OF IAEA COORDINATED PROGRAM STEELS AND WELDS FOR 288 C RADIATION EMBRITTLEMENT RESISTANCE. HAWTHORNE, J. R. Navy, Dept. of, Naval Research Laboratory. February 1982. 44pp. 8203090310. NRL MEMO RPT 46. 12176:091.

Eight steel materials supplied by the Federal Republic of Germany, France and Japan to the International Atomic Energy Agency (IAEA) Program on "Analysis of the Behavior of Advanced Reactor Pressure Vessel Steels Under Neutron Irradiation" were irradiated at 288 degrees C for assessments of relative notch ductility and dynamic fracture toughness change with approximately 2 x 10(19) n/cm(2), E > MeV. Notch ductility and fracture toughness were determined, respectively, by Charpy-V and fatigue precracked Charpy-V test methods. An A533-B steel plate (HSST 03) produced in the USA was included in the irradiation test series for reference.

The materials (plate, weld, forging) were found to be generally more resistant to radiation-induced embrittlement than the reference material. Observed dissimilarities in radiation sensitivity are attributed to copper content differences between the eight materials (0.01 to 0.07 percent copper range) and the reference plate (0.12 percent copper). Radiation resistances, however, corresponded well with the trend of radiation resistance reported for USA-produced steels and welds having similar copper and phosphorus contents.

A general correlation of transition temperature elevations measured independently by the Charpy-V and the precracked Charpy-V test methods was observed.

NUREG/CR-2489: EVALUATION OF THE BUCKLING STRESS CRITERIA FOR STEEL CONTAINMENT OF THE WATTS BAR NUCLEAR REACTOR. Docket Nos. 50-390 And 50-391. SEIDE, P.; WEINGARTEN, V.; MASRI, S. International Structural Engineers, Inc. December 1982. 217pp. 8301100031. 16719:001.

The report reviews the analysis of buckling strength for Watts Bar steel containment shell performed by the Tennessee Valley Authority. The analysis estimates pressure and earthquake loadings for the shell. Total combined load is compared with buckling strength of the shell to establish a margin of safety. The report recommends several improvements on the analysis.

NUREG/CR-2490: HAZARDS TO NUCLEAR POWER PLANTS FROM LARGE LIQUEFIED NATURAL GAS (LNG) SPILLS ON WATER. KOT, C. A.; EICHLER, T. V.; WIEDERMANN, A. H.; et al. Argonne National Laboratory. January 1982. 188pp. 8201270092. ANL-CT-81-17. 11712:309.

The hazards to nuclear power plants arising from large spills of liquefied natural gas (LNG) on water transportion routes are treated by deterministic analytical procedures. Global models, which address the salient features of the LNG spill phenomena are used in the analysis. A coupled computational model for the combined LNG spill, spreading, and fire scenario is developed. To predict the air blast environment in the vicinity of vapor clouds with "pancake-like" geometries, a scalable procedure using both analytical methods and hydrocode calculations is synthesized. Simple response criteria from the fire and weapons effects literature are used to characterize the susceptibility of safety-related power plant systems. The vulnerability of these systems is established either by direct comparison betweeen the LNG threat and the susceptibility criteria or through simple response calculations.

NUREG/CR-2492: SPECIAL NUCLEAR MATERIAL SELF-PROTECTION CRITERIA INVESTIGATION Phases I and II. KOELLING, J. J.; BARTS, E. W. Los Alamos Scientific Laboratory. January 1982. 39pp. 8202160078. LA-9213-MS. 11945:068.

The report was prepared to assist the NRC in reviewing the technical basis for the 100 rem per hour self-protection exemption from physical security requirements as allowed in 10CFR73.6. The report analyzes the risks and difficulty of stealing fuel from a non-power reactor core, the feasibility of detecting the theft of irradiated fuel and the radiation dosages necessary to incapacitate a potential thief. Alternatives to the current 100 rem per hour exemption are proposed and evaluated.

NUREG/CR-2493: AQUEOUS IODINE CHEMISTRY IN LWR ACCIDENTS: Review And Assessment. BELL, J. T.; PALMER, D. 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. 303p. 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 ACCIDENTS: 1969 -1979. A Status Report. Vol. 2 - Appendix B. MINARICK, J. W. ; KUKIELKA, C. A. Science Applications, Inc. June 1982. 1pp. 8207220678. ORNL/NSIC-182. 14021:033.

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-2498: DETERMINATION OF SCALE EFFECT ON SUBCOOLED CRITICAL FLOW. JARRELL, D. B. ; HALL, D. G. EG&G, Inc. February 1982. 41pp. 8203180436. EGG-2127. 12341:172.

This study was conducted to determine the effect of scale on subcooled critical flow mass flux through geometrically similar nozzles. A data base of subcooled critical flow data was compiled and analyzed using the modified Burnell critical flow model. The ratio of the measured to calculated flow rate was correlated with stagnation subcooling to produce a model correction parameter function which is independent of supply pressure. A comparison of the corrective parameter functions associated with each nozzle demonstrates that for subcooled supply conditions and for specific geometry studied, there is no resolvable size effect on critical flow for nozzles with throat diameters in the range of 3 to 300 mm.

NUREG/CR-2500: CRITICAL EXPERIMENTS ON LOW-ENRICHED URANIUM OXIDE SYSTEMS WITH H/U=2.03. Reference Critical Experiments. ROTHE, R. E. ; GOEBEL, G. R. Rockwell International Corp. February 1982. 62pp. 8203030243. RFP-3277. 12119:213.

Seven critical experiments were performed on a horizontal split table machine using 4.48 percent enriched 235U uranium oxide [U(3)O(8)]. The oxide was compacted to a density of 4.68 g/cm(3) and placed in 152-mm cubical aluminum cans. Water was added to achieve an H/U atomic ratio of 2.03. Various arrays of oxide cans were distributed on each half of the split table and the separation between halves reduced until criticality occurred. The critical table separation varied from 4.3 mm to 29.3 mm. These experiments were performed in both plastic and concrete reflectors.

The first five experiments required the addition of a high-enriched (approximately 93% 235U) metal driver to achieve criticality. Critical uranium driver masses ranged from 2.765 kg to 13.730 kg for 5 x 5 x 5 arrays of uranium oxide cans. In all five cases, the center can of the array was deleted to accommodate the driver. The uranium oxide mass was 1859.6 kg. Two additional experiments in the plastic reflector contained either 9.3-mm- or 24.3-mm-thick plastic moderator material between the oxide cans. These later experiments did not require a driver to achieve criticality; and the uranium oxide mass was 723.9 kg for the configuration having the thinner interstitial moderator and 452.4 kg for the other.

NUREG/CR-2501: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A NATURAL CIRCULATION TESTS S-NC-5 AND S-NC-6. O'CONNELL, T. M. EG&G, Inc. February 1982. 52pp. 8204290439. EGG-2162. 12901:351.

This report presents recorded test data for Tests S-NC-5 and S-NC-6 of the Semiscale Mod-2A Natural Circulation Test Series (Series NC). This series of test investigates the thermal-hydraulic phenomena resulting from operational transients or small break loss-of-coolant accidents (LOCAs) involving the loss of mechanical primary coolant circulation in a pressurized water reactor (PWR) system. These tests produce experimental data to develop and assess the analytic capability of computer models used to predict the results of small break LOCAs and operational transients involving the loss of primary coolant pumping capability.

This report presents the uninterpreted data from Tests S-NC-5 and S-NC-6 for future data analysis. The data, presented in the form of graphs in engineering units, have been analyzed only to the extent necessary to ensure that they are reasonable and consistent.

NUREG/CR-2502: USERS GUIDE AND DOCUMENTATION FOR ADSORPTION AND DECAY MODIFICATIONS TO THE USGS SOLUTE TRANSPORT MODEL. TRACY, J. V. Ertec, Inc. January 1982. 80pp. 8203010412. 12075:275.

Development and user documentation for the incorporation of radioactive decay and equilibrium adsorption into the MOC solute transport model of Konikow and Bredehoeft (1974) is provided. The models of radioactive decay, linear isotherm, Langmuir isotherm, and the Freundlich isotherm, are developed as well as the basic flow and conservative transport equations for a ground water system. The report includes the theoretical developments, numerical implementations, data input requirements, a listing of the computer code that is written to comply with Fortran 77 standards, sample input and sample output, and a Bibliography of selected publications on various aspects of the transport problems.

NUREG/CR-2504: CLEAR (CALCULATES LOGICAL EVACUATION AND RESPONSE): A GENERIC TRANSPORTATION NETWORK MODEL FOR THE CALCULATION OF EVACUATION TIME ESTIMATES. MOELLER, M. P.; URBANIK, T.; DESROSIERS, A. E. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1982. 104pp. 8203290058. PNL-3770. 12442:157.

This paper describes the methodology and application of the computer model CLEAR (Calculates Logical Evacuation and Response) which estimates the time required for a specific population density and distribution to evacuate an area using a specific transportation network. The CLEAR model simulated vehicle departure and movement on a transportation network according to the conditions and consequences of traffic flow. These include handling vehicles at intersecting road segments, calculating the velocity of travel on a road segment as a function of its vehicle density, and accounting for the delay of vehicles in traffic queues. The program also models the distribution of times required by individuals to prepare for an evacuation. In order to test its accuracy, the CLEAR model was used to estimate evacuation times for the emergency planning zone surrounding the Beaver Valley Nuclear Power Plant. The Beaver Valley site was selected because evacuation times estimates has previously been prepared by the licensee, Duquesne Light, as well as by the Federal Emergency

Management Agency and the Pennsylvania Emergency Management Agency. A lack of documentation prevented a detailed comparison of the estimated based on the CLEAR model and those obtained by Duquesne Light. However, the CLEAR model results compared favorably with the estimates prepared by the other two agencies.

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-2507: BACKGROUND AND DERIVATION OF ANS-5.4 STANDARD FISSION PRODUCT RELEASE MODEL. TURNER, S.E. Black & Veatch. January 1982. 180pp. 8203010030. 12072:154.

This report summarizes work performed by the ANS-5.4 Working Group on Fuel-Plenum Fission Gas Inventory and is a compilation of individual contributions by members of the Working Group. The report was compiled to document the basis for the ANSI/ANS Standard on "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuels," ANSI/ANS 5.4-1981. The information contained in this report is important to an understanding of the Standard, and has been reviewed and approved by the Working Group.

NUREG/CR-2509: MATERIALS TEST-2 LOCA SIMULATION IN THE NRU REACTOR. BARNER, J.O.; HESSON, G.M.; KING, L.L.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1982. 75pp. 8204010556. PNL-4155. 12491:001.

A simulated loss-of-coolant accident was performed with a full-length test bundle of pressurized water reactor fuel rods. This third experiment of the program produced fuel cladding temperatures exceeding 1033 K (1400 degrees F) for 155 s and resulted in eight ruptured fuel rods. Experiment data and initial results are presented in the form of photographs and graphical summaries.

NUREG/CR-2511: STATUS OF KNOWLEDGE OF RADIATION EMBRITTLEMENT IN USA REACTOR PRESSURE VESSEL STEELS. HAWTHORNE, J. R. Navy, Dept. of, Naval Research Laboratory. February 1982. 35pp. 8203090215. NRL MEMO RPT 47. 12176:138.

Advances by experimental research in the USA toward an improved understanding of property changes in steel by elevated temperature (approximately 288 degrees C) irradiation are summarized. Four areas of investigation are reviewed including the confirmation and demonstration of guidelines for radiation resistant steels, the isolation of metallurgical factors contributing to variable radiation embrittlement sensitivity, the qualification of the in situ heat treatments for periodic vessel embrittlement relief, and the correlation of notch ductility and fracture toughness changes with irradiation.

Overall, the current state of the art provides both a high capability for tailoring steels for radiation service in new vessel construction and a promising method for controlling radiation embrittlement buildup in existing pressure vessel construction.

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 excretion 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-2514: ULTIMATE HEAT SINK THERMAL PERFORMANCE AND WATER UTILIZATION MEASUREMENTS ON COOLING AND SPRAY PONDS. ATHEY, G. F.; HADLOCK, R. K.; ABBEY, O. B. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1982. 185pp. 8203030203. PNL-4159. 12111:111.

A data acquisition research program, entitled "Ultimate Heat Sink Performance Field Experiments," has been completed. The primary objective was to obtain the requisite data to characterize thermal performance and water utilization for cooling ponds and spray ponds at elevated temperatures. Such data are useful for modeling purposes; such modeling efforts were beyond the scope of the present study.

Constructed water retention ponds, provided with absolute seals against seepage, have been chosen as facilities for the data collection program; the first pond was located at Raft River, Idaho, and the second at East Mesa, California. The data illustrate and describe, for both cooling ponds and spray ponds, thermal performance and water utilization as the ponds cool from an initially elevated temperature, nominally 130 degrees F. The data reflect thermal performance and water utilization for meteorological and solar influences which are representative of worst-case combinations of conditions. The data are described and discussed in the text and presented in the form of data volumes as appendices.

NUREG/CR-2515 V01: CRYSTAL RIVER-3 SAFETY STUDY. Volume 1-Main Study. GARCIA, A. A.; LINER, R. T.; AMICO, P. J.; et al. Science Applications, Inc. March 1982. 176pp. 8204160057. SAND81-7229/1. 12715:202.

This report provides a quantitative assessment of certain aspects of the public risk associated with operation of the Crystal River-3 (CR-3) nuclear power plant. The plant uses an 855 MWe pressurized water reactor whose nuclear steam supply system was manufactured by the Babcock and Wilcox (B&W) Company. The assessment includes estimates of the frequency (or probability per year) of radioactivity releases, in each of seven discrete categories, stemming from loss of coolant accidents (LOCAs) and various anticipated transients. The primary objective is to identify and estimate the probabilities of those types of accidents which are most likely to cause releases in these seven categories, and to identify and quantify the combinations of hardware and human faults which contribute most to these probabilities. The report includes a review of plant-specific failure rate data and incorporation of it into the analysis where appropriate; a review of reliability data for pumps, valves and diesels to obtain quantitative information on the probabilities of common mode failures; and a detailed analysis of important operator faults using the methods known as Technique for Human Error Rate Prediction.

NUREG/CR-2515 V02: CRYSTAL RIVER-3 SAFETY STUDY. Volume 2-Appendices. GARCIA, A. A.; LINER, R. T.; AMICO, P. J.; et al. Science Applications, Inc. March 1982. 300pp. 8204010534. SAND81-7229/2. 12488:001.

This report provides a quantitative assessment of certain aspects of the public risk associated with operation of the Crystal River-3 nuclear power plant. The plant uses an 855 MWe pressurized water reactor whose nuclear steam supply system was manufactured by the Babcock and Wilcox (B&W) Company. The assessment includes estimates of the frequency (or probability per year) of radioactivity releases, in each of seven discrete categories, stemming from loss of coolant accidents (LOCAs) and various anticipated transients. The primary objective is to identify and estimate the probabilities of those types of accidents which are most likely to cause releases in these seven categories, and to identify and quantify the combinations of hardware and human faults which contributed most to these probabilities. The report includes a review of plant-specific failure rate data and incorporation of it into the analysis where appropriate; a review of reliability data for pumps, valves and diesels to obtain quantitative information of the probabilities of common mode failures; and a detailed analysis of important operator faults using the method known as Technique for Human Error Rate Prediction.

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-2516 V01 N4: CHARACTERIZATION OF TMI-TYPE WASTES AND SOLID PRODUCTS. Quarterly Progress Report, October-December 1981. SWYLER, K. J. ; DAYAL, R. Brookhaven National Laboratory. August 1982. 44pp. 8209280095. BNL-NUREG-51499. 15545:198.

A research program is under way 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 towards 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-2517: EFFECT OF RADIOLYTIC GAS ON NUCLEAR EXCURSIONS IN AQUEOUS SOLUTIONS. FOREHAND, H. M. Arizona, Univ. of. January 1982. 102pp. 8203020084. 12090:210.

Although not intended to study the evolution of a solution nuclear criticality accident, the Kinetic Experiment on Water Boiler (KEWB) demonstrated the dependence of the nuclear excursion on parameters such as solution temperature and radiolytic gas. Similarly, the CRAC ("Consequence Radiologigues d'un Accident de Criticite") program results indicate the excursion was governed by parameters such as the solution addition rate, initial neutron population, solute concentration, and thermal and radiolytic gas feedback.

The majority of the energy deposited in a fissile solution is by the fission fragments. Energy deposition causes an increase in temperature decreasing density and hence neutron leakage. A second feature is the decomposition of water molecules into H(2) and O(2) in the solution.

Microbubbles are nucleated in the fissile solution by a localized thermal spike generated by a fission fragment. In a supersaturated solution the bubble will grow and produce negative feedback by increasing neutron leakage.

Both an energy and a pressure model have been incorporated into a space-independent kinetic computer code, MACKIN, while the pressure model was also incorporated into a space-dependent code, AZPAD. The models have been successful in predicting the peak power, burst energy, and maximum system pressure for the first burst in both KEWB and CRAC experiments.

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-2520: A HYDRODYNAMIC ANALYSIS OF THE ELECTRON BEAM VAPORIZATION EXPERIMENTS FOR URANIUM DIOXIDE. October 1, 1979-September 31, 1981. GANAPOL, B. D.; CLARK, B. A.; SMITH, M. S.; et al. Arizona, Univ. of. June 1982. 64pp. 8208090016. 14303:047.

The electron-beam experiments designed to generate high temperature vapor pressure data for UO(2) vapor is analyzed by three hydrodynamic codes SIMMER, FARA, and VIOLET. The physical and numerical modeling in each code is different, thus providing a modeling error associated with the analysis. Agreement with the experiment is rather poor indicating that the rate dependent specific heat capacity model proposed by Benson may not be a physical reality.

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. 154pp. 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-2523: DERIVATION OF DOSE CONVERSION FACTORS FOR TRITIUM. KILLOUGH, G.G. Oak Ridge National Laboratory. August 1982. 28pp. 8209270115. ORNL-5853. 15518:262.

For a given intake mode (ingestion, inhalation, absorption through the skin), a dose conversion factor (DCF) is the committed dose equivalent to a specified organ of an individual per unit intake of a radionuclide. One also may consider the effective dose commitment per unit intake, which is a weighted average of organ-specific DCFs, with weights proportional to risks associated with stochastic radiation-induced fatal health effects, as defined by Publication 26 of the International Commission of Radiological Protection (ICRP). This report defines and tabulates organ-specific dose conversion factors and the effective dose commitment per unit intake of tritium. These factors are based on a steady-state model of hydrogen in the tissues of ICRP's Reference Man (ICRP Publication 23) and equilibrium of specific activities between body water and other tissues. The results differ by 27-33% from the estimate on which ICRP Publication 30 recommendations are based. The report also examines a dynamic model of tritium retention in body water, mineral bone, and two compartments representing organically-bound hydrogen. This model is compared with data from human subjects who were observed for extended periods.

The manner of combining the dose conversion factors with measured or model-predicted levels of contamination in man's exposure media (air, drinking water, soil moisture) to estimate dose rate to an individual is briefly discussed.

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 V04: ORNL ROD BUNDLE HEAT TRANSFER TEST DATA. Volume 4. ORNL Small Break LOCA Heat Transfer Test Series II: Experimental Data Report. ANKLAM, T. M.; HUNT, D. F.; FELDE, D. K.; et al. Oak Ridge National Laboratory. July 1982. 133pp. 8208260087. ORNL/NUREG/TM-4. 14594:213.

This report presents experimental data and calculated steady-state and transient instrument uncertainties from the Oak Ridge National Laboratory Small Break LOCA Heat Transfer Test Series II. The subject test series was composed of six combined heat transfer and mixture level swell tests, six additional mixture level swell tests, five high-pressure reflood tests, and five high-pressure boiloff tests. Also, the data and uncertainties are reported from two supplemental mixture level swell tests that were not part of Test Series II. Calculated inlet and outlet mass flows and fuel rod simulator power levels are reported in the report appendices.

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. 257pp. 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 V06: ORNL ROD BUNDLE HEAT TRANSFER TEST DATA. Volume 6. Thermal-Hydraulic Test Facility Experimental Data Report For Test 3.05.5B-Double-Ended Cold-Leg Break Simulation. MULLINS, C. B. ; FELDE, D. K. ; SUTTON, A. G. ; et al. Oak Ridge National Laboratory. July 1982. 234pp. 8209270427. ORNL/NUREG/TM-4. 15531:332.

Thermal-Hydraulic Test Facility (THTF) Test 3.05.5B was conducted by members of the Oak Ridge National Laboratory (ORNL) Pressurized-Water Reactor (PWR) Blowdown Heat Transfer (BDHT) Separate-Effects Program on July 3, 1980. The objective of the program is to investigate heat transfer phenomena believed to occur in PWRs during accidents, including small and large break loss-of-coolant accidents.

Test 3.05.5B was designed to provide transient thermal-hydraulics data in rod bundle geometry under reactor accident-type conditions. Reduced instrument responses are presented. Also included are uncertainties in the instrument reponses, 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. 295pp. 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-2526: LOCA SIMULATION IN NRU PROGRAM. Data Report For Thermal-Hydraulic Experiment 2 (TH-2). MOHR, C. L. ; HESSON, G. M. ; KING, L. L. ; et al. Battelle Memorial Institute, Pacific Northwest

Laboratory. November 1982. 158pp. 8212220265. PNL-4164. 16526:188.

A series of thermal-hydraulic and cladding material experiments are being conducted using LWR fuel as part of the PNL LOCA Simulation Program. Experimental data and initial results from the fourth experiment in the program--thermal-hydraulic experiment 2 (TH-2)--are presented in this report. The program is being conducted in the NRU reactor, Chalk River, Canada. A full-length test bundle containing 12 test rods and 20 guard rods (all nonpressurized) was used to develop reflood control parameters and procedures that will produce a reduced heatup rate or a "flat top" transient for extended periods of time. Variable reflood rates were used, and experimentally determined control system logic parameters were developed. Using these concepts, cladding temperatures from 1033 to 1274 K were produced for 283 seconds.

NUREG/CR-2529: AN INTEGRATED REGIONAL APPROACH TO ENERGY FACILITY SITING. KINCANNON,B.F.; RUTZ,W.L. New England Regional Commission. March 1982. 248pp. 8204070089. 12595:072.

This report summarizes the experience of the six New England States working together on a regional basis to develop a prototype modeling system suitable for assessing environmental and economic issues related to energy facility siting. The modeling system was composed of existing models in three categories; electricity demand forecasting, utility systems planning, and siting/environmental. A number of possible candidate models were used to analyze key energy facility siting issues in New England. The design of the prototype modeling systems is covered in detail dealing with major issues such as model compatibility and linkage, data availability and operational utility of the system in the analysis of actual policy issues.

NUREG/CR-2531: INTRODUCTORY USER'S MANUAL FOR THE U.S. NUCLEAR REGULATORY COMMISSION REACTOR SAFETY RESEARCH DATA BANK. SCOFIELD,N.R.; HARDY,H.A.; LAATS,E.T. EG&G, Inc. February 1982. 103pp. 8203190070. EGG-2164. 12366:240.

The United States Nuclear Regulatory Commission (NRC) has established the NRC/Division of Accident Evaluation (DAE) Data Bank Program to collect, store, and make available data from the many domestic and foreign water reactor safety research programs. Local direction of the program is provided by EG&G Idaho, Inc., prime contractor for the department of Energy (DOE) at the Idaho National Engineering Laboratory.

The NRC/DAE Data Bank program provides a central computer storage mechanism and access software for data that is to be used by code development and assessment groups in meeting the code and correlation needs of the nuclear industry. The administrative portion of the program provides data entry, documentation, training, and advisory services to users and the NRC. The NRC/DAE Data Bank and the capabilities of the data access software are described in this document.

NUREG/CR-2532: CATALOG OF EARTHQUAKES FELT IN THE EASTERN U.S. MEGALOPOLIS 1850-1930. WINKER,L. Pennsylvania State Univ. July 1982. 32pp. 8207290407. 14125:315.

The area covered by this catalog is the region east of the Appalachians from the southern border of Virginia to the southeast tip of New York state. For the period 1850-1930, a total of 192 earthquakes were felt in the area. Of these earthquakes 39 were damaging with intensities of V-VI to IX. The vast majority of the historical sources used to construct the description were local newspapers.

NUREG/CR-2533: THE EFFECT OF STOCHASTIC VARIATION ON ESTIMATES OF THE PROBABILITY OF ENTRAINMENT MORTALITY: Methodology, Results, And User's Guide. CHRISTENSEN, S.; DEANGELIS, D.L. Oak Ridge National Laboratory. July 1982. 84pp. 8208260473. ORNL/TM-7965. 14570:310.

The probability that live fish eggs or larvae, entrained in cooling water, will be killed is an important element in projecting power plant effects on fish stocks. This probability, the entrainment mortality factor, is commonly estimated with one of several relatively simple formulae which use data collected from intake and discharge water. A Monte Carlo simulation model (ENTRAN) was developed to assess the accuracy and precision of entrainment mortality estimates derived via these formulae from field data. After repeated simulation over a range of selected entrainment mortalities for varying biological conditions, the reliability of mortality estimates was evaluated by comparing the estimates with the actual mortality that was selected for the model run.

NUREG/CR-2534: CRITERIA FOR SAFETY-RELATED NUCLEAR POWER PLANT OPERATOR ACTIONS: INITIAL BOILING WATER REACTOR (BWR) SIMULATED EXERCISES. BEARE, A.N.; CROWE, D.S.; KOZINSKY, E.J.; et al. General Physics Corp. November 1982. 110pp. 8301120098. ORNL/TM-8195. 16780:084.

The primary objective of the Safety-Related Operator Action Program at Oak Ridge National Laboratory is to provide a data base to support development of criteria for safety-related action by nuclear power plant operators. This report presents initial data obtained from ten exercises conducted in a boiling water reactor power plant control room simulator. The ten exercises were performed by 24 groups of operators from three utilities. Operator performance was recorded automatically by a program call the Performance Measurement System run on the simulator's computer. Data tapes were subsequently analyzed to extract operator response time (RT) and error rate information. In addition, demographic and subjective data were collected and analyzed in an attempt to identify and evaluate the possible effects of selected performance-shaping factors on operator performance. Operator RTs to the simulated events generally occurred within the intervals allowed in the draft ANSI-N660 design standard; however, they did not appear to be systematically related to the severity of the event, which was the basis for allocation of time margins in the standard. More collective experience in power plant operations was weakly correlated with faster responses. Limited data on omission errors yielded an error rate of greater than five percent.

NUREG/CR-2535: COMMIX-IA THREE-DIMENSIONAL IN-VESSEL SIMULATION OF THE FFTF THERMAL HYDRAULICS. VANKA, S.P.; DOMANUS, H.M.; SHA, W.T. Argonne National Laboratory. March 1982. 80pp. 8203290052. ANL-CT-82-1. 12442:262.

Three-dimensional in-vessel simulation of the FFTF thermal hydraulics has been performed with the COMMIX-1A computer code. The COMMIX-1A code employs the porous media formulation in which the concept of volume porosity, surface permeability, distributed resistance, and distrubuted heat source is used to model the internal structures. The governing equations of conservation of mass, momentum, and energy are solved as a boundary-value problem in space and as an initial-value problem in time. In the present report the calculated results for the steady-state reactor full-power operation are presented, and are compared with experimental measurements, where applicable. Based on the COMMIX calculations, thermal stratification is observed in the upper plenum, together with eddy-type recirculation. Comparison of measured flow rates and temperatures at selected locations has been satisfactory.

NUREG/CR-2539: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS. Annual Progress Report, April 1980-March 1981. EDISON, A. F. Lovelace Biomed & Environmental Research Institute. February 1982. 49pp. 8203180458. 12341:015.

Comparisons of chemical properties and biological behavior of refined uranium ore (yellowcake) are made to identify important properties that influence uranium distribution among organs. These studies will facilitate calculations of organ doses for specific exposures, health risk estimates and will identify important bioassay procedures to improve evaluations of human exposures. Aerosols at four mills were heterogenious and contained approximately 50% of the airborne uranium in particles greater than 12 nanometers aerodynamic diameter. Appreciable amounts of uranium would deposit in the nasopharyngeal region of the respiratory tract if inhaled. Particle size distributions varied with time. An in vitro test to approximate yellowcale solubility in vivo is described. Infrared analyses showed that ammonium deuranate varied from 12% to 100% of yellowcake from five mills and from 12% to 67% of lots from one mill. The ammonium diuranate fraction could not be related to drying temperature reliably. Clearance patterns of inhaled uranium from rat lungs agreed with results from in vitro dissolution and infrared analyses. Results from an experiment to simulate wound contamination by yellowcake included deaths from kidney toxicity of 12 rats exposed to more soluble yellowcake. No rats exposed to less soluble yellowcake died. The caging used might have been a factor.

NUREG/CR-2540: A METHOD FOR THE ANALYSIS OF HYDROGEN AND STEAM RELEASES TO CONTAINMENT DURING DEGRADED CORE COOLING ACCIDENTS. CYBULSKIS, P. Battelle Memorial Institute, Columbus Laboratories. February 1982. 170pp. 8203040175. BMI-2090. 12125:001.

The Nuclear Regulatory Commission is considering requirements that reactor containments be able to accommodate without loss of containment integrity or degradation of vital equipment the large amounts of hydrogen that may be generated during severe degraded core cooling accidents. Conformance with the proposed requirements may entail the installation of hydrogen control system in certain containments. In order to assist with the implementation of these requirements, analyses have been performed to define steam and hydrogen release rates into PWR and BWR containments during representative severe degraded core cooling accidents. These envelopes of hydrogen and steam source terms to the containment can be used for performing containment responses analyses. This approach is intended to obviate the need for extensive case-by-case analyses of the progression of a variety of accident sequences and allow the attention to be focused on the containment response evaluations. The use of the hydrogen and steam release into containment developed in this study is one of several alternatives under consideration by the Nuclear Regulatory Commission.

NUREG/CR-2541: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS: CRITICAL ISSUES AND RECOMMENDED RESEARCH. SHACK, W. J.; NICHOLS, F. A.; KASSNER, T. F.; et al. Argonne National Laboratory. March 1982. 150pp. 8204010542. ANL-82-2. 12487:120.

This report addresses the problem of environmentally assisted cracking in light-water reactor components. The field experience is reviewed, current utility-industry effort to solve the problem is discussed, and the remaining unresolved technical issues are identified. The additional research and development needed by the NRC to develop an independent capability to evaluate current stress-corrosion cracking problems in light-water reactors and their potential solutions is assessed. The problems and research are divided into six general areas: (1) Leak Detection and Nondestructive Evaluation, (2) Analysis of Sensitization, (3) Analysis of Crack Growth, (4) Nonenvironmental Corrective Actions, (5) Environmental Corrective Actions, and (6) Basic Understanding of Stress-corrosion Cracking. It should be noted that this report was essentially completed in December 1980. Although additional information and research results continue to appear, the report still serves as a useful summary of the work up until the end of 1980 and outlines the critical areas that are the subject of current, ongoing research.

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. GINZBURG, 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 high-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 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-2547: EVALUATION OF GEOTECHNICAL SURVEILLANCE TECHNIQUES FOR MONITORING HIGH-LEVEL WASTE REPOSITORY PERFORMANCE. ST JOHN, C. M.; AGGSON, J. R.; HARDY, M. P.; et al. J. F. T. Agapito & Associates, Inc. March 1982. 400pp. 8204210614. 12796:130.

This report recommends geotechnical monitoring programs for evaluation of high-level nuclear waste repository performance. It is based upon assessments of conditions and responses of the geologic media to the site of a repository. The state of technology in geotechnical monitoring was reviewed for applicability to repository monitoring. The recommended geotechnical monitoring program utilizes existing geotechnical instrumentation.

The objectives of the proposed geotechnical monitoring program are to verify that the performance of the components of the repository conform with the design bases. Design verification requires detailed, quantitative response measurement. It is proposed that design verification be conducted in a special test facility constructed immediately after a particular site has been selected for a repository. Monitoring activities outside the test facility should be of a confirmatory nature only. Where repository geological conditions are encountered which are significantly different from the design verification facility (DVF), then more extensive monitoring instrumentation would be in order in those areas.

The proposed geotechnical monitoring activities are delineated in each phase of repository development. The proposed monitoring is detailed for various dimensional scales. Only geotechnical instrumentation presently available or easily within the state of the art is considered and it was concluded that it is possible to define an integrated geotechnical monitoring program that would meet the proposed licensing requirements.

NUREG/CR-2548: REGIONAL TECTONICS AND SEISMICITY OF SOUTHWESTERN IOWA. VAN ECK, O. J. Iowa Geological Survey. March 1982. 94pp. 8204260010. 12847:140.

Vertical faulting of Precambrian basement rock along the flanks of the Midcontinent Geophysical Anomaly (MGA) has been identified on the basis of gravity data. Detailed seismic profiling across the Thurman-Redfield structural zone, which parallels the southern flank of the MGA in Iowa, thus far has not detected faulting in Paleozoic rock associated with the structural zone. Modeling of both Bouguer gravity anomalies and total-field magnetic expr[illegible]on lead to the interpretation that the Precambrian basement in southwest Iowa includes a central horst of intrusives and extrusives extensively faulted, and overlain in some areas by Precambrian clastics, all flanked by deep clastic-filled basins. Microearthquake monitoring has detected numerous very low-magnitude local microearthquakes, especially in the Carbon, Union County vicinity. The source of these local events has not been determined.

NUREG/CR-2549: BACKGROUND STUDY AND PRELIMINARY PLANS FOR A PROGRAM ON THE SAFETY MARGINS OF CONTAINMENTS. BLEJWAS, T. E.; DENNIS, A. W.; WOODFIN, R. L.; et al. Sandia Laboratories. July 1982. 58pp. 8208260112. SAND82-0324. 14577:132.

This report describes a background study and preliminary plans for a program to develop methodology for predicting the capacity of containment structures under severe accident and environmental conditions. Both analytical and experimental efforts were considered in the background study. It is concluded that the end results of the program should be (1) Bench-mark data from scale-model tests on selected classes of containments, and (2) A set of qualified computer programs that can be used to determine the ultimate capacity of steel, reinforced concrete, and prestressed concrete containments subjected to internal pressurization and seismic loadings.

NUREG/CR-2550: CHARCOAL PERFORMANCE UNDER SIMULATED ACCIDENT CONDITIONS, Interim Report on Continuing NRL Problem. DIETZ,V.R. Navy,Dept.of, Naval Research Laboratory. July 1982. 48pp. 8208040200. 14231:321.

Regulatory Guide 1.52 stipulates a radiation level for iodine buildup on the adsorber of 10(9) rads as one of the typical accident conditions for atmospheric clean up systems. The laboratory research in progress seeks to study the combined effects of in-service weathering, exposure to atmosphere contaminants, and radiation doses on the retention of iodine by the carbon. A number of service and weathered carbons have been exposed to the lambda-radiation from the (60) Co source (approximately 1 MeV) and to the radiation from the NRL LINAC facility (approximately 45 MeV). Total radiation levels of 10(7), 10(8) and 10(9) rads were used and the carbons were evaluated before and after irradiation by the methyliodide-131 penetration test (30 degrees C, ASTM-3803-79). Surprising improvements were observed after static (i.e. no air flow) exposures of the carbons to the irradiation. Flow-through exposures during irradiation are now in progress using air or argon flows and with or without methyliodide-127.

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-2553: ETHYLENE PROPYLENE CABLE DEGRADATION DURING LOCA RESEARCH TESTS:TENSILE PROPERTIES AT THE COMPLETION OF ACCELERATED AGING. BUSTARD,L.D. Sandia Laboratories. July 1982. 29pp. 8209210442. SAND82-0346. 14949:005.

Six ethylene-propylene rubber (EPR) insulation materials were aged at elevated temperature and radiation stress exposures common in cable LOCA qualification tests. Material samples were subjected to various simultaneous and sequential aging simulations in preparation for accident environmental exposures. Tensile properties subsequent to the aging exposure sequences are reported. The tensile properties of some, but not all, specimens were sensitive to the order of radiation and elevated temperature stress exposure. Other specimens showed more severe degradation when simultaneously exposed to radiation and elevated temperature as opposed to the sequential exposure to the same stresses. Results illustrate the difficulty in defining a single test procedure for nuclear safety-related qualification of EPR elastomers. A common worst-case sequential aging sequence could not be identified.

NUREG/CR-2554: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A NATURAL CIRCULATION TEST S-NC-10. O'CONNELL,T.M. EG&G, Inc. March 1982. 67pp. 8203120002. EGG-2167. 12252:008.

This report presents recorded test data for Test S-NC-10 of the Semiscale Mod-2A Natural Circulation Test Series. Test S-NC-10 is part of a series of Semiscale tests that investigates the thermal-hydraulic phenomena resulting from operational transients or small break loss-of-coolant accidents (LOCAs) involving the loss of mechnical primary coolant circulation in a pressurized water reactor. These tests produce experimental data to develop and assess the analytic capability of computer models used to predict the results of such small-break LOCAs and operational transients. Test S-NC-10 was designed to supplement data obtained in earlier tests of the natural circulation series, such that more precise statements can be made about single-phase and two-phase natural circulation behavior. Test S-NC-10 was performed in four subsections: Part 1, Part 2, Part 3, and Part 4. This report presents the uninterpreted data from each of these parts for future analysis. The data, presented as graphs in engineering units, have been analyzed only to the extent necessary to ensure their validity as actual measurements of the system phenomena.

NUREG/CR-2555: DATA REQUIREMENTS FOR THE EVALUATION OF STORM SURGE MODELS. HARRIS,D.L. Florida, Univ. of, Gainesville. February 1982. 42pp. 8203040170. UFL/COEL-81-015. 12126:290.

Numerical storm surge prediction models are needed to support real time warnings of potential disaster and to support long term probability estimates of high water occurrence for engineering design, insurance rates and planning procedures for mitigating disasters.

Although the models are complex, they do not reproduce all or the complexities of prototype phenomena. Evaluation of the models by comparing model calculations to prototype measurements are needed to establish the proper level of creditability.

An examination of the model structure suggests that model calculations should provide reasonably good predictions of large scale features of the storm surge, but that local deviations from the large scale phenomena cannot be predicted in detail. An examination of the physical processes involved in storm surge generation and of records from past storm surges indicate the small scale phenomena of local extent, as well as phenomena of the same horizontal and time scales of the storm generally form an integral part of the storm surge.

The small scale effects, however, cannot be eliminated from many historical storm surge records. It is suggested that model evaluations would be most useful, if models were evaluated in terms of their ability to predict the large scale features of the phenomenon. This means that observation sites for model evaluation should be selected to minimize the small scale local effects.

NUREG/CR-2556: LIDAR VELOCITY MEASUREMENTS OF WATERSPOUTS AND AN ONSHORE WIND. SCHWIESOW,R.L. Commerce,Dept. of, National Oceanographic & Atmospheric Administration. February 1982. 42pp. 8203030211. 12112:235.

The focus of this project is remotely-sensed measurement of the wind fields in waterspout vortices. The research used infrared CW Doppler lidars with both airborne and groundbased platforms. The study was expanded to explore the change in wind profile caused by a change in terrain and roughness at a sea shore, and to investigate the applicability of the lidar sensing technique to the flow field in the wake of large structures. The peak velocity observed by the lidar in

21 waterspout data sets ranged from 4.2 mps to 33.6 mps and visible funnel diameters from 6.6 m to 90 m for data taken at altitude between 675 m and 95 m. A horizontal resolution of 0.75 m between data points across the funnel revealed a large azimuthal asymmetry and mixing with the ambient flow and some showing multiple concentric vortex shells. A change of terrain affects the wind profile upwind of the change by at least 200 m in the experiment. Downwind, the increased roughness reduces the mean velocity at low levels, increases turbulence, and causes acceleration, convergence, or both at higher levels. Studies of building wakes require a larger aerosol backscatter coefficient or an improved system compared to the feasibility test conducted.

NUREG/CR-2557: SIMULATED TORNADO WIND FIELDS AND DAMAGE PATTERNS. METCALF, D.R.; PETERSON, R.E. Texas Tech Univ. February 1982. 70pp. 8203040180. 12128:107.

The translational motion as well as the internal motions of the tornado combine to produce the forces experienced by a structure. The combined wind flow and damage patterns are investigated in a series of numerical simulations. The results have been studied and organized on the basis of characteristic properties. Analysis of the damage patterns generated by the simulated tornado wind fields are performed and compared with actual, observed tornado damage patterns. An attempt has been made to categorize tornado damage patterns so that they might be used in field investigations of tornado damage. Various hypothetical sequences of tornado development and behavior are proposed to simulate the resulting tornado damage patterns.

NUREG/CR-2558: CONTEMPT 4/MOD 3, A MULTICOMPARTMENT CONTAINMENT SYSTEMS ANALYSIS PROGRAM. CHENG, T.C.; METCALFE, L.J.; HARTMAN, J.E.; et al. EG&G, Inc. December 1982. 213pp. 8212220282. EGG-2159. 16525:053.

CONTEMPT4/MOD3 is a digital computer program, written in FORTRAN IV, that describes the behavior of multicompartment pressurized water reactor (PWR) containment systems and experimental containment systems subjected to postulated loss-of-coolant accidednt (LOCA) conditions. The program calculates the time variation of compartment pressures, temperatures, mass and energy inventories, heat structure temperature distributions, and intercompartment mass and energy exchange based on usersupplied values of compartment descriptions, time step and edit controls, and selected problem features. Analytical models available to describe containment systems include models for containment fans and pumps, cooling sprays, fan coolers, heat-conducting structures, sump drains, and PWR ice condensers. Dynamic storage allocation (DSA) is used to limit the amount of computer core used for each problem. Optional automatic time step control allows the code to determine time step sizes within limits dictated by the user. Multicompartment capability (up to 999 individual compartments) and generalized, user-oriented input data descriptions permit improved flexibility over previous codes in the CONTEMPT series. Analytical model descriptions, input instructions, and sample problem results are presented.

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-2560: QUALITY ASSURANCE FOR MEASUREMENTS OF IONIZING RADIATION. Annual Report for FY 1981. EISENHOWER, E. H.; ERLICH, M.; LOFTUS, T. P.; et al. Commerce, Dept. of. March 1982. 104pp. 8203290050. 12442:014.

This report describes results of the first year of a program that will enable the Nuclear Regulatory Commission to improve, demonstrate, and document traceability of its measurements to the national physical measurement standards for ionizing radiation. The principal actions being taken are: (1) characterization of the response of thermo-luminescence dosimetry systems used for routine surveillance of nuclear facilities; (2) type testing and characterization of portable survey instruments; and (3) implementation of routine quality assurance services which will demonstrate that regulatory measurements are sufficiently consistent (in agreement) with national measurement standards. During the year, some tests of the TLD system were performed as specified in American National Standard N545-1975, specifically uniformity, reproducibility, dependence of exposure interpretation on the length of the field cycle, energy dependence, light dependence, and moisture dependence. The energy dependence of five portable survey instruments was determined for exposure to photon radiation, and a chamber for exposure to gaseous sources of beta radiation was designed and partly built. The four laboratories which calibrate portable survey instruments for NRC inspectors were invited to participate in measurement quality assurance interactions with NBS, and all expressed a desire to do so.

NUREG/CR-2562: A USER'S GUIDE FOR THE STOCK-RECRUITMENT MODEL VALIDATION PROGRAM. CHRISTENSEN, S.; KIRK, B. L.; GOODYEAR, C. P. Oak Ridge National Laboratory. July 1982. 40pp. 8209230031. ORNL/TM-8216. 14990:303.

SRVAL is a FORTRAN IV computer code designed to aid in assessing the validity of curve-fits of the linearized Ricker stock-recruitment model, modified to incorporate multiple-age spawners and to include an environmental variable, to variously processed annual catch-per-unit-effort statistics for a fish population. It is sometimes asserted that curve-fits of this kind can be used to determine the sensitivity of fish populations to such man-induced stresses as entrainment and impingement at power plants. The SRVAL code was developed to test such assertions. It was utilized in testimony written in connection with the Hudson River Power Case (U.S. Environmental Protection Agency, Region II). This testimony was recently published as a NUREG report. Here, a user's guide for SRVAL is presented.

NUREG/CR-2563: RELATIVE STOCK COMPOSITION OF THE ATLANTIC COAST STRIPED BASS POPULATION: FURTHER ANALYSIS. WINKLE, W. V.; KUMAR, K. D. Oak Ridge National Laboratory. July 1982. 41pp. 8208260440. ORNL/TM-8217. 14570:270.

Fourteen variables derived from thirteen morphological characters were used in a stepwise discriminant analysis and a maximum likelihood analysis to estimate the relative contribution of striped bass (Morone saxatilis) stocks from the Hudson River and Chesapeake Bay to the coastal striped bass population. The analyses made use of the spawning-stock data and ocean data collected by Texas Instruments in 1975, although deletions were made to simplify the data to focus on relative contribution north of Chesapeake Bay and on sex and year-class differences. The discriminant function method misclassified approximately 20% of the spawning-stock fish. Errors in estimates of relative contribution for the spawning stock data were similar for the two methods of analysis. Estimates of relative contribution of the Hudson stock to the coastal population varied considerably among year classes. In particular, the estimated relative contribution for the 1965 year class was between 40 and 50%, while the relative contributions for the 1966, 1968, and 1969 year classes approximately 10% or less. The relative contribution of males was greater than that of females. The two methods of analysis gave similar estimates of relative contribution of the Hudson stock to the coastal population.

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. 13037:186.

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-2566: BWR REFILL-REFLOOD PROGRAM TASK 4.4-CCFL/REFILL SYSTEM EFFECTS TESTS (30 SECTOR):EVALUATION OF PARALLEL CHANNEL PHENOMENA. FINDLAY, J.A. General Electric Co. November 1982. 79pp. 8212010144. EPRI NP-2373. 16291:282.

This report interprets the results from SSTF separate effects test and system response tests and evaluates the parallel channel flow phenomena.

Parallel channel flow of interest occurs when there is a level in the lower plenum which allows redistribution of steam to the channel inlet orifices. Parallel channel effects are evidenced by three different flow regimes that may occur simultaneously. These regimes, identified from SSTF tests, are: (1) counter-current flow, (2) co-current upflow, and (3) liquid downflow.

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. 77pp. 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-2572: QUALITY ASSURANCE PROGRAM PLAN FOR THE REACTOR RESEARCH EXPERIMENT PROGRAMS (RREP). PIPHER, D. G. Sandia Laboratories. July 1982. 30pp. 8209230027. SAND81-2502. 14982:013.

This document describes the Quality Assurance Program plans which will be applied to tasks on Reactor Research Experiments performed on Sandia National Laboratories' reactors. It is the intent of this document that the Quality Assurance program comprise those elements which will provide adequate assurance that all components, equipment, and systems of the experiments will perform as designed, and hence prevent delays and costs due to rejections or failures. This plan has been developed to support the Reactor Research Experiments response to the quality criteria contained in the code of Federal Regulations Title 10, Part 50, Appendix B (10 CFR 50 Appendix B) as they relate to research and development in the nuclear field. This plan is also intended to meet the requirements of ANSI/N402-1976, entitled, "Quality Assurance Program Requirements for Research Reactors." Procedures applicable to the RREP projects and experiments are considered necessary to describe the methods used by SNLA to implement requirements.

NUREG/CR-2577: HISTORICAL SEISMICITY IN ARIZONA. DUBOIS, S. M. ; SBAR, M. L. ; NOWAK, T. A. Arizona, Univ. of. February 1982. 208pp. 8203190078. 12354:033.

An accurate historic earthquake catalog is an essential initial step in the evaluation of seismic risk. Prior to this study, a thorough compilation of Arizona earthquake data did not exist. The objectves of this study were to search for original source material, to document earthquakes felt in Arizona prior to 1980 and to organize these data for further analysis. These goals were met through collection of Arizona felt reports for specific events from national and state archives, local historical societies and published scientific literature. In addition, instrumental data for Arizona earthquakes from International Seismological Summary Bulletins, USGS publications (Earthquake Data Reports and Preliminary Determination of Epicenters) and original station logs were used to verify event times and locations. Copies of some Tucson seismograms were re-evaluated.

NUREG/CR-2579: TWO- AND THREE-DIMENSIONAL DIFFUSION THEORY CALCULATIONS FOR THE POOL CRITICAL ASSEMBLY PRESSURE VESSEL WALL BENCHMARK FACILITY. BALDWIN, C.A. Oak Ridge National Laboratory. July 1982. 57pp. 8208260499. ORNL/TM-8118. 14572:238.

Several diffusion theory calculations for the Pool Critical Assembly Pressure Vessel Wall Benchmark Facility core were performed. Two-dimensional calculations were performed to study the effect of various multigroup cross-section libraries on the calculated power distribution and to study the effect of a small water gap between the core and an aluminum window simulator. Three-dimensional calculated power distributions were compared with an experiment power distribution derived from miniature fission chamber measurements.

NUREG/CR-2580: STRUCTURAL REVIEW OF THE ROBERT E. GINNA NUCLEAR POWER PLANT UNDER COMBINED LOADS FOR THE SYSTEMATIC EVALUATION PROGRAM. WESLEY, D.; BANON, H. Lawrence Livermore Laboratory. March 1982. 73pp. 8204160062. UCRL-15433. 12713:296.

An evaluation of the capacity of the Robert E. Ginna containment structure to withstand combined Loss-of-Coolant-Accident (LOCA) and seismic loads was conducted as part of the Systematic Evaluation Program (SEP). Seismic loads were developed by scaling the loads developed previously in the SEP program for the 0.2g peak ground acceleration SSE to 0.17g which is consistent with the site specific ground response spectra developed by Lawrence Livermore National Laboratory (LLNL). Thermal and pressure loads were developed from pressure and temperature transients developed by LLNL for the LOCA conditions.

An axisymmetric, multilayer shell of revolution analytical model was developed for the containment vessel. The model included the concrete vertical wall and dome and included the steel liner. Appropriate boundary conditions representing the shell-to-base slab interface through neoprene pads were included. Since the base slab is founded on rock and the presence of the neoprene pads essentially isolates the base slab from the containment vessel, the base slab was not included in the model. No details, such as hatches or other penetrations, were evaluated in this phase of the SEP.

The analysis indicates that for the cylindrical portion of the vessel, liner and concrete stresses remain relatively low for the combined load condition and no damage is expected. In the vicinity of the base slab-containment wall interface, some localized yielding of the liner in the knuckle is predicted based on minimum code strength values.

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-2583: STRUCTURAL REVIEW OF THE PALISADES NUCLEAR POWER PLANT UNIT 1 CONTAINMENT STRUCTURE UNDER COMBINED LOADS FOR THE SYSTEMATIC EVALUATION PROGRAM. LIAW, C. Y.; DEBELING, A.; TSAI, N. C. Lawrence Livermore Laboratory. March 1982. 58pp. 8204020051. UCRL-53033. 12492:131.

A structural reassessment of the containment structure of the Palisades Nuclear Power Plant Unit 1 was performed for the Nuclear Regulatory Commission as part of the Systematic Evaluation Program. Conclusions about the ability of the containment structure to withstand the Abnormal/Extreme Environment are presented.

The reassessment focused mainly on the overall structural integrity of the containment building for the Abnormal/Extreme Environment. In this case, the Abnormal Environmental condition is caused by the worst case of either a Loss-of-Coolant Accident or a main steam line break. The extreme Environmental condition is the Safe Shutdown Earthquake.

NUREG/CR-2584: METEOROLOGICAL 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-2585: NUCLEAR POWER PLANT DAMAGE CONTROL MEASURES AND DESIGN CHANGES FOR SABOTAGE PROTECTION. LOBNER, P. R. Science Applications, Inc. * Sandia Laboratories. August 1982. 355pp. 8209270133. SAND82-7011. 15514:001.

This report documents the engineering evaluation of twenty-seven proposed damage control measures and associated system-level design changes that could be of potential benefit in providing protection against sabotage at commercial light water reactor (LWR) power plants.

The damage control measures emphasize the use of existing systems in normal or alternate modes of operation. The proposed system-level design changes are those necessary to support the use of existing systems in alternate modes. To the extent practical, the system-level design changes have been limited in scope to those that could be retrofitted in existing nuclear power plants. The potential applicability of each damage control measure and system-level design change is defined, and the impact of its implementation is subjectively estimated. The potential role of damage control and design change in an integrated sabotage protection system is discussed in this report.

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-2590: PHASE SEPARATION PHENOMENA IN BRANCHING CONDUITS. SABA, N.; LAHEY, R. T. Rensselaer Polytechnic Inst. March 1982. 200pp. 8204010553. 12490:001.

The degree of phase separation of a two-phase (air/water) mixture flowing through a plexiglas tee test section was measured. In addition, flow visualization, using high speed photography, was performed. The experimental design considerations, error analysis and the dependence of the observed phase separation on global parameters, such as inlet quality, mass flux and separation angle, are discussed.

The pressure gradients were measured along with various conduits and the differential pressure was obtained at the tee by extrapolation. It was found that the degree of phase separation was quite pronounced, with the vapor phase preferentially separating into the branch. Using these data, a physically-based empirical model was developed with which to calculate the phasic distribution of a subsonic two-phase mixture in the downstream branches of a branching conduit.

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-2598: NUCLEAR POWER PLANT CONTROL ROOM TASK ANALYSIS:PILOT STUDY FOR PRESSURIZED WATER REACTORS. BARKS,D.B.; KOZINSKY,E.J.; ECKEL,S. Oak Ridge National Laboratory. July 1982. 109pp. 8207210175. ORNL/SUB/79-404. 13990:001.

Report covers nuclear plant task analysis pilot study. Five data sources were investigated to provide information for a task analysis. (1) written operating procedures (event-based); (2) interviews with subject matter experts (the control room operators); (3) videotapes of the control room operators (senior reactor operators and reactor operators) while responding to each event in a simulator; (4) walk-/talk-throughs conducted by control room operators for each event; and (5) simulator data from the PMS. A Westinghouse pressurized water reactor nuclear power plant simulator was utilized in this study. Four abnormal or emergency events were studied: nuclear instrument failure; small break loss of coolant accident (LOCA); steam generator tube leak; and inadvertent safety injection at power. Upon completion of the task analyses, computerized data reduction was performed. A PRIME 1-1000 computer was used to manage the task analytic data base. Videotapes, although the richest single source of information, frequently left some gaps in the data accumulations, e.g. the exact switch manipulation, and the sequence of manipulations was sometimes difficult to discern. The PMS data source provided that data. This study demonstrated the usefulness of a task analysis methodology that combines traditional data collection by analysts with PMS data to provide a complete task analytic data set. The power of a machine-readable computer data base to support various applications of task analytic data was also demonstrated.

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-2601 V01: TECHNOLOGY, SAFETY AND COSTS OF DECOMMISSIONING REFERENCE LIGHT WATER REACTORS FOLLOWING POSTULATED ACCIDENTS: Main Report. MURPHY, E.S.; HOLTER, G.M. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1982. 461pp. 8212270033. PNL-4247. 16569:001.

This report presents a conceptual study of post accident decommissioning of light-water-reactors and the accident cleanup that precedes the decommissioning. The study provides information on the available technology, safety considerations, and probable costs of decommissioning, and the accident cleanup, of a reference PWR and BWR following a postulated accident. Three postulated accident scenarios are used in the report to illustrate a range of technological requirements, costs (in 1981 dollars), occupational radiation doses, potential radiation dose to the public, and other safety impacts. The decommissioning alternatives considered are DECON (immediate decontamination), SAFSTOR (safe storage followed by deferred decontamination), and ENTOMB (entombment). The study evaluates the sensitivity of the costs of accident cleanup to various factors which can influence them.

NUREG/CR-2601 V02: TECHNOLOGY, SAFETY AND COSTS OF DECOMMISSIONING REFERENCE LIGHT WATER REACTORS FOLLOWING POSTULATED ACCIDENTS: Appendices. MURPHY, E.S.; HOLTER, G.M. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1982. 626pp. 8212270232. PNL-4247. 16554:001.

This report presents a conceptual study of post accident decommissioning of light-water-reactors and the accident cleanup that precedes the decommissioning. The study provides information on the available technology, safety considerations, and probable costs of decommissioning, and the accident cleanup, of a reference PWR and BWR following a postulated accident. Three postulated accident scenarios are used in the report to illustrate a range of technological requirements, costs (in 1981 dollars), occupational radiation doses, potential radiation dose to the public, and other safety impacts. The decommissioning alternatives considered are DECON (immediate decontamination), SAFSTOR (safe storage followed by deferred decontamination), and ENTOMB (entombment). The study evaluates the sensitivity of the costs of accident cleanup to various factors which can influence them.

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-2606: ANALYSES OF 1/15 SCALE CREARE BYPASS TRANSIENT EXPERIMENTS. KMETYK, L. N. ; BUXTON, L. D. ; COLE, R. K. Sandia Laboratories. September 1982. 61pp. 8210050397. SAND81-1932. 15625:329.

RELAP4 analyses of several 1/15 scale Creare H-series bypass transient experiments have been done to investigate the effect of using different downcomer nodalizations, physical scales, slip models, and vapor fraction donoring methods. Most of the analyses were thermal equilibrium calculations peformed with RELAP4/MOD5, but a few such calculations were done with RELAP4/MOD6 and RELAP4/MOD7, which contain improved slip models. In order to estimate the importance of nonequilibrium effects, additional analyses were performed with TRAC-PD2, RELAP5 and the nonequilibrium option of RELAP4/MOD7. The purpose of these studies was to determine whether results from Westinghouse's calculations of the Creare experiments, which were done with a UHI-modified version of SATAN, were sufficient to guarantee SATAN would be "conservative" with respect to ECC bypass in full-scale plant analyses.

The two major results of this study are that (1) a nonequilibrium code may be needed to correctly model the dominant flow phenomena of these particular Creare tests, and (2) results from a full-scale nodalization developed via K* scaling criteria cannot be validly compared to the 1/15 scale Creare data. Therefore, the calculations reported here indicate that Westinghouse's Creare analysis results have not proven their UHI-modified version of SATAN will always generate conservative values for ECC bypass.

NUREG/CR-2608: REVIEW OF DESIGN APPROACHES APPLICABLE TO DEWATERING URANIUM MILL TRAILINGS DISPOSAL PITS. GUTNECHT, P. J.; GATES, T. E. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1982. 35pp. 8203310035. PNL-4150. 12462:306.

This report is a review of design approaches in the literature that may be applicable to uranium mill tailings drainage. Tailings dewatering is required in deep mined-out pits used for wet tailings disposal. Agricultural drainage theory is reviewed because it is seen as the most applicable technology. It is concluded that the standard drain-pipe envelope design criteria should be easily adapted. The difference in dewatering objectives and physical characteristics between agricultural and tailings drainage systems prevent direct technology transfer with respect to drain spacing calculations. Recommendations for further research are based on the drainage features unique to uranium mill tailings. It is recommended that transient solutions be applied to describe liquid movement through saturated and partially saturated tailings. Modeling should be used to evaluate the benefits of drainage design approaches after careful consideration of potential construction problems.

NUREG/CR-2609: THERMAL-HYDRAULIC TEST FACILITY BUNDLE 3 IN-CORE INSTRUMENTATION AND OPERATING HISTORY. OTT, L. J. Oak Ridge National Laboratory. September 1982. 104pp. 8210210039. ORNL/NUREG-75. 15774:006.

This report presents the chronological history of the Thermal-Hydraulic Test Facility (THTF) Bundle 3. This bundle is one of the most extensively instrumented bundles ever used in light water reactor safety research. This compilation documents the in-core instrument measurement sites and operating history for the original Bundle 3 and two major refurbishments to the bundle. Additional information (such as radial dimensions and surface emissivity) that will be needed for thermal analysis of the fuel rod simulator responses during THTF testing is also provided.

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. 140pp. 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. 80pp. 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-2616: EVALUATION OF ISOTOPE MIGRATION - LAND BURIAL.Water Chemistry At Commercially Operated Low-Level Radioactive Waste Disposal Sites.Status Report,October 1980-September 1981.

PIETRZAK, R. F. ; CZYSCINSKI, K. S. ; WEISS, A. J. Brookhaven National Laboratory. July 1982. 54pp. 8208260466. BNL-NUREG-51514. 14570:180.

The prime consideration for continued use of shallow land burial practices for the disposal of low-level radioactive waste is the containment of radionuclides. Before additional disposal sites for commercial low-level waste can be licensed, the existing sites must be evaluated in terms of their effectiveness for retaining radionuclides. This study is an attempt to monitor the behavior of existing low-level disposal sites, provide an understanding of significant factors which affect prediction of radionuclide movement along the groundwater flow paths, and assist in the development of criteria for the selection and licensing of future low-level disposal sites.

NUREG/CR-2617: PROPERTIES OF RADIOACTIVE WASTES AND WASTE CONTAINERS. Status Report, October 1980-September 1981. MORCOS, N. ; DAYAL, R. ; WEISS, A. J. Brookhaven National Laboratory. August 1982. 186pp. 8209230040. BNL-NUREG-51515. 14982:045.

This report summarizes work performed during the 1981 fiscal year. Licensing of near surface low-level radioactive waste disposal sites and waste forms/containers requires the ability to predict the dispersibility of radionuclides from waste forms and waste containers disposed in burial sites. The objectives of the research program are to provide an improved understanding of phenomena, testing methodology and data. This improves the NRC's capability to predict low-level waste isolation performance, and to provide a better technical basis for regulatory standards. The areas addressed to meet these objectives during the 1981 fiscal year were: a. Leachability and compressive strength of boric acid waste in Portland III cement. The tracers used for the study were (137)Cs, (85)Sr, and (60)Co. b. Leachability of (137)Cs, (85)Sr, and (60)Co from organic ion exchange resin/Portland III and Lumnite cements. c. Displacement of (137)Cs, (85)Sr, and (60)Co from organic ion exchange resins upon mixing with Portland II and Lumnite cements. d. Leachability of organic ion exchange resins/Bitumen composites using resins in the H+, Na+, Cs+, Sr+2, and SO(4)-2 forms, and (137)Cs and (85)Sr tracers. e. Correlation of (137)Cs leachability from small-scale (laboratory) samples to large-scale waste forms, and f. Hydrostatic testing of DOT 17H drums.

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-2619: HYDRODYNAMIC INSTABILITY AND THERMAL CONVECTION IN A HORIZONTAL LAYER OF TWO IMMISCIBLE FLUIDS WITH INTERNAL HEAT GENERATION. KULACKI, F. A.; NGUYEN, A. T. Ohio State Univ. March 1982. 206pp. 8204210635. 12799:297.

Results are reported of experimental and theoretical investigations of thermal convection in a horizontal layer composed of two immiscible fluids, with uniform internal energy generation in the lower sublayer. A linear stability analysis, numerical study, and experimental measurements are presented. The fluid layer is bounded from below by a rigid, indulated surface and from above, by a rigid isothermal surface. In the stability analysis, conditions for the onset of convection via the linearized perturbation approach are obtained by evaluation of a determinant of a matrix comprising two working matrices that are associated with each fluid sublayer. A simple explicit finite-difference technique is used for the numerical solution of the partial differential equations governing the temperature and flow fields. Experimental measurements of transient and steady convection up to Rayleigh number of 2 x 10(11) in the internally heated sublayer are presented for both the silicone oil-water and heptane-water systems. The Nusselt number is dependent on the sublayer thickness ratio over the entire range of Rayleigh numbers investigated. A thin upper sublayer first decreases the Nusselt number at a given Rayleigh number; a minimum value is reached and, thereafter, the Nusselt number increases when the upper sublayer thickness is increased.

NUREG/CR-2620: AN ANALYSIS OF PROPOSED BWR INLET FLOW BLOCKAGE EXPERIMENTS USING MAYU-4B. NISSLEY, M. E.; GAY, R. R.; LAHEY, R. T. EG&G, Inc. July 1982. 51pp. 8209240527. EGG-2181. 14999:249.

The purpose of this report is to present an analytical study of hypothetical boiling water reactor (BWR) inlet flow blockage accidents. To this end, MAYU-4, a one-dimensional drift-flux digital computer code, was modified to account for nonequilibrium vapor generation, radiation heat transfer, rod bundle countercurrent-flow-limited phenomena, and the exothermic metal-water reaction of the cladding. The revised code, MAYU-4b, was used to investigate postulated BWR inlet flow blockage events and the feasibility of such testing in the Power Burst Facility (PBF) at the Idaho National Engineering Laboratory. It was found that, due to the relatively short length of the PBF core (3 ft), simulation of all facets of a BWR inlet flow blockage event would be difficult. Nevertheless, it may be possible to gain valuable information on fuel failure propagation and the potential for vapor explosions in the PBF.

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-2623: THE ALLOCATION OF FUNCTIONS IN MAN-MACHINE SYSTEMS: A PERSPECTIVE AND LITERATURE REVIEW. PRICE, H. E. ; MAISANO, R. E. ; VAN COTT, H. P. Oak Ridge National Laboratory. July 1982. 149pp. 8207210181. ORNL/SUB/81-902. 13990:110.

This report reviews the literature relevant to allocation of functions and presents a procedure for the allocation process applicable to nuclear power plant control rooms. An historical perspective of man's relationship with technology is given as background. Methods and models that have been developed to aid the allocation process are then considered, followed by examples of real-world applications. The relationship of allocation of function to the system development process is outlined. The report then turns to the proposed procedure of the allocation process. This procedure, conducted in a series of steps, leads to decisions on what roles and functions man and machine will play in the complex man-machine system under consideration. These decisions are based on criteria developed from the literature on human and machine capabilities and limitations. The resultant hypothesized allocations are tested against environmental, system, and psychological constraints. Consideration is also given to human operator acceptance of automation, a crucial question with the increased use of computers and computer-based aids. An example of how this procedure might be applied to one aspect of the nuclear area is detailed.

NUREG/CR-2624: AQUATIC PREDATOR FEEDING IN PATCHY ENVIRONMENTS: ERROR INTRODUCED INTO MODELS BY ASSUMING PREY SPATIAL HOMOGENEITY. HAAR, R. T. ; SWARTZMAN, G. L. Washington, Univ. of. March 1982. 24pp. 8205070350. 13009:286.

Very high biases can be introduced into the computation of ration in ecological models by assuming that prey concentration is spatially homogeneous and that prey density has a saturating effect on predator ration. Using a Monte Carlo simulation of a predator feeding randomly in a negatively binomially distributed prey environment, we were able to quantify the magnitude of this bias as a function of the parameters of the negative binomial distribution and of the function relating ration to prey density. Bias increases with increasing patchiness and decreases with increasing prey density. Estimates of bias is not very typical predator-prey interactions. In these cases bias is not very high, less than 10% in nearly all cases. In cases where bias would be high, we conclude that the predator cannot feed in a random fashion in order to meet metabolic demands and known growth rates.

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-2626: EX-CORE NEUTRON DETECTORS FOR REACTOR VESSEL LEVEL MEASUREMENT. ANDERSON, J. L.; ANDERSON, R. L.; MILLER, G. N. Oak Ridge National Laboratory. March 1982. 74pp. 8203290047. ORNL/TM-8267. 12442:080.

This document, entitled "Ex-Core Neutron Detectors for Reactor Vessel Level Measurement," presents the evaluation of an application by Alabama Power Company and National Nuclear Corporation for a Non-Invasive Coolant Level Monitor for the Farley Power Station, Units 1 and 2.

The principle of the non-invasive coolant level monitoring system is based on the detection of 2.2 MeV photoneutrons produced by the interaction of high energy gamma-rays emitted by fission products in the core, in particular, (140)La, with deuterium in the coolant water inside the vessel. The system utilizes neutron detectors located at the top and at the bottom of the reactor vessel.

NUREG/CR-2627: INADEQUATE CORE COOLING INSTRUMENTATION USING HEATED JUNCTION THERMOCOUPLES FOR REACTOR VESSEL LEVEL MEASUREMENT. ANDERSON, R. L.; ANDERSON, J. L.; MILLER, G. N. Oak Ridge National Laboratory. March 1982. 140pp. 8203180446. ORNL/TM-8268. 12339:255.

This document, entitled "Inadequate Core Cooling Instrumentation Using Heated Junction Thermocouples for Reactor Vessel Level Measurement," presents a technical review of the inadequate core cooling instrumentation with a Reactor Vessel Level Monitoring System using Heated Junction Thermocouples proposed by Combustion Engineering, Inc., for pressurized water reactors.

Emphasis was placed on evaluation of the generic Inadequate Core Cooling (ICC) Instrumentation System as a whole which includes, besides the heated junction thermocouple reactor vessel level measurement, the saturation margin monitor, the core exit thermocouples, the qualified safety parameter display system, and the critical functions monitor system.

NUREG/CR-2628: INADEQUATE CORE COOLING INSTRUMENTATION USING DIFFERENTIAL PRESSURE FOR REACTOR VESSEL LEVEL MEASUREMENT. MILLER, G. N.; ANDERSON, J. L.; ANDERSON, R. L. Oak Ridge National Laboratory. March 1982. 145pp. 8203180451. ORNL/TM-8269. 12340:034.

This document, entitled "Inadequate Core Cooling Instrumentation Using Differential Pressure for Reactor Vessel Level Measurement," presents a technical review of the Inadequate Core Cooling Instrumentation with a Reactor Vessel Level Montitoring System using a differential pressure measurement system proposed by Westinghouse, Inc., for pressurized water reactors.

Emphasis was placed on evaluation of the generic Inadequate Core Cooling (ICC) Instrumentation System as a whole which includes, besides the differential pressure reactor vessel level measurement, the saturation margin monitor, the core exit thermocouples and the display system (either the 7300, an analog display, or the microprocessor based system with a plasma panel display).

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-2640: FACILITY DESCRIPTION--THTF MOD 3 ORNL PWR BDHT SEPARATE-EFFECTS PROGRAM. DURALL, R. L. ; GOULD, S. S. ; MAILEN, G. S. ; et al. Oak Ridge National Laboratory. October 1982. 400pp. 8211120094. ORNL/TM-7842. 16064:133.

This report describes the Thermal-Hydraulic Test Facility (THTF) as modified for tests with Bundle 3, a 64-rod bundle of indirectly electrically heated fuel rod simulators. The report provides a description of the basic facility and instrumentation as well as a test-specific facility description for each of the primary tests run at the THTF during the operational period from June 1979 to February 1981. The primary tests include the Small-Break Loss-of-Coolant Accident (LOCA)-I Test Series (3.02.10C-H), the transient Upflow Film Boiling Tests (3.03.6AR, 3.06.6B, 3.08.6C), the Steady-State Upflow Film Boiling Test Series (3.07.9-), the Double-Ended Cold-Leg Break Test (3.05.5B), the Small-Break LOCA-II Test Series (3.09.10I-X), and the Intermediate-Flow Heat Transfer Test Series (3.10.11A-H). Two major modifications made to the bundle and test section over this period are also described.

NUREG/CR-2641: THE IN-PLANT RELIABILITY DATA BASE FOR NUCLEAR POWER PLANT COMPONENTS: Data Collection And Methodology Report. DRAGO, J. P. ; BORKOWSKI, R. J. ; PIKE, D. H. ; et al. Oak Ridge National Laboratory. November 1982. 93pp. 8212270435. ORNL/TM-8271. 16571:089.

The development of a component reliability data base for use in nuclear power plant probabilistic risk assessments and reliability studies is presented in this report. The sources of the data are the in-plant maintenance work request records from a sample of nuclear power plants. This data base is called the In-Plant Reliability Data (IPRD) system. Features of the IPRD system are compared with other data sources such as the Licensee Event Report system, the Nuclear Plant Reliability Data system, and IEEE Standard 500. Generic descriptions of nuclear power plant systems formulated for IPRD are given.

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. 138pp. 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-2643: A REVIEW OF SELECTED METHODS FOR PROTECTION AGAINST SABOTAGE BY AN INSIDER. GOLDMAN,L.A.; LOBNER,P.R. Science Applications, Inc. * Sandia Laboratories. October 1982. 150pp. 8212130058. SAND82-7036. 16416:198.

This report presents an overview of the following three basic measures for protecting a nuclear power plant against radiological sabotage by an insider: (1) physical protection, (2) damage control, and (3) plant design. Each of these three basic sabotage protection measures is shown to be at least partially effective in protecting against radiological sabotage by an insider, however, no single approach provides a complete solution. Protection against the insider therefore requires an integrated approach which includes the best features of the individual sabotage protection concepts. The basic elements of an integrated insider sabotage protection system are identified. This report also presents an overview of the following insider sabotage-related topics: (1) methodologies for insider analysis, (2) detection of insider sabotage actions, (3) interaction between plant personnel and the insider safeguards system and (4) security clearances. The relationship between these additional topics and the three basic measures for protecting against radiological sabotage by an insider is described.

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-2646: A MODEL FOR BOILING AND DRYOUT IN PARTICLE BEDS. LIPINSKI, R. J. Sandia Laboratories. August 1982. 185pp. 8209210433. SAND82-0765. 14948:123.

Over the last ten years experiments and modeling of dryout in particle beds have produced over fifty papers. Considering only volume-heated beds, over 250 dryout measurements have been made, and are listed in this work. In addition, fifteen models to predict dryout have been produced and are discussed. A model is based on conservation laws for mass, momentum, and energy. The initial coupled differential equations are reduced to a single first-order differential equation with an algebraic equation for the upper boundary condition. The model includes the effects of both laminar and turbulent flow, two-phase friction, and capillary force. The boundary condition at the bed bottom includes the possibility of inflowing liquid and either an adiabatic or a bottom-cooled support structure. The top of the bed may be either channeled or subcooled. In the first case the channel length and the saturation at the base of the channels are predicted. In the latter case, a criterion for penetration of the subcooled zone by channels is obtained.

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-2649: INFLUENCE OF THE MAGNETOMECHANICAL EFFECT IN TESTING OF INDUCTIVELY HEATED FERRITIC STEEL. JONES, W. B. Sandia Laboratories. July 1982. 23pp. 8208260491. SAND82-0752. 14595:009.

The use of a thermocouple spot welded to a specimen shoulder for temperature control in induction heated elevated temperature mechanical tests of ferritic alloys has been found inadequate to provide an unvarying temperature in the specimen gauge. The magneto-mechanical effect sufficiently alters the inductive coupling to produce both temperature cyclng of plus 15K-50K and on overall cooling of 10K with mechanical cycling. These temperature changes can dominate the apparent strain measurements under conditions of limited plasticity.

NUREG/CR-2650: ALLOWABLE SHIPMENT FREQUENCIES FOR THE TRANSPORT OF TOXIC GASES NEAR NUCLEAR POWER PLANTS. BENNETT, D. E.; HEATH, D. C. Sandia Laboratories. November 1982. 37pp. 8212220162. SAND82-0774. 16527:065.

One part of the safety analysis of offsite hazards for a nuclear power plant is consideration of accidents which could release toxic gases or vapors and thus jeopardize plant safety through incapacitation of the control room operators. The purpose of this work is to provide generic, bounding estimates of the maximum allowable shipping frequencies for the transport of a chemical near the plant, such that the regulatory criteria for the protection of the operators are met. A probabilistic methodology was developed and then applied to the truck and rail transport of an example chemical, chlorine. The current regulatory criteria are discussed in detail. For this study, a maximum allowable probability of occurrence of operator incapacitation of 10-5 per year was used in the example calculation for each mode of transport. Comprehensive tables of conditional probabilities are presented. Maximum allowable shipping frequencies are then derived. These frequencies could be used as part of a generic, bounding criterion for the screening of toxic hazards safety analyses.

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-2654: PROCEDURES FOR ANALYZING THE EFFECTIVENESS OF SIREN SYSTEMS FOR ALERTING THE PUBLIC. KEAST, D. N.; TOWERS, D. A.; ANDERSON, G. S.; et al. Bolt, Beranck & Newman, Inc. September 1982. 140pp. 8209280311. PNL-4227. 15548:339.

NUREG-0654, Revision 1 (Criteria for Preparation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants), Appendix 3, discusses requirements of the licensee to implement a prompt notification system within the 10-mile emergency planning zone (EPZ) surrounding a nuclear facility. Sirens are being installed for use as part of or as the entire notification system by many licensees. This report describes a procedure for predicting siren system effectiveness under defined conditions within the EPZ's. The procedure requires a good topographical map and knowledge of the meteorology, demographics, and human activity patterns within the EPZ. The procedure is intended to be applied to systems of sirens and to obtain average results for a large number (30 or more) listener locations.

NUREG/CR-2655: EVALUATION OF THE PROMPT ALERTING SYSTEMS AT FOUR NUCLEAR POWER STATIONS. TOWERS, D. A.; ANDERSON, G. S.; KEAST, D. N.; et al. Bolt, Beranck & Newman, Inc. September 1982. 271pp. 8209280317. PNL-4226. 15549:118.

This report presents evaluations of the prompt notification siren systems at the following four U.S. nuclear power facilities: Trojan, Three Mile Island, Indian Point, and Zion. The objective of these evaluations was to provide examples of an analytical procedure for predicting siren-system effectiveness under specific conditions in the 10-mile emergency planning zone (EPZ) surrounding nuclear power plants. This analytical procedure is discussed in NUREG/CR-2654.

NUREG/CR-2656: LOSS-OF-FEEDWATER TRANSIENTS FOR THE ZION-1 PRESSURIZED WATER REACTOR. Docket No. 50-295. DEMUTH, N. S. ; DOBRANICH, D. ; HUNNINGER, R. J. Los Alamos Scientific Laboratory. July 1982. 83pp. 8209130279. LA-9296-MS. 14781:220.

The response of the Westinghouse Zion-1 pressurized water reactor to transients initiated by loss of main feedwater with auxiliary feedwater unavailable was simulated using the Transient Reactor Analysis Code (TRAC). The normal response mode in which emergency systems perform as designed was first studied to identify critical equipment performance and operator actions necessary for normal recovery. Subsequent analyses were performed to determine the effects of additional equipment failures, such as valves sticking open, and delayed or degraded operation of emergency systems. Strategies were developed for operator actions not covered in existing emergency procedures and were tested using TRAC simulations to evaluate their effectiveness in preventing core uncovery.

NUREG/CR-2661: FLAW MEASUREMENT USING ULTRASONICS IN THICK PRESSURE VESSEL STEEL. COOK, K. V. ; LATIMER, P. J. ; MCCLUNG, R. W. Oak Ridge National Laboratory. August 1982. 32pp. 8209230050. ORNL/TM-8295. 14988:214.

The net effects of such variables as beam width, beam angle, and flaw geometry were considered with regard to their total impact upon flaw measurement. The boundaries of accuracy and repeatability were established for the manual measurement of a limited number of flaws by the American Society of Mechanical Engineers code procedures. Both surface and buried reflectors were considered, and changes were recommended in both the detection and measurement of buried midplane flaws. Correlations were made between the magnitude of lateral beam spread and length measurements. Approximate corrections for lateral beam spread were applied to measurements of large flaw lengths. Correlations were made between code-measured flaw depths and real depths on a limited number of flaws of various depths and orientations. Finally a brief study was made to determine the influence of the angular orientation of the search unit upon the received amplitude from the flaw.

NUREG/CR-2662: WAITING TIMES AND GENERALIZED FIBONACCI SEQUENCES. UPPULURI, V. R. R. ; PATIL, S. A. Oak Ridge National Laboratory. July 1982. 13pp. 8207210025. ORNL/CSD/TM-184. 13993:318.

Suppose we have a multinormal population with k possible outcomes E(1), E(2), . . . , E(k) and associated probabilities pi(1), pi(2), . . . , pi(k). At each of the independent trials, one of the outcomes is observed. One may be interested in the waiting time for the occurrence of a specified event, which consists of a succession of outcomes. In this paper, we consider the probability distribution of the waiting times associated with specified events, and show how they generalize the Fibonacci, Tribonacci, . . . , sequences in different ways. This is possible, since the probability generating functions of the associated waiting time random variable can be utilized to derive the probability distributions.

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-2665: POWER BURST FACILITY THERMOCOUPLE EFFECTS TEST RESULTS REPORT, TEST SERIES TC-1, TC-3, AND TC-4. GARNER, G. W.; MACDONALD, P. E. EG&G, Inc. July 1982. 43pp. 8209230035. EGG-2190. 14982:232.

Fuel rod cladding surface temperatures have been estimated in Loss-of-Fluid Test (LOFT) Facility and in Power Burst Facility loss-of-coolant accident (LOCA) tests using data obtained with thermocouples welded to the cladding outer surface. These cladding temperature estimates have been questioned because cladding surface thermocouples may act as cooling fins and local sites of cladding rewet, thereby delaying the time of occurrence of critical heat flux (CHF) and providing increased surface heat transfer. This report presents the results of three series of light water reactor fuel behavior tests (Thermocouple Effects Test Series TC-1, TC-3, and TC-4) that were performed in the Power Burst Facility to specifically evaluate the influence of cladding surface thermocouples on the thermal behavior of nuclear fuel rods under LOCA conditions. Twelve tests were performed in the three test series. Differences between tests included variations in system thermal-hydraulic conditions and in the initial test rod power level, as well as differences in design of the internal cladding thermocouples. This latter difference provided data for calibration of the surface thermocouples and evaluation of the influence of surface thermocouples on the time of occurrence of critical heat flux and post-CHF heat transfer.

NUREG/CR-2670: JOB ANALYSIS OF MAINTENANCE MECHANIC POSITION FOR THE NUCLEAR POWER PLANT MAINTENANCE PERSONNEL RELIABILITY MODEL. SIEGEL, A. I.; BARTTER, W. D.; KOPSTEIN, F. F. Oak Ridge National Laboratory. July 1982. 109pp. 8207260001. ORNL/TM-8301. 14069:060.

This report is one of a series planned to describe the results of a program to define, develop, validate, and disseminate a methodology for the quantitative prediction of human reliability in the conduct of maintenance tasks in nuclear power plants (NPPs). ORNL has subcontracted portions of this effort to Applied Psychological Services, Inc. An analysis was performed of the job of maintenance mechanics in nuclear power plants in order to provide a part of the information required for modeling nuclear plant maintenance activities. It is believed that such a model would provide substantial insights into the various human, equipment, and environmental factors likely to affect reliability of maintenance personnel, and thereby suggest and allow evaluation of standards, design changes or other modifications to improve reliability and minimize public risk. The task list method of job survey was selected because the approach minimizes data acquisition costs, interferes minimally with the work of job incumbents, is comprehensive and objective, provides quantitative data, and seemed highly appropriate for achieving the goals of the work. In collaboration with supervisory personnel at BWR and PWR nuclear power plants, a comprehensive list of 107 tasks performed by maintenance mechanics was developed. The tasks were grouped within six generic work functions: remove and install, test and repair, inspect and perform preventive maintenance, miscellaneous, communication, and report preparation.

The lists were then assembled into appropriate questionnaire form. The results of the survey are reported in this study.

NUREG/CR-2671: THE MARVIKEN FULL SCALE CRITICAL FLOW TESTS. Summary Report. * MARVIKEN. May 1982. 286pp. 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-2672 V01: SBLOCA OUTSIDE CONTAINMENT AT BROWNS FERRY UNIT 1-ACCIDENT SEQUENCE ANALYSIS. CONDON, W.A.; HARRINGTON, R.M.; GREENE, S.R.; et al. Oak Ridge National Laboratory. November 1982. 280pp. 8212140479. ORNL/TM-8119/V1. 16433:074.

This study describes the predicted response of Unit 1 at the Browns Ferry Nuclear Plant to a postulated small-break loss-of-coolant accident outside of the primary containment. The break has been assumed to occur in the scram discharge volume piping immediately following a reactor scram that cannot be reset. The events before core uncovery are discussed for both the worst-case accident sequence without operator action and for the more likely sequences with operator action. Without operator action, the events after core uncovery would include core meltdown and subsequent containment failure, and this event sequence has been determined through use of the MARCH code. An estimate of the magnitude and timing of the concomitant release of the noble gas, cesium, and iodine-based fission products to the evironment is provided in Volume 2 of this report.

NUREG/CR-2673: EVALUATION OF THERMAL DEVICES FOR DETECTING IN-VESSEL COOLANT LEVELS IN PWRS. HARDY, J.E.; DAVIS, C.E.; TURNAGE, K.G.; et al. Oak Ridge National Laboratory. September 1982. 73pp. 8210210037. ORNL/TM-8306. 15773:295.

From investigations conducted immediately after the Three Mile Island nuclear power plant accident, some safety areas needing improvement were identified. A resulting requirement was the unambiguous detection of the approach to adequate core cooling. Designs to meet this requirement have generally included new instrumentation to monitor the coolant level in the reactor vessel. Thermal sensors proposed for use in pressurized-water reactor (PWR) vessels were tested and evaluated. The thermal devices tested use pairs of K-type thermocouples or resistance temperature detectors to sense the cooling capacity of the medium surrounding the device. One sensor of the pair is heated by an electric current, while the unheated one senses the ambient fluid temperature. The temperature difference between the heated and unheated sensors provides an indication of the cooling capacity of the surrounding fluid. Experiments that simulated the thermal-hydraulic conditions of a postulated PWR loss-of-coolant accident (LOCA) were run, including both natural- and forced-convection two-phase flow tests. Results suggest thermal level devices generally indicate the existence of poor cooling conditions in LOCA environments. Preliminary evaluation of these protection systems is given.

NUREG/CR-2675 V01: RELEVANCE OF BIOTIC PATHWAYS TO THE LONG-TERM REGULATION OF NUCLEAR WASTE DISPOSAL. A Report On Tasks 1 And 2 Of Phase I. MCKENZIE,D.H.; CADWELL,L.L.; CUSHING,L.E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1982. 72pp. 8208180126. PNL-4241. 14391:289.

The purpose of the work reported here was to evaluate the relevance of biotic transport to the assessment of impacts and licensing of low-level waste disposal sites. Available computer models and their recent applications at low-level waste disposal sites are considered. Biotic transport mechanisms and processes for both terrestrial and aquatic systems are presented with examples from existing waste disposal sites. Following a proposed system for ranking radionuclides by their potential for biotic transport, recommendations for completing Phase 1 research are presented. To evaluate the long-term importance of biotic transport at low-level waste sites, scenarios for biotic pathways and mechanisms need to be developed. Scenarios should begin with a description of the waste form and should include a description of biotic processes and mechanisms, approximations of the magnitude of materials transported, and a linkage to processes or mechanisms in existing models. Once these scenarios are in place, existing models could be used to evaluate impacts resulting from biotic transport and to assess the relevance to site selection and licensing of low-level waste disposal sites.

NUREG/CR-2675 V02: RELEVANCE OF BIOTIC PATHWAYS TO THE REGULATION OF NUCLEAR WASTE DISPOSAL:Topical Report On Reference Western Arid Low-Level Waste Sites. MCKENZIE,D.H.; CADWELL,L.L.; EBERHARDT,L.E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1982. 108pp. 8211110079. PNL-4241. 16047:017.

The purpose of the work reported here was to estimate the potential dose to man resulting from biotic transport mechanisms at a reference arid low-level waste site. The site description includes waste inventories, site characteristics, and biological communities. Parameter values for biotic transport are based on data reported in current literature. Calculations of radionuclide decay and waste container decomposition are made to estimate the amount of radioactive material available for biotic transport and exposure scenarios 100 years following site closure. One hundred year dose to man estimates based on biotic transport are estimated to be of the same order of magnitude as dose estimates resulting from the more commonly evaluated intrusion-agricultural scenario reported in NRC's DEIS for 10 CFR 61. These results indicate that biotic transport has the potential to influence low-level waste site performance. The reported lack of potential importance of biotic transport at low-level waste sites in earlier assessment studies is not confirmed by the findings presented in this report. Through biotic transport, radionuclides may be moved to locations where they can enter exposure pathways to man.

NUREG/CR-2676: VESSEL V-8 REPAIR AND PROTECTION OF LOW UPPER-SHELF WELDMENT. DOMIAN,H.A. Babcock & Wilcox Co. * Oak Ridge National Laboratory. July 1982. 212pp. 8209270046. ORNL/SUB/81-858. 15518:026.

This report describes the weld repair of the Heavy Section Steel Technology (HSST) Program Intermediate Test Vessel V-8 and the making of a submerged-arc low upper-shelf test weld as a long seam in the vessel. The development of the low upper-shelf weld metal and its mechanical property characterization, including fracture toughness, is reported. From replicate characterization weld test results it is expected that the vessel seam test weld properties are likely to have the following values:

Room temperature yield strength (YS) greater than/= 66.3 ksi
Upper-shelf energies (USE) less than/= 43.5 ft-lbs
J(1c) less than/= 0.335 kip/in at 300 degrees F
J = 50T less than or equal to 1.131 kip/in at 300 degrees F

These values are not far different than those reported for some irradiated welds.

NUREG/CR-2677: ESP AND NOAH-COMPUTER PROGRAMS FOR FLOOD RISK ANALYSIS OF NUCLEAR POWER PLANTS. WAGNER, D.P.; MONTAGUE, D.F.; ROONEY, J.J.; et al. JBF Associates. July 1982. 269pp. 8209270036. 15517:001.

This report describes a computer program package that aids in assessing the impact of floods on risk from nuclear power plants. The package consists of two distinct computer programs: ESP and NOAH. The ESP program improves the efficiency of a flood analysis by screening accident sequences and identifying accident sequences that are potentially significant contributors to risk in the event of a flood. Input to ESP includes accident sequences from an existing risk assessment and flood screening criteria. The output from ESP includes (a) accident sequences that are potentially significant contributors to risk, (b) specific plant systems contained in these accident sequences that may require detailed analysis, and (c) a quantitative estimate of the flood contribution to risk. The NOAH program provides detailed qualitative analysis of the plant systems identified by ESP. Input includes, (a) the system fault tree from the existing risk assessment, (b) vulnerability elevations for each component represented in the fault tree, and (c) a detailed flood level profile for the plant.

NUREG/CR-2678: FLOOD RISK ANALYSIS METHODOLOGY DEVELOPMENT PROJECT.Final Report. WAGNER, D.P.; CASADA, M.L.; FUSSELL, J.B. Oak Ridge National Laboratory. July 1982. 121pp. 8209230045. ORNL/TM-8314. 14989:157.

This report is the final documentation of a two-year effort to develop a methodology for assessing the impact of flood on nuclear power plant risk. The methodology presented in this document is consistent with conventional fault tree/event tree risk assessment techniques and is intended for application as a part of an overall probabilistic risk assessment (PRA). The methodology also satisfies many of the requirements for flood analysis identified in the Nuclear Regulatory Commission's recent Draft PRA Procedures Guide (NUREG/CR-2300). The project also resulted in two computer programs that aid in portions of the flood analysis. These computer programs represent a major effort of the project and are fully described in a separate document that is the user's manual for both computer programs. These computer programs are used extensively in the example applications discussed in the appendices of this report. This report assumes the reader is familiar with fault tree/event tree terminology and conventional probabilistic risk assessment techniques.

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-2688: EFFECTS OF DISPERSED PARTICULATE OR DROPLET PHASE ON THE RAYLEIGH-TAYLOR INSTABILITY OF A GAS-LIQUID INTERFACE. MOSZYNSKI, J. R.; GINSBERG, T. Brookhaven National Laboratory. July 1982. 34pp. 8207210022. BNL-NUREG-51533. 13996:153.

The growth of Rayleigh-Taylor instabilities is studied in relation to liquid entrainment at the interface between accelerating fluids of unequal density. The upper fluid is pure liquid, and the lower fluid is a mixture of vapor and heavy dispersed droplet or particulate phase. Entrainment through this mechanism would occur when the liquid spikes grow into the lower fluid and, eventually, separate into droplets. This work estimates the effect of the presence of heavy droplets or particulates in the immediate vicinity of the interface on the early (linear) stages of instability growth.

The growth of the Taylor instability is computed using a porous medium model of the multi-phase lower fluid, which assumes that the dispersed phase is characterized by infinite inertia. The vapor simply flows around the dispersed phase. The model is described and calculation results are presented for the rate of instability growth during HCDA bubble expansion. Results are compared with the classical Taylor theory which neglects the presence of the dispersed phase, and with a homogenious model of the multi-phase bubble.

NUREG/CR-2689: TOM MIX:A COMPUTER CODE FOR CALCULATING STEAM EXPLOSION PHENOMENA. DRUMHELLER, D. S. Sandia Laboratories. July 1982. 69pp. 8209270414. SAND81-2520. 15529:314.

A mathematical model for a mixture of hot liquid drops mixed and in film boiling with water is developed. It is used to examine the propagation and growth of pressure disturbances which result in a steam explosion. The model is incorporated into a wave-propagation computer code and an example problem is examined. These calculations suggest that pressure disturbances will propagate and grow in the mixture even if the drops do not fragment. These pressure pulses initiate fragmentation which ultimately controls the release of energy during a steam explosion.

NUREG/CR-2690: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM. Quarterly Report, October-December 1981. THOMPSON, S. L. Sandia Laboratories. July 1982. 39pp. 8208180241. SAND82-0953. 14389:192.

The RALOC computer program for the analysis of gas transport in subdivided compartments was initially acquired by Sandia in March 1981 from the Gesellschaft fur Reaktorsicherheit. Two sensitivity studies were completed with RALOC this quarter. These studies were performed in order to evaluate the reliability of RALOC and to determine the sensitivity of the code to certain input parameters. Both studies modeled the Grand Gulf Mark III containment building. Parameters varied in the first study included maximum time step allowed during the implicit integration of the differential equations and the degree of discretization used in modeling containment. The parameters in the second study included the amount and rate of hydrogen injection, the injection temperature of the hydrogen, and the initial temperature distribution in the containment.

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-2697: URANIUM OXIDE AND SODIUM OXIDE AEROSOL EXPERIMENTS: NSSP MIXED-OXIDE TESTS 303-307, DATA RECORD REPORT. ADAMS, R. E.; KRESS, T. S.; TOBIAS, M. L. Oak Ridge National Laboratory. November 1982. 108pp. 8212130135. ORNL/TM-8325. 16417:258.

This data record report summarizes five tests, involving mixtures of uranium oxide and sodim oxide aerosols, conducted in the Nuclear Safety Pilot Plant project at Oak Ridge National Laboratory. The goal of this project is to establish the validity (or level of conservatism) of the aerosol behavioral code, HAARM-3, and follow-on codes under development at Battelle Columbus Laboratories for the U.S. Nuclear Regulatory Commission. Descriptions of the five tests with tables and graphs summarizing the results are included.

NUREG/CR-2698: AEROSOL CHARACTERIZATION FROM A SIMULATED HCDA: October 1976-October 1981. ZANOTELLI, W. A.; ROESCH, D. L.; MILLER, G. D. Mound Laboratory/Monsanto. August 1982. 58pp. 8209280315. MLM-2927. 15532:284.

In support of the Nuclear Regulatory Commission, Mound Facility completed a program to produce and characterize the primary aerosols that could result from a hypothetical core-disruptive accident (HCDA) in a liquid metal fast breeder reactor (LMFBR). These tests were conducted under simulated conditions that can be related to the actual events that would occur under such a critical excursion.

The two main objectives of the study were: 1) the investigation of the chemical reactions and compounds formed from the short-lived vapor state existing in the HCDA event, and 2) the acquisition of a thermodynamic data base for the possible reactions and species formed from the interactions of the U-Pu-O-Na system.

The simulation of the HCDA event was accomplished by the laser evaporation technique.

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-2701: AN ASSESSMENT OF FUEL FOAMING POTENTIAL DURING CORE MELTDOWN ACCIDENTS. CRONENBERG, A.W.; CROUCHER, P.W.; MACDONALD, P.E. EG&G, Inc. November 1982. 61pp. 8212130113. EGG-2191. 16417:168.

Fuel melting during severe core damage accidents can be expected to lead to the rapid release of fission gas from the fuel matrix and the volatilization of low boiling point metallic inclusions, which can significantly influence molten fuel swelling/foaming behavior. A quantitative analysis of UO (2) foaming potential is therefore presented, based on an assessment of the time characteristics of bubble growth, surface escape, film thinning, and bubble coalescense. Analysis indicates that although the potential exists for early molten UO (2) foaming, such foams are basically unstable and tend to collapse, thereby releasing volatilized fission products from the molten fuel debris. Release of such fission products will impact not only the course of melt progression during a meltdown accident but also the radiological source term, and can result in as much as a 30 to 40% reduction in the residual debris decay heat level. In addition, such foam collapse indicates that the molten core debris can be expected to be relatively dense, which will impact debris cooling characteristics.

NUREG/CR-2704: U.S. REACTOR SPENT-FUEL STORAGE CAPABILITIES. LEE, W.J.; HOFFMAN, C.C.; CAVINESS, C.K. Nuclear Assurance Corp. 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-2705: TRAINING COURSE NO 1: THE IMPLEMENTATION OF FEMWATER (ORNL-5567) COMPUTER PROGRAM. YEH, G.T. Oak Ridge National Laboratory. June 1982. 169pp. 8208180222. ORNL/TM-8327. 14389:023.

This report documents a training course conducted for the U.S.

Nuclear Regulatory Commission (NRC) on the implementation of a Finite Element Model of WATER flow throuh saturated-unsaturated porous media (FEMWATER) - ORNL-5567. In addition to presenting basic program operation (APPENDIX A-V), the course also covered the following topics: (1) Mathemtical equations and physical principles that lead to the code development (APPENDIX A-I), (2) The finite element method (APPENDIX A-II), (3) Finite-element derivation of FEMWATER (APPENDIX A-III), (4) FEMWATER program structure (APPENDIX A-IV), (5) Uniqueness and limitations of FEMWATER (APPENDIX A-VI), (6) Running of four sample problems (APPENDIX B-B) to demonstrate various options that FEMWATER can handle. The purpose of the training seminar is to enable the NRC staff to use the model (and to be able to modify the code if necessary) for checking information provided by a licensee, for evaluating alternative sites and designs for burial, and for comparing their results from other methods of solution.

NUREG/CR-2706: TRAINING COURSE NO 2: THE IMPLEMENTATION OF FEMWASTE (ORNL-5601) COMPUTER PROGRAM. YEH, G. T. Oak Ridge National Laboratory. November 1982. 136pp. 8302030252. ORNL/TM-8328. 17023:018.

This report documents a training course conducted for the U.S. Nuclear Regulatory Commission (NRC) on the implementation of a Finite Element Model of WASTE transport through saturated-unsaturated porous media (FEMWASTE) - ORNL-5601. In addition to presenting basic program operations (Appendices A-V through A-VII), the course also covers the following topics: (1) Heuristic derivation of governing equations based on physical and chemical principles (APPENDIX A-I), (2) finite element derivation of FEMWASTE (APPENDIX A-II), (3) various numerical schemes provided by FEMWASTE (APPENDIX A-III), (4) FEMWASTE program structure (APPENDIX A-IV), and (5) running the three samples problems (APPENDIX B) to demonstrate various options that FEMWASTE can handle. The purpose of the training seminar is to enable NRC staff to use the model (and to be able to modify the code, if necessary) for checking information provided by a licensee, for evaluating alternative sites and designs for burial, and for comparing their results from other methods of solution.

NUREG/CR-2707: ORNL SMALL-BREAK LOCA HEAT TRANSFER TEST SERIES I: COMPARISONS OF EXPERIMENTAL DATA WITH VENDOR MODELS FOR LOW-FLOW STEAM HEAT TRANSFER. ANKLAM, T. M. Oak Ridge National Laboratory. July 1982. 61pp. 8208260080. ORNL/TM-8329. 14577:320.

Comparisons are presented of data generated during the first Oak Ridge National Laboratory Small-Break Loss-of-Coolant Accident Heat Transfer Test Series with predictions of reactor vendor heat transfer correlations. Vendor correlations were found to predict experimental heat transfer coefficients relatively well. Standard errors of fit ranged from 12 to 18% for turbulent flow and were approximately 24% for transition-to-turbulent flow. Despite reasonable overall agreement, several of the correlations do not predict temperature ratio trends in the data.

NUREG/CR-2708: PRELIMINARY NEUTRONICS CALCULATIONS FOR THE SUPER SARA TEST PROGRAM. JENQUIN, U. P. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1982. 48pp. 8301190446. PNL-4268. 16849:033.

The Super SARA Test Program (SSTP) will simulate small-break loss-of-coolant accidents events in light-water reactors. Neutronics analyses have been performed in support of the SSTP in ESSOR reactor. This report contains calculated power distributions for uniformly enriched UO (2) test assemblies and power coupling factors for the test assemblies relative to the driver assemblies.

NUREG/CR-2709: CRITICAL EXPERIMENTS WITH 4.31 WT% U-235 ENRICHED U-DIOXIDE IN HIGHLY BORATED WATER LATTICES. DURST, B. M. ; BIERMAN, S. R. ; CLAYTON, E. D. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1982. 25pp. 8209130292. PNL-4267. 14781:303.

A series of critical experiments were performed with 4.31 wt% (235)U enriched UO(2) fuel rods immersed in water containing various concentrations of boron ranging up to 2.55 g/l. The boron was added in the form of boric acid (H(3)BO(3)). Critical experiment data were obtained for two different lattice pitches wherein the water-to-uranium oxide volume ratios were 1.59 and 1.09. The experiments provide benchmarks on heavily borated systems for use in validating calculational techniques employed in analyzing fuel shipping casks and spent fuel storage systems that may utilize boron for criticality control.

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-2712: ANALYSIS OF TRAINING AND CERTIFICATION OF OPERATIONS TECHNICIANS AT INDEPENDENT SPENT FUEL STORAGE INSTALLATIONS. HOTTMAN, S. B. ; BATEMAN, R. P. ; BIERS, D. W. Sandia Laboratories. August 1982. 317pp. 8208130479. SAND82-7051. 14355:001.

This doument presents the results of a task analysis and recommendations for the training and certification of operations technicians at independent spent fuel storage installations. Its purpose is to provide a technical basis for initial and continuation training for operations technicians at Independent Spent Fuel Storage Installations (ISFSIs). It also provides guidance for testing operations technicians to ensure that training objectives have been achieved. The recommended testing provides a basis for certification of ISFSI operators. The basis for this handbook was a task analysis conducted at the ISFSI at Morris, Illinois. Supervisors were interviewed and a preliminary job analysis was used to determine required operator skills. Training, safety and operating documents and checklists were reviewed and task inventory forms were developed with the help of ISFSI supervisors. Operations Technicians were then interviewed and the task inventory forms filled out with information on task frequency, difficulty, hazard, time to complete and error potential. These data were analyzed to determine required operator skills and proficiency levels necessary for safe ISFSI operations.

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-2714: ANALYSIS OF FAILURE RATE DATA WITH COMPOUND MODELS. SHULTIS,J.K.; ECKHOFF,N.D.; JOHNSON,D.E.; et al. Kansas State Univ. July 1982. 115pp. 8208260097. EGG-2206. 14577:017.

One of the most useful statistical models to describe the failure of components with inherently low failure rates is the compound or Bayesian model. In this model the failure rate of a component is assumed to vary among all similar components according to some usually unknown prior distribution. The Poisson distribution, which gives the probability of having F component failures in operation time T for a given failure rate, is then averaged over all possible failure rates described by the prior distribution to obtain the compound or marginal distribution.

There are many procedures for determining the explicit form of the prior distribution that describes the failure rate variation among a given set of components. In this study, the prior distribution for a set of components was estimated from past failure data of the components in question by first assuming a functional form for the prior distribution (i.e., a prior family), and then estimating values of the prior parameters from observed failure data. Three methods were used to estimate values of the prior parameters: (1) matching the data moments to those of the prior distribution (PMM), (2) matching the moments of the marginal distribution to those of the data (MMMM), and (3) the maximum likelihood method based on the marginal distribution (MMLM).

NUREG/CR-2716 V01: REACTOR SAFETY RESEARCH PROGRAMS. Quarterly Report, January-March 1982. EDLER, S. K. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1982. 69pp. 8208170306. PNL-4275-1. 14371:305.

This document summarizes work performed by Pacific Northwest Laboratory (PNL) from January 1 through March 31, 1982, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission (NRC). Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining the strength of structural graphite, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. Experimental data and analytical models are being provided to aid in decision-making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Core thermal models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Fuel assemblies and analytical support are being provided for experimental programs at other facilities, including loss-of-coolant accident (LOCA) simulation tests at the NRU reactor, Chalk River, Canada; fuel rod deformation, severe fuel damage, and postaccident coolability tests for the ESSOR reactor Super Sara Test Program, Ispra, Italy; the instrumented fuel assembly irradiation program at Halden, Norway; and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory (INEL), Idaho Falls, Idaho. These programs will provide data for computer modeling of reactor system and fuel performance during various abnormal operating conditions.

NUREG/CR-2716 V02: REACTOR SAFETY RESEARCH PROGRAMS. Quarterly Report, April - June 1982. EDLER, S. K. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1982. 70pp. 8212270449. PNL-4275-2. 16566:229.

This document summarizes work performed by Pacific Northwest Laboratory (PNL) from April 1 through June 30, 1982 for the Division of Accident Evaluation and the Division of Engineering Technology. Results of Graphite NDT research (eddy current & ultrasonic testing and oxidation) are reported. Acoustic Emission monitoring of the Watts Bar reactor and pipe material evaluation testing are reported. Progress on the reliability of ultrasonic ISI techniques is reported. Fission gas release and PIE data for fuel rods irradiated in the HBWR (Norway) is reported. Progress on the pipe-to-pipe impact program is reported. Progress on the SFD test subassembly procurement program is reported. Results of calculations with the COBRA/TRAC computer code are reported. Results of the MT-4 LOCA experiment conducted in the NRU reactor are reported. Progress on the examination of the Surry steam generator is reported.

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-2719: EVALUATION OF INSERVICE INSPECTIONS OF GREASED PRESTRESSING TENDONS. DOUGAN, J.R. Oak Ridge National Laboratory. September 1982. 76pp. 8210150569. ORNL/TM-8278. 15716:169.

The Nuclear Regulatory Commission (NRC) Regulatory Guide 1.35 (Rev. 2), contains guidelines for sample selection, visual inspection of anchorage components, monitoring of prestress force, analysis of the grease for impurities, and tensile testing of tendon material. Due to varying interpretation by utilities of the current version, the NRC has issued, on a "for comment" basis, a proposed third revision to Regulatory Guide 1.35 along with a proposed companion guide, Regulatory Guide 1.35.1. This study analyzes the available utility surveillance data and evaluates the current and proposed guidelines. Comments from utility and industry personnel were factored into the analysis. The results indicated that the majority of the few incidences of problems or abnormalities that occurred were minor in nature and did not threaten the structural integrity of the containment. All the available surveillance reports concluded that the respective containments were in good condition. However, the surveillance reports did reveal discrepancies between utilities in the interpretation of results. Evaluation of the proposed guidelines also revealed areas that could be modified to provide improved results; that is, some consideration should be given to modifying the sample size, the tendon force levels should be evaluated both as a group and individually, and the acceptable levels of impurities in the grease should be reexamined as more data become available.

NUREG/CR-2720: ANALYSIS OF PERFORMANCE OF WESTINGHOUSE REACTOR VESSEL LEVEL INDICATING SYSTEM FOR TESTS AT SEMISCALE. HARDY, J.E.; MILLER, G.N. Oak Ridge National Laboratory. November 1982. 61pp. 8212130123. ORNL/TM-8336. 16410:220.

The Westinghouse Reactor Vessel Level Indicating System (RVLIS), a differential pressure level measurement system, was tested at SEMISCALE. This report contains the analyses of these tests and the conclusions of these analyses. The tests performed included small break and intermediate break tests. Also, frequency response and natural circulation tests were run and analyzed. The RVLIS always indicated a level less than the two phase froth level. The RVLIS output in early small break tests indicated a level 200 cm greater than actual collapsed liquid level. This discrepancy was caused by structural differences between SEMISCALE and a Westinghouse reactor. Once modifications were made so that SEMISCALE better simulated a Westinghouse PWR, the maximum difference between RVLIS and SEMISCALE instrumentation was 30 cm or 3% which is less than the stated uncertainty of the Westinghouse RVLIS.

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-2723: ESTIMATES OF THE FINANCIAL CONSEQUENCES OF NUCLEAR POWER REACTOR ACCIDENTS. STRIP, D. R. Sandia Laboratories. November 1982. 180pp. 8212220204. SAND82-1110. 16527:156.

This report develops preliminary techniques for estimating the financial consequences of potential nuclear power reactor accidents. Offsite cost estimates are based on CRAC2 calculations. Costs are assigned to health effects as well as property damage. Onsite costs are estimated for worker health effects, replacement power, and cleanup costs. Several classes of costs are not included, such as indirect costs, socio-economic costs, and health care costs. Present value discounting is explained and then used to calculate the life cycle cost of the risks of potential reactor accidents. Results of the financial consequence estimates for 156 reactor-site combinations are summarized, and detailed estimates are provided in an appendix. The results indicate that, in general, onsite costs dominate the consequences of potential accidents.

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-2727 V02: ECOLOGICAL STUDIES OF WOOD-BORING BIVALVES IN THE VICINITY OF THE OYSTER CREEK NUCLEAR GENERATING STATION. Progress Report, December 1981-Februaury 1982. HOAGLAND, K. E.; CROCKET, L. Academy of Natural Sciences. August 1982. 38pp. 8209230056. 14982:275.

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 tolerance of 3 species are also under investigation in the laboratory. Competition among the species is being analyzed. In the winter of 1981, the generating station experienced a prolonged outage. The reproductive cycle of the shipworms was not extended. Teredo bartschi was very abundant at one station in Oyster Creek and moderately abundant at a second, but did not exist elsewhere in Barnegat Bay. Some specimens of Teredo bartscho contained larvae in the gills in February. According to laboratory experiments, Teredo navalis is able to remain active at temperatures as low as 4 degrees C, whereas T. bartschi ceases activity (withdraws its siphons) at about 13 degrees C.

NUREG/CR-2727 V03: ECOLOGICAL STUDIES OF WOOD-BORING BIVALVES IN THE VICINITY OF THE OYSTER CREEK NUCLEAR GENERATING STATION. Progress Report, March - May 1982. HOAGLAND, K. E. Academy of Natural Sciences of Philadelphia. November 1982. 43pp. 8212270480. 16548:234.

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. The adult population of Teredo bartschi survived the winter and spring of 1981-82 better than it did previous cold periods without a thermal effluent. Lack of an effluent was due to a prolonged outage of the generating station. There was no spring outbreak of shipworms. The introduced species appears established at one station near but outside of Oyster Creek. Three teredinid species coexist in Oyster Creek. Larvae of T. bartschi and T. navalis have similar responses to reduced salinity. Bankia gouldi is the fastest-growing of the teredinids found in New Jersey, and has the lowest annual mortality.

NUREG/CR-2727 V04: ECOLOGICAL STUDIES OF WOOD-BORING BIVALVES IN THE VICINITY OF THE OYSTER CREEK NUCLEAR GENERATING STATION. Docket No. 50-219. (General Public Utilities) HOAGLAND, K. E. Academy of Natural Sciences of Philadelphia. December 1982. 51pp. 8301100018. 16719:218.

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. Adult populations of Teredo bartschi existed in both Oyster Creek and Forked River in the summer of 1982, but the species was rare. There was no large settlement of this or any other teredinid species in Barnegat Bay. Teredo navalis was the most common species in the monthly panels. The fouling community reached its maximum yearly diversity in June-July. There was a thermal effluent causing a delta T of 3-4 degrees Centigrade during most of the summer, and salinity in Oyster Creek and Forked River was similr to that of Barnegat Bay. The lack of shipworm outbreak in 1982 may be related to the low delta T in summer, plus the lack of a thermal effluent in the preceding winter-spring period.

NUREG/CR-2730: HYDROGEN BURN SURVIVAL:PRELIMINARY THERMAL MODEL AND TEST RESULTS. MCCULLOCH,W.H.; RATZEL,A.C.; KEMPKA,S.N.; et al. Sandia Laboratories. September 1982. 36pp. 8210150571. SAND82-1150. 15716:132.

This report documents preliminary Hydrogen Burn Survival (HBS) Program experimental and analytical work conducted through February 1982. The effects of hydrogen deflagrations on safety-related equipment in nuclear power plant containment buildings are considered. Preliminary results from hydrogen deflagration experiments in the Sandia Variable Geometry Experimental System (VGES) are presented and analytical predictions for these tests are compared and discussed. Analytical estimates of component thermal responses to hydrogen deflagrations in the upper and lower compartments of an ice condenser, pressurized water reactor are also presented.

NUREG/CR-2731: AN EVALUATION OF THE SAFETY ASPECTS OF THE DESIGN AND OPERATION OF TEMPORARY/MOBILE RADIOACTIVE WASTE SOLIDIFICATION SYSTEMS. MCDONALD,F.N.; MCLURE,L.W. Exxon Nuclear Co., Inc. (subs. of Exxon Corp.). August 1982. 43pp. 8209130284. 14781:173.

An evaluation of the safety aspects of the design and operation of temporary/mobile radioactive waste solidification systems in use at commercial nuclear power reactors was completed. The study was undertaken in response to a General Accounting Office report issued in August 1978, calling for "more regulatory oversight of commercial low-level radioactive waste treatment." After reviewing the design and operation of three different vendor-operated waste solidification systems, it is clear that there are areas in which the vendors can improve their services. However, the vendors generally do a good job of solidifying waste in a manner that is well controlled and safe.

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-2733: EXPERIMENT DATA REPORT FOR LOFT BORON DILUTION EXPERIMENT L6-6. STITT, B. D.; DEVINE, J. M. EG&G, Inc. July 1982. 139pp. 8207220001. EGG-2197. 14025:010.

Selected pertinent and uninterpreted data from the sixth anticipated transient experiment (Experiment L6-6) 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 thermal-hydraulic conditions during a postulated loss-of-coolant accident. The operation of the LOFT system is typical of large [approximately 1000 MW(e)], commercial PWR operations. Experiment L6-6 simulated a boron dilution accident by injecting demineralized water into the primary coolant system (PCS) at a rate of 0.47 L/s while the reactor was in cold shutdown condition with the control rods withdrawn. System pressure was maintained at approximately 285 kPa throughout the experiment. The experiment was divided into two parts. In the first part, L6-6A, a recirculation flow of 4.7 L/s was maintained through the PCS and criticality was achieved 7416 plus or minus 10 s after the initiation of the dilution flow. The second part, L6-6B, was identical to L6-6A except that a recirculation flow of 9.5 L/s was maintained and criticality occurred at 8085 plus or minus 10 s.

NUREG/CR-2734: AN ASSESSMENT OF FUEL ROD FAILURE PROPAGATION IN LIGHT WATER REACTORS: MODEL DEVELOPMENT, CALCULATIONAL RESULTS AND CONCLUSIONS. WEHNER, T. R.; TOMKINS, J. L.; BAARS, R. E.; et al. Los Alamos Scientific Laboratory. October 1982. 218pp. 8211190334. LA-9370-MS. 16169:084.

This report summarizes analyses of fuel rod failure propagation (FFP) during normal and off-normal conditions in commercial light water reactors (LWRs). FFP is defined to occur when the failure of one fuel rod causes the failure of an adjacent rod, which causes the failure of another rod adjacent to it, etc. Potential FFP initiators are examined, including gas ejection from a failed fuel rod, molten and solid fuel release from a failed fuel rod, during normal and off-normal conditions, and local coolant flow constrictions of channel blockage, rod ballooning, and rod bowing. FFP induced by departure-from-nucleate-boiling (DNB) propagation is also discussed.

Results for gas ejection and limited fuel release from a failed rod indicate no FFP potential. If a significant fraction of the fuel inventory from a failed rod should desinter into fine particles that simultaneously interact with the coolant, then FFP could be possible. Because cladding ruptures large enough to permit a large-scale interaction have not been observed in commercial reactor experience, FFP is still very unlikely. The data base relevant to DNB propagation is sparse; however, the fact that DNB propagation has never been observed strengthens the theoretical arguments against such a propagation mode.

NUREG/CR-2735: 2D/3D ANALYSIS PROGRAM REPORT-1981. KIRCHNER, W. L.; WILLIAMS, K. A. Los Alamos Scientific Laboratory. October 1982. 61pp. 8211110650. LA-9376-PR. 16050:181.

The United States Nuclear Regulatory Commission (USNRC) is currently engaged in a multinational experimental and analytical research program (known as 2D/3D) on multidimensional thermal-hydraulic behavior during loss-of-coolant accidents (LOCAs) in large pressurized water reactors (PWRs). This report documents the key results and findings from efforts during FY 1981. The Transient Reactor Analysis Code (TRAC), the main analytical tool in this program, was demonstrated to be a powerful tool for reactor safety analysis. By correctly predicting key results from the experimental test facilities over a wide range of test conditions, a significant level of confidence in the code was obtained. Complementary TRAC analyses of postulated PWR accidents show substantial safety margins below current licensing requirements. Future model development and assessment activities for TRAC are outlined and future 2D/3D activities of importance to NRC licensing activities are also described.

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. 117pp. 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-2738: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A INTERMEDIATE BREAK TEST SERIES (Test S-IB-3). SACKETT, K. E.; CLEGG, L. B. EG&G, Inc. July 1982. 52pp. 8207260002. EGG-2198. 14068:358.

This report presents test data recorded for Test S-IB-3 of the Semiscale Mod-2A Intermediate Break Test Series. This test is one of a series of Semiscale tests investigating thermalhydraulic 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. The primary objective of Test S-IB-3 was to provide reference data for comparison of Semiscale test results with those from the LOB1 B-RIM test conducted in the Loop Blowdown Investigation Facility at Ispra, Italy. Test S-IB-3 was conducted from initial conditions closely approximating the specified initial conditions, which were: system pressure, 15.5 MPa; cold leg temperature, 563 K; and core power level, 1.44 MW. This report presents the uninterpreted data from the test 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-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-2749 V01: SOCIOECONOMIC IMPACTS OF NUCLEAR GENERATING STATIONS: ARKANSAS NUCLEAR ONE STATION CASE STUDY. Docket Nos. 50-313 and 50-368. (Arkansas Power And Light Company) PIJAWKA, K. D. Mountain West Research, Inc. July 1982. 193pp. 8208090021. 14304:332.

This report documents a case study of the socioeconomic impacts of the construction and operation of the Arkansas One nuclear power station. It is part of a major post-licensing study of the socioeconomic impacts at twelve nuclear power stations. The case study covers the period beginning with the announcement of plans to construct the reactor and ending in the period 1980-81. The case study deals with changes in the economy, population, settlement patterns and housing, local government and public services, social structure, and public response in the study area during the construction/operation of the reactor.

A regional modeling approach is used to trace the impact of construction/operation on the local economy, labor market, and housing market. Emphasis in the study is on the attribution of socioeconomic impacts to the reactor or other causal factors. As part of the study of local public response to the construction/operation of the reactor, the effects of the Three Mile Island accident are examined.

NUREG/CR-2749 V02: SOCIOECONOMIC IMPACTS OF NUCLEAR GENERATING STATIONS: CALVERT CLIFFS CASE STUDY. Docket Nos. 50-317 And 50-318. (Baltimore Gas And Electric Company) FLYNN, J. Social Impact Research, Inc. * Mountain West Research, Inc. July 1982. 219pp. 8208040323. 14230:001.

This report documents a case study of the socioeconomic impacts of the construction and operation of the Calvert Cliffs nuclear power station. It is part of a major post-licensing study of the socioeconomic impacts at twelve nuclear power stations. The case study covers the period beginning with the announcement of plans to construct the reactor and ending in the period 1980-81. The case study deals with changes in the economy, population, settlement patterns and housing, local government and public services, social structure, and public response in the study area during the construction/operation of the reactor.

A regional modeling approach is used to trace the impact of construction/operation on the local economy, labor market, and housing market. Emphasis in the study is on the attribution of socioeconomic impacts to the reactor or other causal factors. As part of the study of local public response to the construction/operation of the reactor, the effects of the Three Mile Island accident are examined.

NUREG/CR-2749 V03: SOCIOECONOMIC IMPACTS OF NUCLEAR GENERATING STATIONS: CRYSTAL RIVER UNIT 3 CASE STUDY. Docket No. 50-302. (Florida Power Corporation) BERGMAN, P.A. Mountain West Research, Inc. * Social Impact Research, Inc. July 1982. 237pp. 8208040215. 14234:225.

This report documents a case study of the socioeconomic impacts of the construction and operation of the Crystal River Unit 3 nuclear power station. It is part of a major post-licensing study of the socioeconomic impacts at twelve nuclear power stations. The case study covers the period beginning with the announcement of plans to construct the reactor and ending in the period 1980-81. The case study deals with changes in the economy, population, settlement patterns and housing, local government and public services, social structure, and public response in the study area during the construction/operation of the reactor.

A regional modeling approach is used to trace the impact of construction/operation on the local economy, labor market, and housing market. Emphasis in the study is on the attribution of socioeconomic impacts to the reactor or other causal factors. As part of the study of local public response to the construction/operation of the reactor, the effects of the Three Mile Island accident are examined.

NUREG/CR-2749 V04: SOCIOECONOMIC IMPACTS OF NUCLEAR GENERATING STATIONS: D.C. COOK CASE STUDY. Docket Nos. 50-315 And 50-316. (Indiana And Michigan Electric Company) BRANCH, K. Mountain West Research, Inc. * Social Impact Research, Inc. July 1982. 274pp. 8208090019. 14304:058.

This report documents a case study of the socioeconomic impacts of the construction and operation of the D. C. Cook nuclear power station. It is part of a major post-licensing study of the socioeconomic impacts at twelve nuclear power stations. The case study covers the period beginning with the announcement of plans to construct the reactor and ending in the period 1980-81. The case study deals with changes in the economy, population, settlement patterns and housing, local government and public services, social structure, and public response in the study area during the construction/operation of the reactor.

A regional modeling approach is used to trace the impact of construction/operation on the local economy, labor market, and housing market. Emphasis in the study is on the attribution of socioeconomic impacts to the reactor or other causal factors. As part of the study of local public response to the construction/operation of the reactor, the effects of the Three Mile Island accident are examined.

NUREG/CR-2749 V05: SOCIOECONOMIC IMPACTS OF NUCLEAR GENERATING STATIONS: DIABLO CANYON CASE STUDY. Docket Nos. 50-275 And 50-323. (Pacific Gas And Electric Company) PIJAWKA, K. D. ; YAGUINTO, G. Mountain West Research, Inc. * Social Impact Research, Inc. July 1982. 214pp. 8208170010. 14374:106.

This report documents a case study of the socioeconomic impacts of the construction and operation of the Diablo Canyon nuclear power station. It is part of a major post-licensing study of the socioeconomic impacts at twelve nuclear power stations. The case study covers the period beginning with the announcement of plans to construct the reactor and ending in the period 1980-81. The case study deals with changes in the economy, population, settlement patterns and housing, local government and public services, social structure, and public response in the study area during the construction/operation of the reactor.

A regional modeling approach is used to trace the impact of construction/operation on the local economy, labor market, and housing market. Emphasis in the study is on the attribution of socioeconomic impacts to the reactor or other causal factors. As part of the study of local public response to the construction/operation of the reactor, the effects of the Three Mile Island accident are examined.

NUREG/CR-2749 V06: SOCIOECONOMIC IMPACTS OF NUCLEAR GENERATING STATIONS: NINE MILE PLANT AND FITZPATRICK CASE STUDY. Docket Nos. 50-220, 50-410 & 50-333. (Niagara Mohawk Power Corporation And Power Authority Of The State of New York) BRANCH, K. ; MEALE, R. Mountain West Research, Inc. *; et al. Social Impact Research, Inc. July 1982. 288pp. 8208130115. 14341:001.

This report documents a case study of the socioeconomic impacts of the construction and operation of the Nine Mile Point and Fitzpatrick nuclear power stations. It is part of a major post-licensing study of the socioeconomic impacts at twelve nuclear power stations. The case study covers the period beginning with the announcement of plans to construct the reactor and ending in the period 1980-81. The case study deals with changes in the economy, population, settlement patterns and housing, local government and public services, social structure, and public response in the study area during the construction/operation of the reactor.

A regional modeling approach is used to trace the impact of construction/operation on the local economy, labor market, and housing market. Emphasis in the study is on the attribution of socioeconomic impacts to the reactor or other causal factors. As part of the study of local public response to the construction/operation of the reactor, the effects of the Three Mile Island accident are examined.

NUREG/CR-2749 V07: SOCIOECONOMIC IMPACTS OF NUCLEAR GENERATING STATIONS: OCONEE CASE STUDY. Docket Nos. 50-269 And 50-270. (Duke Power Company) FLYNN, J. Social Impact Research, Inc. * Mountain West Research, Inc. July 1982. 194pp. 8208040311. 14229:001.

This report documents a case study of the socioeconomic impacts of the construction and operation of the Oconee nuclear power station. It is part of a major post-licensing study of the socioeconomic impacts at twelve nuclear power stations. The case study covers the period beginning with the announcement of plans to construct the reactor and ending in the period 1980-81. The case study deals with changes in the economy, population, settlement patterns and housing, local government and public services, social structure, and public response in the study area during the construction/operation of the reactor.

A regional modeling approach is used to trace the impact of construction/operation on the local economy, labor market, and housing market. Emphasis in the study is on the attribution of socioeconomic impacts to the reactor or other causal factors. As part of the study of local public response to the construction/operation of the reactor, the effects of the Three Mile Island accident are examined.

NUREG/CR-2749 V08: SOCIOECONOMIC IMPACTS OF NUCLEAR GENERATING STATIONS: PEACH BOTTOM CASE STUDY. Docket Nos. 50-277 And 50-278. (Philadelphia Electric Company) PIJAWKA, K. D. Mountain West Research, Inc. * Social Impact Research, Inc. July 1982. 217pp. 8208060357. 14272:143.

This report documents a case study of the socioeconomic impacts of the construction and operation of the Peach Bottom nuclear power station. It is part of a major post-licensing study of the socioeconomic impacts at twelve nuclear power stations. The case study covers the period beginning with the announcement of plans to construct the reactor and ending in the period, 1980-81. The case study deals with changes in the economy, population, settlement patterns and housing, local government and public services, social structures, and public response in the study area during the construction/operation of the reactor.

A regional modeling approach is used to trace the impact of construction/operation on the local economy, labor market, and housing market. Emphasis in the study is on the attributions of socioeconomic impacts to the reactor or other causal factors. As part of the study of local public response to the construction/operation of the reactor, the effects of the Three Mile Island accident are examined.

NUREG/CR-2749 V09: SOCIOECONOMIC IMPACTS OF NUCLEAR GENERATING STATIONS: RANCHO SECO CASE STUDY. Docket No. 50-312. (Sacramento Municipal Utility District) BERGMAN, P. A. Mountain West Research, Inc. * Social Impact Research, Inc. July 1982. 227pp. 8208040208. 14235:102.

This report documents a case study of the socioeconomic impacts of the construction and operation of the Rancho Seco nuclear power station. It is part of a major post-licensing study of the socioeconomic impacts at twelve nuclear power stations. The case study covers the period beginning with the announcement of plans to construct the reactor and ending in the period 1980-81. The case study deals with changes in the economy, population, settlement patterns and housing, local government and public services, social structure, and public response in the study area during the construction/operation of the reactor.

A regional modeling approach is used to trace the impact of construction/operation on the local economy, labor market, and housing market. Emphasis in the study is on the attribution of socioeconomic impacts to the reactor or other causal factors. As part of the study of local public response to the construction/operation of the reactor, the effects of the Three Mile Island accident are examined.

NUREG/CR-2749 V10: SOCIOECONOMIC IMPACTS OF NUCLEAR GENERATING STATIONS: SAINT LUCIE CASE STUDY. Docket Nos. 50-335 And 50-389. (Florida Power And Light Company) WEISIGER, M. L.; PIJAWKA, K. D. Mountain West Research, Inc. * Social Impact Research, Inc. July 1982. 304pp. 8208090014. 14303:114.

This report documents a case study of the socioeconomic impacts of the construction and operation of the St. Lucie nuclear power station. It is part of a major post-licensing study of the socioeconomic impacts at twelve nuclear power stations. The case study covers the period beginning with the announcement of plans to construct the reactor and ending in the period 1980-81. The case study deals with changes in the economy, population, settlement patterns and housing, local government and public services, social structure, and public response in the study area during the construction/operation of the reactor.

A regional modeling approach is used to trace the impact of construction/operation on the local economy, labor market, and housing market. Emphasis in the study is on the attribution of socioeconomic impacts to the reactor or other causal factors. As part of the study of local public response to the construction/operation of the reactor, the effects of the Three Mile Island accident are examined.

NUREG/CR-2749 V11: SOCIOECONOMIC IMPACTS OF NUCLEAR GENERATING STATIONS: SURRY CASE STUDY. Docket Nos. 50-280 And 50-281. (Virginia Electric Power And Light Company) FLYNN, J. Social Impact Research, Inc. * Mountain West Research, Inc. July 1982. 182pp. 8208040245. 14233:001.

This report documents a case study of the socioeconomic impacts of the construction and operation of the Surry nuclear power station. It is part of a major post-licensing study of the socioeconomic impacts at twelve nuclear power stations. The case study covers the period beginning with the announcement of plans to construct the reactor and ending in the period 1980-81. The case study deals with changes in the economy, population, settlement patterns and housing, local government and public services, social structure, and public response in the study area during the construction/operation of the reactor.

A regional modeling approach is used to trace the impact of construction/operation on the local economy, labor market, and housing market. Emphasis in the study is on the attribution of socioeconomic impacts to the reactor or other causal factors. As part of the study of local public response to the construction/operation of the reactor, the effects of the Three Mile Island accident are examined.

NUREG/CR-2749 V12: SOCIOECONOMIC IMPACTS OF NUCLEAR GENERATING STATIONS: THREE MILE ISLAND CASE STUDY. Docket Nos. 50-289 And 50-320. (Metropolitan Edison, et al.) FLYNN, C. Social Impact Research, Inc. * Mountain West Research, Inc. July 1982. 239pp. 8208090010.

14302:168.

This report documents a case study of the socioeconomic impacts of the construction and operation of the Three Mile Island nuclear power station. It is part of a major post-licensing study of the socioeconomic impacts at twelve nuclear power stations. The case study covers the period beginning with the announcement of plans to construct the reactor and ending in the period 1980-81. The case study deals with changes in the economy, population, settlement patterns and housing, local government and public services, social structure, and public response in the study area during the construction/operation of the reactor.

A regional modeling approach is used to trace the impact of construction/operation on the local economy, labor market, and housing market. Emphasis in the study is on the attribution of socioeconomic impacts to the reactor or other causal factors. As part of the study of local public response to the construction/operation of the reactor, the effects of the Three Mile Island accident are examined.

NUREG/CR-2750: SOCIOECONOMIC IMPACTS OF NUCLEAR GENERATING STATIONS. Summary Report On The NRC Post-Licensing Studies. CHALMERS, D. ; PIJAWKA, K. D. ; BRANCH, K. ; et al. Mountain West Research, Inc. July 1982. 244pp. 8208130473. 14352:194.

The Post-Licensing Studies had four objectives. The first was to identify the socioeconomic effects resulting from the construction and operation of each of twelve nuclear power stations. The socioeconomic variables examined included: economic, demographic, housing, government, public response, and social organization characteristics. The second objective was to determine the way in which the identified effects were evaluated by study area groups. The third objective was to identify the determinants of the project-related effects. This task required knowledge of what combination of site, project, or other determinants was responsible for the project-related effects and for the evaluation of the effects. The fourth objective was to make recommendations with respect to assessment methodologies that could best be used to project the socioeconomic effects of the construction and operation of proposed nuclear generating stations. The objectives of the Post-Licensing Studies are met by the twelve individual case studies and by the Summary Report. The case studies identified the nuclear power stations and describe the evaluation of the effects by area residents. The Summary Report describes the collective findings of the individual case studies, compares the findings across sites to identify possible determinants of the effects, and examines the implication of the findings for future siting decisions and for the methodology most appropriate for projective assessments.

NUREG/CR-2751 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM. Quarterly Progress Report For January-March 1982. WHITMAN, G. D. ; BRYAN, R. H. Oak Ridge National Laboratory. August 1982. 139pp. 8209270436. ORNL/TM-8369/V1. 15531:193.

The Heavy-Section Steel Technology 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. The three-dimensional finite-element program for elastic-plastic fracture mechanics was used with a deformation plasticity model on test vessel analysis. Subcontractors analyzed the last thermal-shock experiment and continued development of a standard procedure for crack-arrest toughness testing and investigation of transition from cleavage to fibrous fracture. Work progressed toward initial testing of specimens in the Fourth HSST Irradiation Series. Parametric analysis of overcooling accidents was completed, and improvements to the OCA-I code were made. Intermediate test vessel V-8A was flawed and is in preparation for testing. A major part of the design of a pressurized-thermal-shock test facility was completed. Test equipment and specimens were prepared for clad plate fracture tests.

NUREG/CR-2752 V01: ADVANCED TWO-PHASE FLOW INSTRUMENTATION PROGRAM. Quarterly Progress Report For January-March 1982. HARDY, J. E.; ROGERS, S. C.; MILLER, G. N. Oak Ridge National Laboratory. July 1982. 17pp. 8208260438. ORNL/TM-8365. 14567:317.

The performance of the Westinghouse Reactor Vessel Level Indicating System (RVLIS) was analyzed for a series of tests in the Semiscale Test Facility. A summary of the results and conclusions from those small-break simulation experiments is presented. The Westinghouse RVLIS indicated a level measurement that was lower than the vessel coolant level for all tests. The RVLIS indication was at times higher than the vessel collapsed liquid level. This discrepancy was apparently created by structural differences in the Semiscale facility and a Westinghouse pressurized-water reactor (PWR). Modifications were made to Semiscale to better simulate a Westinghouse PWR, and a retest was completed. The RVLIS measurements showed much better agreement with the vessel collapsed liquid level for this retest.

NUREG/CR-2753: ANALYSIS OF A DOUBLE-ENDED COLD-LEG BREAK SIMULATION--THTF TEST 3.05.5B. CRADDICK, W. G.; PEVEY, R. E. Oak Ridge National Laboratory. October 1982. 111pp. 8212060465. ORNL-5886. 16347:216.

On July 3, 1980, an experiment was performed in the Oak Ridge National Laboratory Thermal-Hydraulic Test Facility tht simulated a double-ended cold-leg break pressurized-water reactor (PWR) accident. Analysis of the experiment revealed that nuclear fuel rods exposed to the same hydrodynamic evironment as that which existed in the experiment would have departed from nucleate boiling both earlier and later than the fuel rod simulator (FRS), depending on the size of the gap between the nuclear fuel pellets and cladding and on the initial power of the nuclear fuel rod. Comparison of the results of the current experiment, which used an FRS bundle with geometry similar to 17 x 17 PWR fuel assemblies, to the results of earlier experiments, which used an FRS bundle with geometry similar to 15 x 15 PWR fuel assemblies, revealed no differences that can be attributed to the difference in geometries.

NUREG/CR-2754: CRITICAL REVIEW OF STUDIES ON ATMOSPHERIC DISPERSION IN COASTAL REGIONS. SHEARER, D. L.; KALEEL, R. J. TRC Environmental Consultants, Inc. * Battelle Memorial Institute, Pacific Northwest Laboratory. September 1982. 59pp. 8210050418. PNL-4292. 15625:237.

This report was prepared in an attempt to assess the current information and data bases that exist based on studies conducted in coastal regions. Reports covering research and meteorological measurements conducted for industrial purposes, utility needs, military objectives, and academic studies have been obtained and critically reviewed. The report provides an interpretation of the extent of existing usable information, an indication of the potential for tailoring existing research toward present NRC information needs, and recommendations for several follow-up studies which could provide valuable additional information through reanalysis of the data. Emphasis was placed on the identification and acquisition of data from atmospheric tracer studies conducted in coastal regions. A total of 225 references were identified which deal with the coastal atmosphere, including meteorological and tracer measurement programs, theoretical descriptions of the relevant processes, and dispersion models.

NUREG/CR-2757: ZIRCALOY CLADDING EMBRITTLEMENT CRITERIA: COMPARISON OF IN-PILE AND OUT-OF-PILE RESULTS. HAGGAG, F. M. EG&G, Inc. August 1982. 59pp. 8209210438. EGG-2175. 14948:309.

Zircaloy-4 cladding embrittlement data from both in-pile and out-of-pile experiments are compared and correlated with embrittlement criteria based on the fraction of the remaining beta phase, the extent of oxidation (equivalent cladding reacted), and the oxygen concentration in the beta phase. The in-pile data are from the Power-Cooling-Mismatch and Irradiation Effects Test Series performed in the Power Burst Facility reactor at the Idaho National Engineering Laboratory. The out-of-pile data are from isothermal oxidation experiments conducted at Argonne National Laboratory on simulated fuel rods in high-temperature steam. The zircaloy embrittlement criteria most applicable to severe fuel damage conditions are identified.

NUREG/CR-2758: A PARAMETRIC STUDY OF CONTAINMENT EMERGENCY SUMP PERFORMANCE. WEIGAND, G. G.; KREIN, M. S.; WESTER, M. J.; et al. Sandia Laboratories. July 1982. 208pp. 8207290405. SAND82-0624. 14153:001.

A systematically structured test program designed to characterize the hydraulic behavior of full-scale emergency core cooling system (ECCS) sumps under a broad ranje of geometric configurations and flow conditions has been conducted. The effects of potential accident induced perturbations on sump performance were also evaluated. The perturbations included screen blockage, nonuniform approach flows, break flow and ice condenser drain flow impingement, and obstructions. In addition, the effects of elevated water temperature and the performance of vortex suppression devices have been established. The results show that the vortices are unstable and that vortex size and type is not a reliable indicator to adjudge air ingestion or swirl behavior. Measured air withdrawal rates were generally less than 1-2 percent and the measured swirl in the outlet pipes was small. An envelope curve analysis of the data was developed, and it gives the "bounded" performance response of the sump as a function of the flow variables.

These results are being used to develop comprehensive design and review guidelines for ECCS sumps. Additionally, the test results will be used in developing the resolution of Unresolved Safety Issue A-43, "Containment Emergency Sump Performance."

NUREG/CR-2759: A PARAMETRIC STUDY OF CONTAINMENT EMERGENCY SUMP PERFORMANCE: RESULTS OF VERTICAL OUTLET SUMP TESTS. KREIN, M. S.; WESTER, M. J.; STROM, P. O.; et al. Sandia Laboratories. October 1982. 113pp. 8210270338. SAND82-7062. 15847:271.

This report presents the results of a test program designed to characterize the hydraulic performance of sumps with vertical outlets. The tests were performed for a wide range of geometric and flow variables typical of ECCS sumps. The work on vertical outlet sumps presented here supplements a broader test program characterizing sumps with horizontal outlets.

In addition to a parametric evaluation of the operating characteristics of vertical outlet sumps under normal approach flow conditions, the effects of perturbations to the approach flow have been considered. The effectiveness of two vortex suppression devices was demonstrated.

Vortex severity was found to be an unreliable indicator of air ingestion levels or outlet swirl performance. Air ingestion levels were found to be generally less than 2 percent void fraction under the test conditions. Outlet swirl angles remained generally below 3 degrees. Nonuniform approach flows perturbations resulted in some increased levels of air ingestion and outlet swirl angle over unperturbed approach conditions.

The results indicate that no major differences exist between the performance of vertical and horizontal sumps.

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-2761: RESULTS OF VORTEX SUPPRESSOR TESTS, SINGLE OUTLET SUMP TESTS, AND MISCELLANEOUS SENSITIVITY TESTS. PADMANABHAN, M. Alden Research Laboratory. * Sandia Laboratories. September 1982. 115pp. 8210050388. SAND82-7065. 15625:120.

Full scale tests of flow conditions in Containment Recirculation Sumps for nuclear power stations were conducted at the Alden Research Laboratory to provide sump hydraulic design and performance data for use in resolving the Unresolved Safety Issue, A-43 "Containment Sump Performance."

This document is a report of the results in investigations conducted as a part of Phase II of the test program, including (a) vortex suppressor tests of two commonly used suppressors, (b) single outlet sump tests and comparison to double outlet sumps, and (c) tests to study the effects on the hydraulic performance of a solid partition wall in a double outlet sump, pump overspeed, outlet pipe diameter, and bellmouth entrances.

Test data on single and double outlet sumps were used for an envelope analysis so as to derive appropriate maximum bounding values for average vortex types, air-withdrawals, pipe swirl, and inlet loss coefficients versus Froude number. These bounding values are compared with the bounding values of the Phase I test. Results of the envelope analysis and an evaluation of other results would provide a data base for use in the preparation of sump design and in their evaluation, and thereby assists in the resolution of the Unresolved Safety Issue, A-43 "Containment Sump Performance."

NUREG/CR-2762: AN ORIGEN2 MODEL AND RESULTS FOR THE CLINCH RIVER BREEDER REACTOR. CROFF, A. G. ; BJERKE, M. A. Oak Ridge National Laboratory. July 1982. 116pp. 8208170303. ORNL-5884. 14372:138.

Reactor physics calculations and literature information acquisition have led to the development of a Clinch River Breeder Reactor (CRBR) model for the ORIGEN2 computer code. The model is based on cross sections taken directly from physics codes. Details are presented concerning the physical description of the fuel assemblies, the fuel management scheme, irradiation parameters, and initial material compositions. The ORIGEN2 model for the CRBR has been implemented, resulting in the production of graphical and tabular characteristics (radioactivity, thermal power, and toxicity) of CRBR spent fuel, high-level waste, and fuel-assembly structural material waste as a function of decay time. Characteristics for pressurized water reactors (PWRs), commercial liquid-metal fast breeder reactors (LMFBRs), and the Fast Flux Test Facility (FFTF) have also been included in this report for comparison with the CRBR data.

NUREG/CR-2763: LOSS OF COOLANT ACCIDENT (LOCA) SIMULATION TESTS ON POLYMERS: THE IMPORTANCE OF INCLUDING OXYGEN. GILLEN, K. T. ; CLOUGH, R. L. ; GANOUNA-COHEN; et al. Sandia Laboratories. August 1982. 42pp. 8209270354. SAND82-1071. 15512:254.

Experiments were performed to survey the effects on material degradation of both aging conditions and the oxygen concentration during LOCA simulation. Changes for a number of commercial materials commonly used as electric cable jackets and insulations in nuclear power plant applications were monitored in terms of weight, mechanical properties, solubility measurements and infrared spectroscopy. For a number of these materials (an EPR insulation, a chloroprene jacket and a PVC jacket), the concentration of oxygen during LOCA simulation was found to be an important parameter. For the first two materials, more degredation occurred when oxygen was present; for PVC, substantially increased swelling occurred as the oxygen concentration was lowered. Aging conditions were also found to have a very substantial influence. In particular, for a number of the materials, lowering the radiation dose rate used for aging led to enhanced degradation after both the aging and the LOCA simulation. The different materials examined showed very different behaviors in terms of the degradation resulting from aging and from LOCA simulation.

NUREG/CR-2765: A MATHEMATICAL MODEL FOR RADON DIFFUSION IN EARTHEN MATERIALS. NIELSON, K. K. ; ROGERS, V. C. Rogers Engineering Co., Inc. * Battelle Memorial Institute, Pacific Northwest Laboratory. October 1982. 49pp. 8212010115. PNL-4301. 16291:031.

Radon migration in porous, earthen materials is characterized by diffusion in both the air and water components of the system as well as by the interaction of the radon between the air and water. The size distribution and configuration of the pore spaces and their moisture distributions are key parameters in determining the radon diffusion coefficient for bulk material. A mathematical model is developed and presented for calculating radon diffusion coefficients solely from the moisture content and pore size distribution of a soil, reducing the need for resorting to radon diffusion measurements. The resulting diffusion coefficients increase with the median pore diameter of the soil and decrease with increasing widths of the pore size distribution. The calculated diffusion coefficients are suitable for use in simple homogeneous-medium diffusion experiments for predicting radon transport and compare well with measured diffusion coefficients and with empirical diffusion coefficient correlations.

NUREG/CR-2767: X-RAY MEASUREMENTS OF WATER FOG DENSITY. CAMP, A. L. Sandia Laboratories. December 1982. 42pp. 8301'20101. SAND82-1292. 16780:194.

Water fog densities were measured in a laboratory experiment using x-ray diagnostics. Fog densities were measured, varying the flow rate, nozzle type, nozzle configuration, nozzle height above the x-ray beam, and water surface tension. Suspended water volume fractions between 0.0008 and 0.00074 percent were measured. The fog density increases approximately as the square root of the flow rate; the other parameters had little effect on the density.

NUREG/CR-2768: LITERATURE REVIEW OF MODELS FOR ESTIMATING SOIL EROSION AND DEPOSITION FROM WIND STRESSES ON URANIUM MILL TAILINGS COVERS. BANDER, T. J. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1982. 29pp. 8212270487. PNL-4302. 16548:278.

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. The mechanics of wind erosion, as well as of soil deposition, are discussed in this report. Several wind erosion models are reviewed to determine if they can be used to estimate the erosion of soil from a mill tailings cover. One model, developed by W. S. Chepil, contains the most important factors that describe variables that influence wind erosion. Particular features of other models are also discussed, as well as the application of Chepil's model to a particular tailings pile. For this particular tailings pile, the estimated erosion was almost one inch per year for an unprotected tailings soil surface. Wide variability in the deposition velocity and lack of adequate deposition models preclude reliable estimates of the rate at which airborne particles are deposited.

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-2773: COMMIX 1-A THREE DIMENSIONAL IN-VESSEL SIMULATION OF THE FFTF TRANSIENT THERMAL HYDRAULICS. VANKA, S. P.; DOMANUS, H. M.; SHA, W. T. Argonne National Laboratory. August 1982. 106pp. 8209230024. ANL-CT-82-14. 14977:229.

The three-dimensional flow and temperature fields occurring in the FFTF during a flow transient followed by a reactor scram have been simulated by the COMMIX-1A computer code. The transient simulated corresponds to the tests conducted at the Hanford Engineering and Development Laboratory. The COMMIX-1A code employs the porous media formulation in which the concepts of volume porosity, surface permeability, distributed resistance, and distributed heat source are used to model a flow domain with internal structures. The governing equations for conservation of mass, momentum, and energy are solved as a boundary-value problem in space and as an initial-value problem in time. The present report presents the calculated results for the steady-state reactor full-power operation and for a transient from full flow down to natural circulation combined with a power scram. The results are compared with experimental measurements, where applicable.

NUREG/CR-2774 V01: PHYSICS OF REACTOR SAFETY. Quarterly Report, January-March 1982. * Argonne National Laboratory. July 1982. 31pp. 8208120362. ANL-82-24 V01. 14340:037.

This quarterly progress report summarizes work done during the months of January-March 1982. 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 accidents under natural convection conditions. An executive summary is provided including a statement of the findings and recommendations of the report.

NUREG/CR-2774 V02: PHYSICS OF REACTOR SAFETY. April-June 1982. * Argonne National Laboratory. November 1982. 43pp. 8212130143. ANL-82-24. 16415:164.

This Quarterly progress report summarizes work done during the months of April-June 1982 in Argonne National Laboratory's Applied Physics and Components Technology Divisions for the Division of Reactor Safety Research of the U.S. Nuclear Regulatory Commission. 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 accidents under natural convection conditions. An executive summary is provided including a statement of the findings and recommendations of the report.

NUREG/CR-2775: A TRAC-PD2 ANALYSIS OF A LARGE-BREAK LOSS-OF-COOLANT ACCIDENT IN A TYPICAL US PWR. IRELAND, J.R. Los Alamos Scientific Laboratory. August 1982. 108pp. 8209210473. LA-9411-MS. 14949:035.

A 200 percent double-ended cold-leg break loss-of-coolant accident (LOCA) in a typical US pressurized water reactor (PWR) was simulated using the Transient Reactor Analysis Code (TRAC-PD2). The reactor system modeled represented a "typical" US PWR with four loops (three intact, one broken) and cold-leg emergency-core-cooling systems (ECCS). The finely noded TRAC model employed 440 three-dimensional (r, θ, z) vessel cells along with approximately 300 one-dimensional cells that modeled the primary system loops. The calculated peak cladding temperature of 950 K occurred during blowdown and the cladding temperature excursion was terminated at 175 s, when complete core quenching occurred. Accumulator flows were initiated at 10 s, when the system pressure reached 4.08 MPa, and the refill phase ended at 36 s, when the lower plenum refilled. During reflood, both bottom and falling film quench fronts were calculated. Top quenching was caused by entrainment from the lower plenum and lower core regions. The entrained liquid was sufficient to form a small, saturated pool (0.3 m deep) above the upper core support plate (UCSP). Also, some of the entrained liquid was carried out the hot legs and vaporized in the steam generators. Strong multidimensional effects were calculated in the reactor vessel, particularly with respect to rod quenching. The calculation shows that some rods located in core regions closest to the intact cold legs (ECCS injection points) quench 125 s sooner than rods located in core regions next to the broken loop.

NUREG/CR-2776: ALARMS WITHIN ADVANCED DISPLAY SYSTEMS: ALTERNATIVES AND PERFORMANCE MEASURES. DANCHAK, M.M. EG&G, Inc. September 1982. 55pp. 8209270459. EGG-2202. 15526:307.

This study surveys five advanced alarm handling systems in industries having problems similar to nuclear process control. The survey identifies the uniqueness of each system as well as features common to all. One such common feature is the use of alphanumeric alarm message strings displayed on cathode ray tubes (CRT). The study presents alternatives for display of this information and dynamic techniques for the addition and deletion of alarms. A software package is described that incorporates the alternatives and allows low-fidelity experiments to be conducted in an environment that simulates nuclear process control. The package was used to test static aspects of alarm CRTs and led to the conclusion that quantitative data should come before qualitative data in alarm message strings. Methods for low-fidelity testing of display dynamics are also discussed.

NUREG/CR-2777: THE RELEASE OF FISSION GAS DURING TRANSIENT HEATING OF LWR FUEL. GEHL, S.M. Argonne National Laboratory. August 1982. 74pp. 8209270127. ANL-80-108. 15516:015.

The direct electrical heating technique was used to study fission-gas release and mechanical behavior of irradiated light-water reactor (LWR) fuels during thermal transients. An empirical correlation between fission-gas release and transient temperature history was developed for power-cooling mismatch (PCM) and anticipated transients. Gas release during the refill portion of a design-basis loss of cooling accident was estimated to be less than 1%. Fission-gas release during PCM accidents was found to be controlled by intergranular microcracking and the interlinkage of tunnels on grain edges. For high-gas-release transients, the fractional gas release was shown to be equal to the fractional coverage of grain boundaries by microcracks. Temperature calculations indicated that microcracking causes a significant decrease in the fuel thermal conductivity.

NUREG/CR-2778: CORRELATION FOR NUCLEATION SITE DENSITY AND ITS EFFECT ON INTERFACIAL AREA. KOCAMUSTAFAOGUL; CHEN, I.Y.; ISHII, M. Argonne National Laboratory. August 1982. 66pp. 8209270454. ANL-82-32. 15528:258.

The bubble number density is important for the determination of interfacial area in boiling two-phase flow. The interfacial area is a key parameter affecting the interfacial transfer of mass, momentum and energy between phases. For a two-fluid model formulation of two-phase flow analyses, therefore, the bubble number density is quite important, however, there have been no correlations available to calculate this parameter. In view of this, a new correlation for the number density of the active wall nucleation site as well as the calculational method to obtain the bubble number density in boiling flow were developed here. The model was developed first for a pool boiling system and then it was extended to a forced convection system.

NUREG/CR-2779: ARAC TESTING FOR POTENTIAL NUCLEAR REGULATORY COMMISSION METEOROLOGICAL STAFF USE. ROSEN, L.C. Lawrence Livermore Laboratory. November 1982. 43pp. 8301100017. UCRL-53039. 16718:311.

A Nuclear Regulatory Commission (NRC) sponsored project to examine and assess the potential of the Atmospheric Release Advisory Capability (ARAC) for contributing to the emergency response capabilities of the NRC staff is discussed herein. Preliminary planning, installation, and testing of the ARAC site facility at the NRC Incident Response Center are summarized. ARAC participation in two days of field testing at Idaho Falls, Idaho, in July 1981 is examined. The ARAC system is evaluated with emphasis on communications, meteorology, the suite of models contained within the ARAC system, and the staff. The implications of this project in designing the next-generation ARAC system to service federal and state needs are assessed.

NUREG/CR-2781: EVALUATION OF WATER HAMMER EVENTS IN LIGHT WATER REACTOR PLANTS. UFFER, R.A.; BANERJEE, S.; BUCKHOLZ, F.B.; et al. Quadrex Corp. July 1982. 121pp. 8207290406. QUAD-1-82-018. 14127:193.

This document presents the results of an evaluation of water hammer events in LWR power plants. The evaluation was based upon reports of actual events, typical plant design drawings and operation procedures. Included in this report are design and operating recommendations for the prevention or mitigation of water hammer occurrence.

NUREG/CR-2782: A SUMMARY OF REPOSITORY SITING MODELS. THOMAS, S.D.; ROSS, B.; MERCER, J.W. Geotrans, Inc. July 1982. 231pp. 8208130474. 14353:188.

This report is the first in a series of reports that will provide critical reviews and summaries of computer programs that can be used to analyze the potential performance of a high-level radioactive waste repository. The computer programs identified address the following phenomena: saturated and unsaturated subsurface flow, heat transport, solute transport, surface water runoff, geomechanical interactions, and geochemical interactions. The report identifies 183 computer programs that can be used to analyze a repository site and provides a summary description of 31 computer programs. The summary descriptions can be used: to assist in code evaluations, to facilitate code comparison, to determine applicability of codes to specific problems, to identify code deficiencies, and to provide a screening mechanism for code selection.

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-2784: BEHAVIOR OF WATER SPRAY INJECTED INTO AIR/STEAM ENVIRONMENT. LEE, S. Y.; TANKIN, R. S. Northwestern Univ. August 1982. 152pp. 8208260433. 14570:028.

The behavior of water spray injected into both an air and a steam environment was studied. The water spray was divided into two parts - sheet portion and droplet portion. An analytical model is proposed for explaining the spray behavior. Experiments were performed to substantiate the analytical results. The holographic pictures were used to obtain the droplet size distribution. These size distributions were used for computing the motion of spray droplets in the analytical model. For the sprays used in this study, most of the heat transfer occurs in the sheet portion rather than at the droplet portion of the spray. In addition, the spray angle is primarily governed by the sheet portion. The axial extent (length) of sheet is a very important parameter in determining the spray angle. A correlation is obtained experimentally for breakup length in terms of the Weber number and the Jakob number.

NUREG/CR-2787 V01: INTERIM RELIABILITY EVALUATION PROGRAM: ANALYSIS OF THE ARKANSAS NUCLEAR ONE-UNIT 1 NUCLEAR POWER PLANT. Docket No. 50-313. (Arkansas Power And Light Company) KOLB, G. J. Sandia Laboratories. August 1982. 296pp. 8208260479. SAND82-0978. 14571:035.

This report represents the results of the analysis of Arkansas Nuclear One (ANO) Unit 1 nuclear power plant which was performed as part of the Interim Reliability Evaluation Program (IREP). The IREP has several objectives, two of which are, (1) identification, of those accident sequences which are expected to dominate the public health and safety risks, and (2) the development of state-of-the-art plant system models which can be used as a foundation for subsequent, more intensive applications of probabilistic risk assessment. The primary methodological tools used in the analysis were event trees and fault trees. These tools were used to study core melt accidents initiated by loss of coolant accidents (LOCAs) of six different break size ranges and eight different types of transients. The study emphasis was on the estimation of core melt accident sequence frequencies. Core melt accidents with the highest frequency were analyzed in terms of containment phenomenology, and associated radioactive material release categories were estimated.

NUREG/CR-2787 V02: INTERIM RELIABILITY EVALUATION PROGRAM: ANALYSIS OF THE ARKANSAS NUCLEAR ONE-UNIT 1 NUCLEAR POWER PLANT. Appendices A-D. Docket No. 50-313. (Arkansas Power And Light Company) KOLB, G. J. Sandia Laboratories. October 1982. 1091pp. 8212060492. SAND82-0978. 16348:294.

This report containing Appendices A-D, represents the results of the analysis of Arkansas Nuclear One (ANO) Unit 1 nuclear power plant which was performed as part of the Interim Reliability Evaluation Program (IREP). The objectives of the analysis are (1) preliminary identification of those accident sequences expected to dominate the public health and safety risks, (2) development of state-of-the-art plant system models to be used as a foundation for subsequent, more intensive applications of probabilistic risk assessment. The primary methodological tools used in the analysis were event trees and fault trees. These tools were used to study core melt accidents initiated by loss of coolant accidents (LOCAs) of six different break size ranges and eight different types of transients. Volume 1, the Main Report emphasized the estimation of core melt accident frequencies. Core melt accidents with the highest frequency were analyzed in terms of containment phenomenology, and associated radioactive material release categories were estimated. This report, Volume 2, Appendices A-D, documents the systematic event tree analysis. Six LOCA systematic event trees were constructed and are presented. These are divided into two subsections.

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. 262pp. 8207060002. CWA 4010-FR. 13716:001.

The objective of this study was to develop force-time impact signature data for use in the d-~ign 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.

NUREG/CR-2791: METHODOLOGY FOR EVALUATION OF INSULATION DEBRIS EFFECTS. Containment Emergency Sump Performance, Unresolved Safety Issue A-43. WYSOCKI, J.; KOLBE, R. Sandia Laboratories. September 1982. 271pp. 8210150552. SAND82-7067. 15724:170.

The postulated failure of high energy piping within a light water reactor containment has raised safety questions related to the generation of insulation debris, the migration of such debris to the containment emergency sump screens and the potential for severe screen blockage. High, or total, screen blockages could result in impairment of the long-term RHR recirculation systems. Debris considerations are an integral part of the Unresolved Safety Issue A-43, "Containment Emergency Sump Performance." This report develops calculational methods and debris that might be generated by a LOCA, the transport of such debris, methods for estimating screen blockages and attendant pressure losses. Conservative assumptions are employed and the calculations are shown to be plant specific since the types and quantities of insulation, equipment location, and break locations establish initiating events. Plant layout and sump location, which determine migration paths, are also important. Calculational procedures for estimating break jet impingement effects and blocked screen pressure losses are included. Five operating plants which were analyzed using this debris evaluation methodology are given in the Appendices. The calculations clearly illustrate plant dependency.

NUREG/CR-2792: AN ASSESSMENT OF RESIDUAL HEAT REMOVAL AND CONTAINMENT SPRAY PUMP PERFORMANCE UNDER AIR AND DEBRIS INGESTING CONDITIONS. KAMATH, P. S.; TANTILLO, T. J.; SWIFT, W. L. Creare, Inc. September 1982. 110pp. 8210050360. 15625:010.

This report presents an assessment of the performance of Residual Heat Removal (RHR) and Containment Spray pumps during the recirculation phase of reactor core and containment cooldown following a Loss-of-Coolant Accident (LOCA). The pump fluid is expected to contain debris such as insulation and may ingest air depending on sump conditions.

Findings show that for pumps at normal flow rates operating with sufficient Net Positive Suction Head (NPSH), pump performance degradation is negligible if air ingestion quantities are less than 2% by volume. If air ingestion quantities exceed this amount, degradation may be severe depending on pump design, speed and flow rate. Small quantities of air will increase NPSH requirements for these pumps. For the types and quantities of debris likely to be present in the recirculating fluid, pump performance degradation is expected to be negligible. In the event of shaft seal failure due to wear or loss of cooling fluid, seal safety bushings limit leakage rates.

NUREG/CR-2793: TECHNIQUES FOR ELEVATED TEMPERATURE MECHANICAL TESTING. SCHMALE, D. T.; CROSS, R. W. Sandia Laboratories. August 1982. 35pp. 8208260037. 14576:342.

Commercially available standard mechanical test systems are generally designed to do tension tests at ambient temperatures. Experimental requirements which include the sometimes mutually exclusive requirements of high temperatures, strain measurement and control, precise specimen alignment, and fully reversed loading require substantial modifications to the standard equipment. Programs in solar central receiver technology and advanced nuclear reactor safety require the examination of low cycle fatigue and creep-fatigue behaviors in several structural alloys at elevated temperatures. The specialized techniques devised to successfully perform these tests are described here along with a discussion of the differing requirements and behavior of magnetic and nonmagnetic materials.

NUREG/CR-2794: COMPUTER USE IN THE HIGH TEMPERATURE MECHANICAL METALLURGY TESTING LABORATORY. CROSS, R. W. Sandia Laboratories. HUFNAGEL, R. EG&G, Inc. July 1982. 43pp. 8208260434. SAND82-0799. 14568:001.

High speed digital computers have come into use in the high temperature metals testing laboratory. This paper presents some of the advantages and disadvantages of the use of computers for test control, data acquisition, and the graphing of test results. While tests have been run for years without the use of computers, the significant enhancements offered by the computer in accuracy, repeatability, user interaction, and management of large amounts of data make the use of computers almost mandatory when data from many tests by different laboratories are compared and evaluated. The particular case of the use of MTS hydraulic test frames with the DEC PDP-11/34 computer is treated here, under the assumption that other computer-based systems would display similar advantages and similar problems.

NUREG/CR-2796: COMPRESSED-AIR AND BACKUP NITROGEN SYSTEMS IN NUCLEAR POWER PLANTS. HAGEN, E. W. Oak Ridge National Laboratory. July 1982. 86pp. 8208170316. ORNL/NSIC-206. 14372:052.

This report reviews and evaluates the performance of the compressed-air and pressurized-nitrogen gas systems in commercial nuclear power units. The information was collected from readily available operating experiences, licensee event reports, system designs in safety analysis reports, and regulatory documents. The results are collated and analyzed for significance and impacts on power plant safety performance.

Under certain circumstances, the "fail-safe" philosophy for a piece of equipment or subsystem of the compressed-air systems initiated a series of actions culminating in reactor transient or unit scram. However, based on this study of prevailing operating experiences, reclassifying the compressed-gas systems to a higher safety level will neither prevent (not mitigate) the reoccurrences of such happenings nor alleviate nuclear power plant problems caused by inadequate maintenance, operating procedures, and/or practices. Conversely, because most of the problems were derived from the sources listed previously, upgrading of both maintenance and operating procedures will not only result in substantial improvement in the performance and availability of the compressed-air (and backup nitrogen) systems but in improved overall plant performance.

NUREG/CR-2797: EVALUATION OF EVENTS INVOLVING SERVICE WATER SYSTEMS IN NUCLEAR POWER PLANTS. HARIED, J. A. Oak Ridge National Laboratory. November 1982. 69pp. 8212270492. ORNL/NSIC-207. 16548:164.

This report reviews and evaluates events involving the plant service water systems in U.S. commercial boiling- and pressurized-water reactors from January 1979 through June 1981. The information was collected from operating experiences, licensee event reports, system designs in safety analysis reports, and other regulatory documents. The results were collated and analyzed according to (1) safety significance and (2) cause of event.

Sixteen reports describes events in the last 2.5 years of operating experience that meet the criteria for safety significance; another 21 reports describe events that were judged to be possibly safety significant. At Brunswick 1 on April 19, 1981, the plant service water system did not fulfill its safety design criteria on demand. However, an alternate cooling path was established, and the reactor was adequately cooled. Few human errors caused events of safety significance, though they were frequent enough to affect plant performances. Differences between events at boiling- and pressurized-water reactors were minimal.

NUREG/CR-2798: EVALUATION OF EVENTS INVOLVING UNPLANNED BORON DILUTIONS IN NUCLEAR POWER PLANTS. HAGEN, E. W. Oak Ridge National Laboratory. July 1982. 40pp. 8208130481. ORNL/NSIC-208. 14351:118.

This report reviews and evaluates events concerned with the inadvertent dilution of boron concentrations to the reactor coolant system for pressurized-water-cooled thermal reactors in commercial service. The safety concern is the unplanned addition of reactivity. The information was collected from operating experiences, licensee event reports, system designs in safety analysis reports, and regulatory documents. The results are collated and analyzed for significance and impact on power plant safety performance.

Several operating experience events were selected for analysis because they meet the criteria for safety significance. However, no boron dilution incidents resulted in a reactivity excursion or transient that scrammed a unit, nor was a reactor protection system challenged by any of the events. The most common cause for unplanned boron dilutions was human error, of which one was a common-mode/common-cause failure. For each recorded event, the operator had sufficient time to diagnose and correct the cause of the inadvertent dilution before the shutdown safety margin was lost or seriously challenged.

NUREG/CR-2799: EVALUATION OF EVENTS INVOLVING DECAY HEAT REMOVAL SYSTEMS IN NUCLEAR POWER PLANTS. HARIED, J. A. Oak Ridge National Laboratory. July 1982. 101pp. 8208130467. ORNL/NSIC-209. 14351:296.

This report reviews and evaluates events placed in the NSIC file involving the removal of decay heat in U.S. commercial boiling- and pressurized-water reactors from June 1979 through June 1981. The information was collected from operating experiences, licensee event reports, system designs in safety analysis reports, and other regulatory documents. The results were collated and analyzed according to safety significance and cause of event.

Thirty-eight reported events in these 2.1 years meet the criteria for safety significance. Steam bubble formation in the reactor vessel head during natural circulation cooldown at St. Lucie 1 was the most significant event; operator awareness of the possibility of this occurrence and preparedness for dealing with it was the most important recommendation. Cavitation of residual heat removal pumps during decay heat removal operation was the most common potentially significant event. Davis-Besse 1 had several instances in which an inadvertent signal to the safety features actuation system caused the operating residual heat removal pumps to align to the dry sump causing pump cavitation.

NUREG/CR-2802: INTERIM RELIABILITY EVALUATION PROGRAM:ANALYSIS OF THE BROWNS FERRY,UNIT 1,NUCLEAR PLANT.Docket No. 50-259. (Tennessee Valley Authority) MAYS,S.E.; POLOSKI,J.P.; SULLIVAN,W.H.; et al. EG&G, Inc. August 1982. 121pp. 8209270137. EGG-2199. 15519:004.

A probabilistic risk assessment (PRA) was made of the Browns Ferry, Unit 1, nuclear plant as part of the Nuclear Regulatory Commission's Interim Reliability Evaluation Program (IREP). Specific goals of the study were to identify the dominant contributors to core melt and develop a foundation for more extensive use of PRA methods. Event tree and fault tree analyses were used to estimate the frequency of accident sequences initiated by transients and loss of coolant accidents. Dominant sequences were grouped according to common containment failure modes and corresponding release categories on the basis of comparison with analyses of similar designs. Each of eight dominant sequences for Browns Ferry, Unit 2, were initiated by postulated plant transients. Six of the eight sequences involved failure of the long-term decay heat removal functions of the residual heat removal system. These sequences account for 73% of the sum of the dominant sequence frequencies. The other two sequences involved an anticipated transient without a (subsequent) scram and account for 27% of the sum of the dominant sequence frequencies. While no LOCA-initiated sequences were dominant contributors to the frequency of core melt accidents, two of the eight dominant sequences involved transient-induced stuck-open relief valve scenarios. The results show that the single most important factor in reducing the risk of a core melt accident at Browns Ferry, Unit 1, is providing reliable long-term decay heat removal capability; the next most important factor would be providing more reliable means to ensure that the reactor can be rapidly shut down and maintained subcritical.

NUREG/CR-2802 APP A: INTERIM RELIABILITY EVALUATION PROGRAM:ANALYSIS OF THE BROWNS FERRY,UNIT 1,NUCLEAR PLANT.Appendix A-Event Trees. Docket No. 50-259.(Tennessee Valley Authority) MAYS,S.E.; POLOSKI,J.P.; SULLIVAN,W.H.; et al. EG&G, Inc. August 1982. 72pp. 8209270446. EGG-2199. 15528:084.

A probabilistic risk assessment (PRA) was made of the Browns Ferry, Unit 1, nuclear plant as part of the Nuclear Regulatory Commission's Interim Reliability Evaluation Program (IREP). Specific goals of the study were to identify the dominant contributors to core melt and develop a foundation for more extensive use of PRA methods. Event tree and fault tree analyses were used to estimate the frequency of accident sequences initiated by transients and loss of coolant accidents. Dominant sequences were grouped according to common containment failure modes and corresponding release categories on the basis of comparison with analyses of similar designs. Each of eight dominant sequences for Browns Ferry, Unit 1, were initiated by postulated plant transients. Six of the eight sequences involved failure of the long-term decay heat removal functions of the residual heat removal system. These sequences account for 73% of the sum of the dominant sequence frequencies. The other two sequences involved an anticipated transient without a (subsequent) scram and account for 27% of the sum of the dominant sequence frequencies. While no LOCA-initiated sequences were dominant contributors to the frequency of core melt accidents, two of the eight dominant sequences involved transient-induced stuck-open relief valve scenarios. The results show that the single most important factor in reducing the risk of a core melt accident at Browns Ferry, Unit 1, is providing reliable long-term decay heat removal capability; the next most important factor would be providing more reliable means to ensure that the reactor can be rapidly shut down and maintained subcritical.

NUREG/CR-2802 APP B: INTERIM RELIABILITY EVALUATION PROGRAM: ANALYSIS OF THE BROWNS FERRY, UNIT 1, NUCLEAR PLANT. Appendix B-System Descriptions And Fault Trees. Docket No. 50-259. (Tennessee Valley Authority) MAYS, S. E.; POLOSKI, J. P.; SULLIVAN, W. H.; et al. EG&G, Inc. August 1982. 502pp. 8209270360. EGG-2199. 15519:212.

A probabilistic risk assessment (PRA) was made of the Browns Ferry, Unit 1, nuclear plant as part of Nuclear Regulatory Commission's Interim Reliability Evaluation Program (IREP). Specific goals of the study were to identify the dominant contributors to core melt and develop a foundation for more extensive use of PRA methods. Event tree and fault tree analyses were used to estimate the frequency of accident sequences initiated by transients and loss of coolant accidents. Dominant sequences were grouped according to common containment failure modes and corresponding release categories on the basis of comparison with analyses of similar designs. Each of eight dominant sequences for Browns Ferry, Unit 1, were initiated by postulated plant transients. Six of eight sequences involved failure of long-term decay heat removal functions of the residual heat removal system. These sequences account for 73% of sum of the dominant sequence frequencies. The other two sequences involved an anticipated transient without a (subsequent) scram and account for 27% of the sum of the dominant sequence frequencies. While no LOCA-initiated sequences were dominant contributors to the frequency of core melt accidents, two of the eight dominant sequences involved transient-induced stuck-open relief valve scenarios. The results show that the single most important factor in reducing the risk of core melt accident at Browns Ferry, Unit 1, is providing reliable long-term decay heat removal capability; the next most important factor would be providing more reliable means to ensure that the reactor can be rapidly shut down and maintained subcritical.

NUREG/CR-2802 APP C: INTERIM RELIABILITY EVALUATION PROGRAM: ANALYSIS OF THE BROWNS FERRY, UNIT 1, NUCLEAR PLANT. Appendix C-Sequence Quantification. Docket No. 50-259. (Tennessee Valley Authority) MAYS, S. E.; POLOSKI, J. P.; SULLIVAN, W. H.; et al. EG&G, Inc. August 1982. 98pp. 8209270448. EGG-2199. 15528:158.

A probabilistic risk assessment (PRA) was made of the Browns Ferry, Unit 1, nuclear plant as part of the Nuclear Regulatory Commission's Interim Reliability Evaluation Program (IREP). Specific goals of the study were to identify the dominant contributors to core melt and develop a foundation for more extensive use of PRA methods. Event tree and fault tree analyses were used to estimate the frequency of accident sequences initiated by transients and loss of coolant accidents. Dominant sequences were grouped according to common containment failure modes and corresponding release categories on the basis of comparison with analyses of similar designs. Each of eight dominant sequences for Browns Ferry, Unit 1, were initiated by postulated plant transients. Six of the eight sequences involved failure of the long-term decay heat removal functions of the residual heat removal system. These sequences account for 73% of the sum of the dominant sequence frequencies. The other two sequences involved an anticipated transient without a (subsequent) scram and account for 27% of the sum of the dominant sequence frequencies. While no LOCA-initiated sequences were dominant contributors to the frequency of core melt accidents, two of the eight dominant sequences involved transient-induced stuck-open relief valve scenarios. The results show that the single most important factor in reducing the risk of a core melt accident at Browns Ferry, Unit 1, is providing reliable long-term decay heat removal capability; the next most important factor would be providing more reliable means to ensure that the reactor can be rapidly shut down and maintained subcritical.

NUREG/CR-2807: TEMPERATURE ESTIMATES FROM ZIRCALOY OXIDATION KINETICS AND MICROSTRUCTURES. OLSEN,C.S. EG&G, Inc. November 1982. 53pp. 8212060480. EGG-2207. 16344:310.

Results of a comprehensive literature search are presented to evaluate the suitability of available zircaloy microstructural and oxidation data for estimating anticipated reactor fuel rod cladding temperatures. Additional oxidation experiments were conducted to evaluate low-temperature zircaloy oxidation characteristics for postirradiation estimation of cladding temperature by metallographic examination. Results of these experiments were used to calculate peak cladding temperatures of electrical heater rods and nuclear fuel rods that had been subjected to reactor temperature transients. Comparison of the calculated and measured peak cladding temperatures for these rods indicates that oxidation kinetics is a viable technique for estimating peak cladding temperatures over a broad temperature range. However, further improvement in zircaloy microstructure technology is necessary for precise estimation of peak cladding temperatures by microstructural examination.

NUREG/CR-2809 V01: AEROSOL RELEASE AND TRANSPORT PROGRAM. Quarterly Progress Report for January-March 1982. ADAMS,R.E.; TOBIAS,M.L. Oak Ridge National Laboratory. August 1982. 36pp. 8209270442. ORNL/TM-8397/V1. 15528:328.

This report summarizes progress for the Aerosol Release and Transport Program for the period January-March 1982. Topics discussed include (1) the source term experimental program in the Fuel Aerosol Simulant Facility; (2) Fe_2O_3 in steam (light-water reactor accident) aerosol experiments in the Nuclear Safety Pilot Plant (NSPP); (3) core-melt experiments in the Containment Research Instllation-II Facility, including studies of the behavior of fission product simulant elements as well as control rod silver alloy components; and (4) analytical modeling of moisture balance in steam experiments in the NSPP.

NUREG/CR-2811: DEVELOPMENT OF TECHNIQUES FOR FABRICATION OF FILM PROBE SENSOR ASSEMBLY. MOORHEAD, A. J. Oak Ridge National Laboratory. November 1982. 40pp. 8212060460. ORNL-5895. 16347:327.

Pulsed laser welding and brazing techniques were developed for fabrication of sensors designed to measure liquid film properties in out-of-reactor safety tests that simulate a loss-of-coolant accident in a pressurized-water nuclear reactor. These sensors were made possible by unique ceramic-to-metal seal system based on a cermet insulator and a brazing filler metal, both developed at ORNL. This seal system was shown to resist steam to an exposure of at least 100 h at 700 degrees Centigrade (1292 degrees Farenheit) and to resist repetitive thermal transients of 300 degrees Centigrade/s (540 degrees Farenheit). The film property sensors described differ from our earlier work in that (1) the sensor subassembly body contains larger electrodes and consequently a larger cermet insulator and (2) the triaxial instrumentation cables, which also contain a high-temperature ceramic-to-metal seal, are only 1.60 mm (0.063 inches) in diameter rather than the 3.18-mm (0.125-inch) diameter used previously. Thus, this report describes the successful adaptation of the previously developed ceramic-to-metal seal system to sensor subassembly containing both significantly larger and smaller brazed components.

NUREG/CR-2814 V01: NUCLEAR REACTOR SAFETY. January 1-March 31, 1982. STEVENSON, M. G. Los Alamos Scientific Laboratory. October 1982. 42pp. 8211120047. LA-9442-PR. 16066:255.

The work that is highlighted here represents accomplishments for the period January 1 - March 31, 1982 in reactor safety research. Presented are brief overviews compiled by project, along with a bibliography of Technical Notes and publications written during this quarter. Progress is reported in the following programs, TRAC Code Development, Thermal-Hydraulic Analysis for Reactor Safety Research, TRAC Independent Assessment, TRAC Applications to 2D/3D, Advanced Converter Safety Research, Upper Structure Dynamics Experiments, Methods for Safety Analysis, TRAC Calculational Assistance and User Liaison and the Severe Accidents Sequence Analysis Program (SASA).

NUREG/CR-2818: PROTECTIVE MEASURES AND REGULATORY STRATEGIES FOR CORE MELT ACCIDENTS. * International Energy Associates, Ltd. * Sandia Laboratories. October 1982. 47pp. 8211110640. SAND82-7071. 16048:140.

The effort that is documented in this report was initiated in the summer of 1980, at a time when the Nuclear Regulatory Commission (NRC) was considering rulemaking that would likely require significant design modifications to nuclear power plants in order to deal with (i.e., prevent and/or mitigate) core-damage and core-melt accidents. During the period of draft review of this report, the NRC began to focus on the concept of a safety goal. This development will allow, in our opinion, a more rational basis for evaluating the need for and extent of possible rulemaking for core-melt accidents. The following should first establish the need for core-melt rulemaking (e.g., based on risk) and, subsequently, a consistent strategy for implementing regulatory changes, if any. Consequently, this report does not now offer any particularly unique or innovative recommendations. It does, however, in our opinion, summarize the key issues associated with attempts to develop regulatory modifications to address core-melt accidents.

NUREG/CR-2820: RESOLUTION OF SHIPPER-RECEIVER DIFFERENCES. JOHNSTON, J. W. ; BROWNS, R. J. ; STEWART, K. B. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1982. 117pp. 8210120125. 15688:328.

This report describes statistical procedures for resolving significant shipper-receiver differences (SRD's) for nuclear material transfers and calculating the best estimate of the true amount of special nuclear material transferred. The resolution of a SRD should generally be accomplished by identifying and removing its causes. Physical and statistical methods and logical stepwise procedures for investigating an SRD under various circumstances are recommended in this report. It is concluded that the best estimates of the true quantities of special nuclear material in a transaction are obtained by calculating the inverse variance-weighted values of the net weights, element fractions, and isotope fractions using the measurement results from the shipper, receiver, and referee laboratory, if any, that are the best available data.

NUREG/CR-2822: CONCENTRATIONS OF COPPER-BINDING PROTEINS IN LIVERS OF BLUEGILLS FROM THE COOLING LAKE AT THE H. B. ROBINSON NUCLEAR POWER STATION. HARRISON, F. L. ; LAM, J. R. Lawrence Livermore Laboratory. November 1982. 40pp. 8212140459. UCRL-53041. 16418:325.

Bluegills collected from the cooling lake of the H. B. Robinson Nuclear Power Station near the effluent discharge, near the water intake to the cooling system, and from a control population in a local pond were examined for total copper in muscle and liver tissues and metalloproteins in different compartments of liver tissues. Also, copper changes in the environment were reflected in liver but not in muscle tissue. Liver metalloproteins were separated into low molecular weight (LMW), intermediate molecular weight (IMW), and high molecular weight (HMW) protein fractions using high performance liquid chromatography. Large differences in kinds and quantities of metals associated with metalloproteins were found. Copper concentrations in the LMW proteins (metallothionein-like proteins) were highest in bluegills from the discharge site and lowest in those from the control pond. Evidence of overloading of the metallothionein-like protein detoxification system was found in bluegills at the discharge site. These data and that from related studies indicate that the labile copper released from the cooling system of the H. B. Robinson Nuclear Power Station may be implicated in the increased deformaties and reduced reproductive capacity found in the bluegills population in the adjacent cooling lake.

NUREG/CR-2823: A REVIEW OF THE IMPACT OF COPPER RELEASED INTO MARINE AND ESTUARINE ENVIRONMENTS. HARRISON, F. L. Lawrence Livermore Laboratory. November 1982. 127pp. 8212140485. UCRL-53042. 16431:031.

Information on the concentrations of copper in abiotic and biotic compartments of marine and estuarine ecosystems and the effects on biota of increased amounts of copper in the water and sediments were reviewed. Data compiled and discussed include the quantities and physicochemical forms of copper in the water column, the concentrations of copper in the bedload sediments and interstitial waters, and the concentrations of copper in primary producers, annelid worms, molluscs, crustacea, minor invertebrates, and fishes. In addition, the acute and sublethal effects of copper on the same groups of biota were presented as well as data on copper concentration factors. This information can be used to evaluate for different types of ecosystems the ranges of concentrations that occur in nature, to identify ecosystems that are or may be impacted by copper released from industrial and urban sources, and to assess the effects on biota of the use of copper alloys in nuclear power station cooling systems.

NUREG/CR-2825: BWR 4/MARK I ACCIDENT SEQUENCES ASSESSMENT. YUE, D. D.; COLE, T. E. Oak Ridge National Laboratory. December 1982. 130pp. 8301120103. ORNL/TM-8148. 16780:237.

This work uses the MARCH computer code to investigate the major events that may occur at a BWR 4/Mark I nuclear power plant following a number of postulated transients. These events are, in turn, correlated to the Nuclear Regulatory Commission Emergency Action Level Guidelines. The Browns Ferry Nuclear Plant Unit 1 was used as a model in this study. Under the assumptions used in this study, all accident sequences analyzed would eventually result in core-melt and containment breach unless the operator took corrective action. In each sequence, the effect of parameter variations on the accident progression has also been investigated. Results of this study show that in most core meltdown sequences overtemperature in the drywell electric penetration assembly (EPA) seals would be the dominant failure mode except for sequences TW, S (2) I, and S (2) J, in which there is a total loss of decay heat removal capability, with resultant higher pressure buildup in the containment. For the latter sequences, overpressurization would be the dominant containment failure mode. With the assumptions concerning EPA seal failure used in this study, both failure modes would result in containment breach much sooner and correspond to lower containment pressure than those predicted in WASH-1400 for similar sequences. The amount of fission product releases outside the containment, on the other hand, might be greatly reduced to deposition of fission products and filtering effect of EPA seals following degraded core accidents.

NUREG/CR-2826: EXPERIMENT DATA REPORT FOR LOFT LARGE BREAK LOSS-OF-COOLANT EXPERIMENT L2-5. BAYLESS, P. D.; DIVINE, J. M. EG&G, Inc. August 1982. 324pp. 8209230062. EGG-2210. 14991:001.

Selected pertinent and uninterpreted data from the third nuclear large break loss-of-coolant experiment (Experiment L2-5) 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 [--1000 MW(e)] commercial PWR operations. Experiment L2-5 simulated a double-ended offset shear of a cold leg in the primary coolant system. The primary coolant pumps were tripped within 1 s after the break initiation, simulating a loss of site power. Consistent with the loss of power, the starting of the high- and low-pressure injection systems was delayed. The peak fuel rod cladding temperature achieved was 1078 (+ or -) 13 K. The emergency core cooling system re-covered the core and quenched the cladding. No evidence of core damage was detected.

NUREG/CR-2828: NUCLEAR CONTROL ROOM MODIFICATIONS AND THE ROLE OF TRANSFER OF TRAINING PRINCIPLES: A Review of Issues And Research. SAWYER, C. R.; PAIN, R. F.; COTT, H. V.; et al. EG&G, Inc. September 1982. 79pp. 8209280331. EGG-2211. 15532:205.

The goal of this project was to survey applied and theoretical studies dealing with the effect of control room change on operator performance under high stress conditions. Our survey did not find any directly applicable applied studies, hence our attention centered on the theoretical literature dealing with transfer of training. These findings were then used to develop a series of examples which illustrate the kinds of modifications that enhance control room performance and those that detract from it. Crews will readily adapt to or learn to use many control room additions and modifications. In other words, there is a positive transfer of training from the original design to the modified design. However, there is a possibility that some changes, though they conform to good human engineering standards, promote negative transfer of training. That is, the habits and patterns crews used before the modification interfere with learning and use of the changed controls, displays, or procedures. In every case modifications must be examined to assess whether or not they will disrupt or facilitate the process of transfer from the old to the new control room situation.

NUREG/CR-2831: A TEST OF THE CONTROLLABLE UNIT APPROACH (CUA) CONCEPT IN A LOW-ENRICHMENT URANIUM FUEL-FABRICATOR FACILITY. FOSTER, K. W. Mound Laboratory/Monsanto. October 1982. 108pp. 8210150549. MLM-2957. 15724:061.

The Controllable Unit Approach (CUA) to nuclear material control and accounting (MC&A) was developed by Monsanto Research Corporation staff at Mound Facility for the Nuclear Regulatory Research, to demonstrate the feasibility of controlling nuclear material processes to specific performance criteria. In order to demonstrate the applicability of the CUA methodology to processes requiring material control and accounting, Mound entered into a contract with a commercial nuclear-fuel manufacturer and applied CUA methodology to its high-throughput, low-enrichment uranium fuel fabrication process. The scope of the project was to evaluate the process to determine the added safeguards potential that could be achieved from the current measurement system. The methodology was used to identify the dominant errors in the process, and resulted in suggestions for potential refinement.

NUREG/CR-2832: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS. Annual Progress Report, April 1981-March 1982. EIDSON, A. F. Lovelace Biomed & Environmental Research Institute. December 1982. 58pp. 8212220228. LMF-97. 16526:001.

The problems addressed are the protection of uranium mill workers from occupational exposure to uranium through routine bioassay programs and the assessment of accidental worker exposures. Chemical properties of refined uranium ore (yellowcake) and the uranium distribution patterns among organs are compared. These studies will facilitate calculations of organ doses for specific exposures and will identify important bioassay procedures. Results of yellowcake sampling at four uranium mills are related to specific packaging steps and to predictions of appreciable upper respiratory tract deposition rates for the aerosols, if inhaled by a worker without respiratory protection. In vitro dissolution techniques were evaluated and their uses for interpreting urinary bioassay data are described. Preliminary results from an inhalation experiment using rats indicate that clearance of uranium from lung agreed with in vitro dissolution and infrared analyses of the yellowcake used. Preliminary results from an experiment to simulate wound contamination by yellowcake showed that more of the implanted dose of a less soluble form of yellowcake was retained at the would site than of a more soluble form at 32 days after implantation. A two-year study of yellowcake from two mills was initiated. Twenty Beagle dogs were exposed by nose-only inhalation to soluble yellowcake and 20 to the less soluble form.

NUREG/CR-2833 V01: CRITICAL HUMAN FACTORS ISSUES IN NUCLEAR POWER REGULATION & RECOMMENDED COMPREHENSIVE HUMAN FACTORS LONG RANGE PLAN: Executive Summary. HOPKINS,C.O.; SNYDER,H.L.; PRICE,H.E.; et al. Human Factors Society,Inc. August 1982. 65pp. 8208260002. 14587:007.

This comprehensive long range human factors plan for nuclear reactor regulation was developed by a Study Group of the Human Factors Society, Inc. This Study Group was selected by the Society to provide a balanced, experienced human factors perspective to the application of human factors scientific and engineering knowledge to nuclear power generation.

The report is presented in three volumes. Volume 1 contains an Executive Summary of the 18-month effort and its conclusions. Volume 2 summarizes all known nuclear human factors activity, evaluating this activity, wherever adequate information is available, and describes the recommended long-range (10-year) plan for human factors in regulation. Volume 3 elaborates upon each of the human factors issues and areas of concern that have led to recommendations in the long range plan.

NUREG/CR-2833 V02: CRITICAL HUMAN FACTORS ISSUES IN NUCLEAR POWER REGULATION AND RECOMMENDED COMPREHENSIVE HUMAN FACTORS LONG-RANGE PLAN: Program Evaluation And Recommended Long Range Plan. HOPKINS,C.O.; SNYDER,H.L.; PRICE,H.E.; et al. Human Factors Society,Inc. August 1982. 384pp. 8209020159. 14724:217.

This comprehensive long range human factors plan for nuclear reactor regulation was developed by a Study Group of the Human Factors Society, Inc. This Study Group was selected by the Society to provide a balanced, experienced human factors perspective to the application of human factors scientific and engineering knowledge to nuclear power generation.

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NUREG/CR-2833 V03: CRITICAL HUMAN FACTORS ISSUES IN NUCLEAR POWER REGULATION & RECOMMENDED COMPREHENSIVE HUMAN FACTORS LONG-RANGE PLAN: Critical Discussions of Human Factors Areas of Concern. HOPKINS,C.O.; SNYDER,H.L.; PRICE,H.E. Human Factors Society,Inc. August 1982. 279pp. 8208260017. 14587:298.

This comprehensive long range human factors plan for nuclear reactor regulation was developed by a Study Group of the Human Factors Society, Inc. This Study Group was selected by the Society to provide a balanced, experienced human factors perspective to the application of human factors scientific and engineering knowledge to nuclear power generation.

The report is presented in three volumes. Volume 1 contains an Executive Summary of the 18-month effort and its conclusions. Volume 2 summarizes all known nuclear human factors activity, evaluating this activity, wherever adequate information is available, and describes the recommended long-range (10-year) plan for human factors in regulation. Volume 3 elaborates upon each of the human factors issues and areas of concern that have led to recommendations in the long range plan.

NUREG/CR-2834: INDEPENDENT SEISMIC EVALUATION OF THE DIABLO CANYON UNIT 1 CONTAINMENT ANNULUS STRUCTURE AND SELECTED PIPING SYSTEMS. PHILIPPACOPOULO; REICH, M.; BEZLER, P. Brookhaven National Laboratory. August 1982. 177pp. 8209130289. BNL-NUREG-51566. 14780:135.

An independent review and development of the vertical floor spectra for the Unit 1 containment annulus structure of the Diablo Canyon Power Plant was carried out using a detailed three-dimensional model. The developed floor spectra were then utilized for confirmatory evaluations of two selected piping systems. The latter were evaluated by the envelope response spectrum method, and by the independent support motion response spectrum method. ASME Class 2 evaluations of the two systems were also performed. Finally, a confirmatory evaluation was carried out for the model utilized by URS/Blum for the development of the vertical floor response spectra.

NUREG/CR-2836 V01 P1: BUCKLING OF STEEL CONTAINMENT SHELLS. Task 1A: Dynamic Response And Buckling Of Offshore Power Systems Floating Nuclear Plant Containment Vessel. MELLER, E.; BUSHNELL, D. Lockheed Palo Alto Research Laboratory. December 1982. 261pp. 8301190372. LMSC D812950. 16858:186.

Static and dynamic analyses of the steel containment vessel of the Offshore Power Systems (OPS) floating nuclear plant were conducted with use of several computer programs developed at the Lockheed Missiles and Space Company (LMSC). These analyses were conducted as part of Task 1, "Evaluation of Two Steel Containment Designs." The main purpose was to evaluate the OPS containment shell with respect to buckling.

The report is divided into two main sections, the first giving results from modal vibration and linear dynamic response analyses of the the OPS containment neglecting and including the large equipment hatch and the second giving results from buckling analyses for a variety of models.

Good agreement is obtained for static stress and buckling predictions from PANDA, BOSOR4, BOSOR5, and STAGSC-1 for cases in which more than one of these computer programs can be applied to the same configuration and loading. It is very important to include nonlinear material behavior (plasticity) in the computerized models for buckling.

NUREG/CR-2836 V01 P2: BUCKLING OF STEEL CONTAINMENT SHELLS. Task 1b: Buckling Of Washington Public Power Supply Systems Plant No. 2 Containment Vessel. MELLER, E.; BUSHNELL, D. Lockheed Palo Alto Research Laboratory. December 1982. 72pp. 8301190224. LMSC D812950. 16860:324.

Static buckling analyses of the steel containment vessel of the Washington Public Power Supply Systems' (WPPSS) plant No. 2 were conducted with use of several computer programs developed at the Lockheed Missiles and Space Company (LMSC). These analyses were conducted as part of Task 1, "Evaluation of Two Steel Containment Designs."

The report is divided into two main sections. The first gives results from analyses of the containment as if it were axisymmetric (computerized models with use of BOSOR4, BOSOR5, and PANDA), and the second gives results from a STAGSC-1 model in which the largest penetration is included.

Good agreement is obtained from analyses with BOSOR5 and STAGSC-1 for a case in which both of these computer programs were applied to the same configuration and loading. It is important to include nonlinear material behavior (plasticity) in the computerized models for collapse.

Predictions of collapse from STAGSC-1 indicate that the largest penetration of the WPPSS-2 containment vessel is reinforced such that there is no decrease in load carrying capability below that indicated from models in which this penetration is neglected. A collapse load factor of 3.6 times the loads postulated by Pittsburgh Des Moines Steel (PDM) is indicated. The buckling mode is axisymmetric collapse. Bifurcation buckling involving nonaxisymmetric modes occurs at higher load factors than 3.6.

NUREG/CR-2836 V02: BUCKLING OF STEEL CONTAINMENT SHELLS. Task 2: Elastic Plastic Collapse Of Nonuniformly Axially Compressed Ring-Stiffened Cylindrical Shells With Reinforced Openings. MELLER, E. ; BUSHNELL, D. Lockheed Palo Alto Research Laboratory. December 1982. 125pp. 8301190220. LMSC D812950. 16866:023.

Results of nonlinear finite element analysis of several axially compressed, ring-stiffened steel, cylindrical shells are presented, including comparisons with tests conducted at the Los Alamos National Laoratory. The specimens, all with radius-to-thickness ratio of about 450, have reinforced circular openings that cut across various numbers of ring stiffeners. The cylinders are loaded by enforced axial displacement applied to thick end plates halfway between the axis of revolution and the cylindrical shell wall. Measured imperfections are included in the analysis of a ring-stiffened specimen without any cutout. Inclusion of the imperfection field reduces the predicted collapse load by only eight per cent. Reinforced openings that cut one, two or three rings reduce the collapse load by 14, 21 and 22 per cent, respectively.

NUREG/CR-2836 V03: BUCKLING OF STEEL CONTAINMENT SHELLS. Task 3: Parameter Studies. ALMROTH, B. O. ; RANKIN, C. ; BUSHNELL, D. Lockheed Palo Alto Research Laboratory. December 1982. 82pp. 8301190188. LMSC D812950. 16858:110.

The content of this volume is related to the choice of knockdown factors to be used in the design of containment vessels. The presently used design recommendations are based on experimental results pertaining to shells subjected to uniform static loading. Relatively few experiments have been performed on fabricated shells typical of civil engineering structures. A variety of factors have some effect on the imperfection sensitivity of shell structures, such as ratios between basic dimensions degree of stiffening, inelastic deformation, duration and uniformity of loading. Therefore, it is hardly possible to collect sufficient empirical information to allow for the development of truly satisfactory design procedures. It is suggested here that the situation can be greatly improved if numerical experimentations is used to enlarge deterministic imperfections inherent in the manufacturing process and truly random imperfections. The utilization of numerical results to verify or modify code recommendations is demonstrated for ring stiffened inelastic shell segments. Some effects on buckling of nonuniformity in loading are discussed.

NUREG/CR-2836 V04: BUCKLING OF STEEL CONTAINMENT SHEELS. Task 4: Use Of The PANDA Program For Simple Buckling Analyses Of Stiffened Cylindrical Shells. BUSHNELL, D. Lockheed Palo Alto Research Laboratory. December 1982. 83pp. 8301190216. LMSC D812950. 16860:075.

Under Task 4 the PANDA computer program was modified to permit calculation of critical load interaction curves for buckling of stiffened cylindrical shells with stiffeners running axially or circumferentially or both. Knockdown factors for geometric imperfections and plasticity reduction factors were introduced so that interaction curves can now be calculated for imperfect elastic-plastic shells. The knockdown factors and plasticity reduction factors are computed from a modified form of ASME Code Case N-284. The new version of PANDA was checked by making numerous comparisons with tests on fabricated stiffened cylinders.

NUREG/CR-2837: PNL TECHNICAL REVIEW OF PRESSURIZED THERMAL SHOCK ISSUES. TAYLOR, T. T.; PEDERSON, L. T.; APLEY, W. J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1982. 147pp. 8208130475. PNL-4327. 14351:149.

Pacific Northwest Laboratory (PNL) was asked to develop and recommend a regulatory position that the Nuclear Regulatory Commission (NRC) should adopt regarding the ability of reactor pressure vessels to withstand the effects of pressurized thermal shock(PTS). Licensees of eight pressurized water reactors provided NRC with estimates of remaining effective full power years before corrective actions would be required to prevent an unsafe operating condition. PNL reviewed these responses and the results of supporting research and concluded that none of the eight reactors would undergo vessel failures from a PTS event before several more years of operation. Operator actions, however, were often required to terminate a PTS event before it deteriorated to the point where failure could occur. Therefore, the near-term (less than one year) recommendation is to upgrade, on a site-specific basis, operational procedures, training, and control room instrumentation. Also, uniform criteria should be developed by NRC for use during future licensee analyses. Finally, it was recommended that NRC upgrade nondestructive inspection techniques used during vessel examinations and become more involved in the evaluation of annealing requirements.

NUREG/CR-2838: HYDROACOUSTIC BIOMASS ESTIMATION TECHNIQUES. KANCIRUK, P. Oak Ridge National Laboratory. December 1982. 291pp. 8301120104. ORNL/TM-8304. 16781:007.

The use of hydroacoustic (sonar) biomass estimation techniques as they might apply to power plant aquatic environmental monitoring programs is discussed. Background information on the physics of sound in water and basic hydroacoustic equipment is presented. The hydroacoustic literature is reviewed with examples provided of successful technique application toward a variety of monitoring and assessment goals. The results of a hydroacoustic user survey are presented in an appendix, along with an extensive, computer-indexed hydroacoustic bibliography. Hydroacoustic biomass estimation techniques are quantitative, cost-effective stock assessment tools, providing information not obtainable with more traditional survey methods. Hydroacoustic techniques are particularly adaptable to monitoring programs at power plant sites and should be strongly considered in designing operational monitoring systems.

NUREG/CR-2840: ANNOTATED TSUNAMI BIBLIOGRAPHY 1962-1976. PARARAS-CARAYAN; DONG, B.; FARMER, R. Internatl Tsunami Info Center, Intergovt Oceanographic Comm. August 1982. 515pp. 8209280335. 15546:001.

This compilation contains annotated citations to nearly 3,000 tsunami-related publications from 1962 to 1976 in English and several other languages. The foreign-language citations have English titles and abstracts.

NUREG/CR-2842: TRANSPORTATION OF RADIOACTIVE MATERIAL IN SOUTH CAROLINA. October 1980-September 1981. SAPPINGTON, R. South Carolina, State of. July 1982. 38pp. 8208120366. 14332:268.

Transportation of radioactive materials into and within South Carolina was studied for the fourth and final year under the joint NRC/DOT contract. The majority of the data presented results from inspections of shipments containing low-level radioactive wastes. A small number of packages containing other radioactive materials was also inspected. The results indicate a significant decrease in the number of violations of State and Federal regulations over the study period from October 1978 through September 1981. This may be attributed to the enforcement of new legislation and increased efforts of field inspectors. Major recommendations are:

1. The Department of Transportation should increase the emphasis on carrier equipment inspections.

2. Clarification and revision of Motor Carrier Safety regulations is needed. This should include provisions for required, frequent carrier inspections of equipment.

3. Carriers and drivers should be trained and knowledgeable as to their responsibilities and in applicable regulations.

NUREG/CR-2844: NONFUEL OPERATION AND MAINTENANCE COSTS FOR LARGE STEAM ELECTRIC POWER PLANTS-1982. MYERS, M.L.; FULLER, L.C.; BOWERS, H.I. Oak Ridge National Laboratory. October 1982. 60PP. 8211150688. ORNL/TM-8324. 16073:231.

Revised guidelines for 1982 are presented for estimating annual nonfuel operation and maintenance costs for large steam-electric power plants, specifically light-water-reactor plants and coal-fired plants. Previous guidelines were published in January 1979 in ORNL/TM-6467, A Procedure for Estimating Nonfuel Operation and Maintenance Costs for Large Steam-Electric Power Plants. Estimates for coal-fired plants include the option of scrubbing for flue-gas desulfurization. A computer program, OMCOST, that covers all plant options is also revised.

NUREG/CR-2845: DIGRD: AN INTERACTIVE GRID GENERATING PROGRAM. FOOTE, H.P.; RICE, W.A.; KINCAID, C.T. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1982. 61pp. 8212270512. PNL-4345. 16548:079.

Pacific Northwest Laboratory has completed the development and documentation of an interactive grid generating program (DIGRD, digitize grid). This program is designed to rapidly generate or modify grids necessary for the unsaturated flow code TRUST. In addition to the code, a user's manual was prepared. Unfortunately, the computer hardware that comprises the PNL interactive graphics capability is unique. Direct technology transfer is not possible, therefore, this report is intended to convey the utility of interactive graphics in supplying a grid generating capability. DIGRD has already been effectively used in the preparation of grids for the analysis of leachate movement from uranium mill tailings. The principal conclusion is that the interactive graphics employed in DIGRD are useful and economical in the development of complex grids. Grid generation activities that previously took between a half- and a full-man month can now be completed in less than a week. DIGRD users have recommended development of a uniform grid of either rectangles or equilateral triangles, which could be superimposed on any domain and then adjusted through the DIGRD program to match the boundaries of a tailings disposal facility. This improvement to the DIGRD package could further reduce the effort in grid generation while providing more optimal grids.

NUREG/CR-2846: ALTERNATIVE NUCLEAR FUEL CYCLE ARRANGEMENTS FOR PROLIFERATION RESISTANCE: AN OVERVIEW OF REGULATORY FACTORS. O'BRIEN, J.N. Brookhaven National Laboratory. August 1982. 437pp. 8210220135. BNL/NUREG-51567. 15794:090.

This anthology was prepared for the NRC by Brookhaven National Laboratory under contract to the Office of Nuclear Regulatory Research. It represents a compilation of a number of reports developed to familiarize NRC staff with the assessment of alternative fuel cycles. This project examined safeguards issues associated with alternative fuel cycles with particular emphasis in how NRC jurisdiction and interests may be affected by their use. Specific issues examined include the problems related to multinational fuel cycle facilities, potential effects on the US/IAEA agreement, development of an algorithm for ranking potential fuel cycles as to their desirability, and licensing issues associated with candidate fuel cycles.

NUREG/CR-2850 V01: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1979. BAKER, D.A.; PELOGUIN, R.A. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1982. 69pp. 8301190466. PNL-4221. 16850:001.

Population radiation dose commitments have been estimated from reported radionuclide releases from commercial power reactors operating during 1979. Fifty-year dose commitments from a one-year exposure were calculated from both liquid and atmospheric releases for four population groups (infant, child, teen-ager and adult) residing between 2 and 80 km from each site. The 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 1300 person-rem to a low of 0.0002 person-rem with an arithmetic mean of 38 person-rem. The total population dose for all sites was estimated at 1800 person-rem to the 94 million people considered at risk. The average individual dose commitment from all pathways on a site basis ranged from a low of 2 X 10 (-6) mrem to a high of 0.7 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-2851: TRANSPORTATION OF RADIOACTIVE MATERIAL IN ILLINOIS. June 1980-June 1981. MENWEG, M.; NORDIN, J.; SIMONIN, J.M. Illinois, State of. July 1982. 78pp. 8208260066. 14587:072.

The fourth year surveillance program was performed with the purpose of continuing the collaborative program between the State of Illinois, the NRC and DOT for the surveillance of radioactive material in surface transport within the State. Information related to handling practices and the condition of packages, adherence to transportation regulations, and other pertinent data was acquired from vehicle inspections. On the basis of the number of reports submitted by State troopers, the surface transit flow of radioactive materials is low. However, the O'Hare Airport Surveillance Study indicates the transit flow of radioactive materials is significantly higher than indicated by highway surveillance studies. Most vehicles surveyed had radiation levels below the DOT limits and that the most frequent violations found were improper shipping papers, improperly prepared or missing shipping labels on packages, and improper placarding. Some recommendations include: A federal agency should offer courses to shippers, carriers, and their drivers explaining DOT regulations concerning radioactive material shipments; A federal agency should develop a specific course on enforcement and interpretation of radioactive material regulations; DOT regulations should be revised taking into consideration the hazardous nature, from a health physics aspect, of low-level shipments of radioactive material; Surveillance of medical radioactive material shipments should be intensified because drivers are exposed to more than the limit of 2 mR/hr.

NUREG/CR-2852: TRANSPORTATION OF RADIOACTIVE MATERIAL IN NEVADA. September 1980-September 1981. * Nevada, State of. July 1982. 26pp. 8208180227. 14389:337.

This report describes a study conducted by the State of Nevada between September 1980 and September 1981 concerning the transportation of radioactive materials. Data is presented on the surveillance of airport terminals, inspection of shipments of radioactive materials to the state disposal site, and on the transportation of radioactive materials to the largest user in the state, the Nevada Test Site. It was noted in the metropolitan areas that taxicabs were used in the delivery of medical isotopes. The major recommendations resulting from the study were: future state surveillance should include hazardous materials, the U. S. Department of Transportation should tell the state the results of its enforcement, and the U. S. Nuclear Regulatory Commission should inspect its licensees packaging of radioactive waste.

NUREG/CR-2855: SUBCHANNEL ANALYSIS OF MULTIPLE CHF EVENTS. REDDY, D. G. ; FIGHETTI, C. F. Columbia Univ. * Brookhaven National Laboratory. August 1982. 162pp. 8209270468. BNL-NUREG-51570. 15530:016.

This report provides the results of an NRC sponsored study to determine the adequacy of critical heat flux (CHF) correlations presently used in the nuclear industry in predicting higher order CHF occurrences. The BAW-2, CE-1, W-3, and Columbia CHF correlations were used to analyze test data obtained from experiments conducted at the Heat Transfer Research Facility of Columbia University. The conclusions drawn from this study are: (1) higher order CHF events have the same characteristics as the first CHF; and (2) the CHF correlations presently available are adequate in predicting higher order CHF events.

NUREG/CR-2857: LWR STEAM SPIKE PHENOMENOLOGY: DEBRIS BED QUENCHING EXPERIMENTS. GINSBERG, T. ; KLEIN, J. ; KLAGES, J. ; et al. Brookhaven National Laboratory. December 1982. 83pp. 8301120196. BNL-NUREG-51571. 16789:158.

An experimental investigation is reported whose objective is to provide an understanding of the thermal interaction between superheated core debris and water during postulated light-water reactor degraded core accidents. The experiment was designed to study the heat transfer characteristics of superheated spheres as they are quenched in a packed bed configuration by an overlying pool of water. The results of the experiment are applied to understanding of the containment "steam spike" phenomenon.

NUREG/CR-2858: PAVAN: AN ATMOSPHERIC DISPERSION PROGRAM FOR EVALUATING DESIGN BASIS ACCIDENTAL RELEASES OF RADIOACTIVE MATERIALS FROM NUCLEAR POWER STATIONS. BANDER, T. J. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1982. 150pp. 8301100003. 16712:177.

This report provides a user's guide for the NRC computer program, PAVAN, which is a program used by the U. S. Nuclear Regulatory Commission to estimate downwind ground-level air concentrations for potential accidental releases of radioactive material from nuclear facilities. Such an assessment is required by 10 CFR Part 100 and 10 CFR Part 50. The program implements the guidance provided in Regulartory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." Using joint frequency distributions of wind direction and wind speed by atmospheric stability, the program provides relative air concentration (X/Q) values as functions of direction for various time periods at the exclusion area boundary (EAB) and the outer boundary of the low population zone (LPZ). Calculations of X/Q values can be made for assumed ground-level releases (e.g., through building penetrations and vents) or elevated releases from free-standing stacks. Various options may be selected by the user. They can account for variation in the location of release points, additional plume dispersion due to building wakes, plume meander under low wind speed conditions, and adjustments to consider non-straight trajectories. It computes an effective plume height using the physical release height which can be reduced by inputted terrain features. It cannot handle multiple emission sources.

NUREG/CR-2859: EVALUATION OF AIRCRAFT CRASH HAZARDS FOR NUCLEAR POWER PLANTS. KOT, C. A.; LIN, H. C.; VAN ERP, J. B.; et al. Argonne National Laboratory. September 1982. 128pp. 8210150557. ANL-CT-81-32. 15723:294.

The state of knowledge concerning aircraft crash hazards to nuclear power plants is critically evaluated. This effort is part of a study to analyze the potential effects of offsite hazards upon the safety of nuclear power plants and to develop a technical basis for the assessment of siting approaches for such facilities. The evaluation includes the deterministic modeling of aircraft crash scenarios and threat environments, the estimation of the effects on and the response of the vital plant systems, and the probabilistic aspects of the crash problem, i.e., data bases and statistical methodologies. Also critically reviewed are past licensing experience and regulatory practice with respect to aircraft crash hazards.

NUREG/CR-2861: IMAGE ANALYSIS FOR FACILITY SITING: A COMPARISON OF LOW AND HIGH-ALTITUDE IMAGE INTERPRETABILITY FOR LAND USE/LAND COVER MAPPING. BORELLA, H. M. EG&G, Inc. ESTES, J. E.; EZRA, C. E.; et al. California, Univ. of, Santa Barbara. November 1982. 76PP. 8302030208. S-744-R. 17013:221.

This study documents for two sites in Pennsylvania the interpretability of commercially acquired low-altitude and existing high-altitude aerial photography in terms of time, costs, and accuracy for Anderson Level II land use/land cover mapping. Information extracted from the imagery is to be used in the evaluation process for siting energy facilities.

Land use/land cover maps were drawn at 1:24,000 scale using commercially flown color infrared photography. Maps at 1:62,500 scale were made using existing high-altitude photography obtained from the United States Geological Surveys' EROS Data Center. The USGS-developed Anderson classification scheme (Level II) was used.

Detailed accuracy of the maps generated by manual image analysis was assessed by employing a stratified unaligned sampling procedure with additional samples selected and analyzed to insure adequate class representation. Both "area weighted" and "b class" accuracies were documented and field-verified. A discrepancy map was also drawn to illustrate difference in classifications between the two map scales. Map accuracy statistics as well as figures on production costs and the time needed to compile the two map sets were compiled and tabulated.

NUREG/CR-2862: GEOMORPHIC PROCESSES AND EVOLUTION OF BUTTERMILK VALLEY AND SELECTED TRIBUTARIES, WEST VALLEY, NEW YORK. BOOTHROYD, J. C.; TIMSON, B. S.; DUNNE, L. A. New York, State of. July 1982. 117pp. 8210210041. 15785:255.

Repetitive bar and channel mapping at several scales, class size and movement measurements, suspended-sediment sampling, and stream gaging of a 5 km reach of Buttermilk Creek and selected tributaries at West Valley, New York, were performed to determine short-term depositional and erosional processes, and long-term valley changes in the vicinity of the Western New York Nuclear Service Center. Changes to bar-and-channel geometry in Buttermilk Creek result from migration of large transverse bars in equilibrium with large floods (i.e. Hurricane Fredric, September 1979) with large amounts of lower terrace gravels being recycled. Downslope movement of landslides by slumping and earthflow is a continuous small volumetric sediment source, except for infrequent large scale gravity block deposits. Bedload transport, suspended-load sediment transport, and reservoir infill rates compare well with the denudation rate (6600 m3 yr-1). Middle-to high-level fluvial terraces in Buttermilk Creek are either adjacent to tributary confluences and preserved by an excess of bedload over transport capacity, or survive due to the stable channel on the opposite side of the valley for unknown reasons. Convex longitudinal profile of Franks Creek/Erdman Brook suggests instability with continued rapid downcutting. Valley widening occurs by parallel slope retreat. Future lowering of Buttermilk Creek is controlled by bedrock floors in Cattaraugus Creek and lower Buttermilk Creek. Tributary lowering and widening will continue independent of Buttermilk Creek base level changes.

NUREG/CR-2864: IDENTIFICATION OF SAFETY RELATED EQUIPMENT FOR ANALYSIS AND TESTING IN THE HYDROGEN BURN SURVIVAL PROGRAM. DANDINI, V. J. Sandia Laboratories. November 1982. 26pp. 8212130118. SAND82-1684. 16417:225.

In order to develop a methodology for determining the survivability of safety-related equipment in reactor safety systems it is first necessary to identify that equipment. The method used for selecting and identifying specific components in PWR ice condenser containment is outlined. A list of equipment showing plant, manufacturer and model number is contained in the Appendix.

NUREG/CR-2865: HYDROGEN COMBUSTION IN AQUEOUS FOAMS. BAER, M. R.; GRIFFITHS, S. K.; SHEPHERD, J. E. Sandia Laboratories. November 1982. 78pp. 8212270156. SAND82-0917. 16553:249.

Water fogs are recognized as an effective means to mitigate the effects of large-scale hydrogen combustion that might accompany some loss-of-coolant nuclear reactor accidents. Fogs of sufficiently high density to produce large beneficial effects may, however, be difficult to generate and maintain. An alternate method of suspending the desired mass of water is via high expansion-ratio aqueous foams. Because, in practice, the foam would be generated using the combustible gaseous contents of the containment vessel, combustion occurs inside the foam cells. Although foams generated with inert gas have been well studied for use in fire fighting, little is known about combustion in foams generated with flammable mixtures. To help assess the usefulness of aqueous foams in a mitigation plan, we have conducted several open tube tests and over one hundred closed vessel tests of hydrogen/air combustion with and without foam. Above 15% hydrogen concentration, the foam causes a significant reduction in the pressure rise. The maximum effect occurs at about 28% hydrogen (the stoichiometric limit is 29.6% hydrogen) where the peak overpressure is reduced by two and one-half. Despite this overall pressure reduction, the flame speed is increased by up to an order of magnitude for combustion in the foam and strong pressure fluctuations are observed near a hydrogen concentration of 23%.

NUREG/CR-2866: A SIMULANT-MATERIAL EXPERIMENTAL INVESTIGATION OF FLOW DYNAMICS IN THE CRBR UPPER CORE STRUCTURE. WILHELM, D.; STARKOVICH, V. S.; CHAPYAK, E. J. Los Alamos Scientific Laboratory. November 1982. 82pp. 8212130157. LA-9478-MS. 16415:284.

The results of a simulant-material experimental investigation of flow dynamics in the Clinch River Breeder Reactor (CRBR) Upper Core Structure are described. The methodology used to design the experimental apparatus and select test conditions is detailed. Numerous comparisons between experimental data and SIMMER-II Code calculations are presented with both advantages and limitations of the SIMMER modeling features identified.

NUREG/CR-2869: DIRECTORY AND PROFILE OF LICENSED URANIUM-RECOVERY FACILITIES. WARKENTIEN, J.; BUSCH, L. S.; DEPUE, J. Argonne National Laboratory. July 1982. 107pp. 8208260505. ANL/ES-12[illegible] [illegible]4595:066.

The Directory and Profile of Licensed Uranium-Recov[illegible] [illegible]acilities presents facts, data, and information about conventional u[illegible]ium mills, in-situ mining facilities, heap leach operations, and other operations which process and produce marketable quantities of yellowcake. In the United States, such facilities are found in Agreement States (Arizona, Colorado, Florida, Louisiana, New Mexico, Texas, and Washington) and in Non-Agreement States (Utah and Wyoming).

Each facility is described on a case-by-case basis. Reporting of information on the conventional uranium mills begins with a brief narrative description that outlines general and specific characteristics about the site. Data sheets summarize the principal operating characteristics of the facility by listing the following information: location/ownership, licensing data, processing of uranium, characteristics of effluent releases and/or tailings, and radiological parameters. For in-situ and heap leach facilities, only data sheets are included.

Data and information in this report is current through calendar year 1981. The final section contains a description of the system used for storage and retrieval of all information presented.

NUREG/CR-2872: EXPERIMENTAL VERIFICATION OF A CRACKED FUEL MECHANICAL MODEL. WILLIFORD,R.E. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1982. 166pp. 8301190184. PNL-4377. 16860:158.

The report describes the results of a series of laboratory experiments conducted to independently verify a model that describes the rod average nonlinear mechanical behavior of cracked fuel under normal operating conditions. The analytical model is briefly described, and each experiment is discussed in detail. The experiments were conducted to verify the general behavior and numerical values for the three primary independent modeling parameters (effective crack roughness, effective gap roughness, and total crack length) and to verify the model predictions that the effective Young's moduli for cracked fuel systems were substantially less than those for solid pellets. The model parameters and predictions were confirmed, and new insight was gained concerning the complexities of cracked fuel mechanics.

NUREG/CR-2873 V01: NUCLEAR FUEL CYCLE RISK ASSESSMENT:DESCRIPTIONS OF REPRESENTATIVE NON-REACTOR FACILITIES.Sections 1-14. SCHNEIDER,K.J.; BICKFORD,W.E.; DALING,P.M.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1982. 376pp. 8210120102. PNL-4306. 15686:313.

This report, the first from the NRC's Fuel Cycle Risk Assessment Program, defines and describes fuel cycle elements that are being considered in the program. One type of facility (and in some cases two) is described that is representative of each non-reactor element of the fuel cycle. The descriptions are based on real industrial-scale facilities that are current state-of-the-art, or on conceptual facilities where none now exist.

The fuel cycles considered in this report are for Light Water Reactors with once-through flow of spent fuel, and with plutonium and uranium recycle. Representative facilities for the following fuel cycle elements are described for uranium (or uranium plus plutonium where appropriate): mining, milling, conversion, enrichment, fuel fabrication, mixed-oxide fuel refabrication, fuel reprocessing, spent fuel storage, high-level waste storage, transuranic waste storage, spent fuel and high-level and transuranic waste disposal, low-level and intermediate-level waste disposal, and transportation. For each representative facility the description includes: mainline process, effluent processing and waste management, facility and hardware description, safety-related information (the emphasis of this report) and potential alternative concepts for that fuel cycle element.

NUREG/CR-2873 V02: NUCLEAR FUEL CYCLE RISK ASSESSMENT:DESCRIPTIONS OF REPRESENTATIVE NON-REACTOR FACILITIES.Sections 15-19. SCHNEIDER,K.J.; BICKFORD,W.E.; DALING,P.M.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1982. 361pp. 8210120092. PNL-4306. 15687:328.

This report, the first from the NRC's Fuel Cycle Risk Assessment Program, defines and describes fuel cycle elements that are being considered in the program. One type of facility (and in some cases two) is described that is representative of each non-reactor element of the fuel cycle. The descriptions are based on real industrial-scale facilities that are current state-of-the-art, or on conceptual facilities where none now exist.

The fuel cycles considered in this report are for Light Water Reactors with once-through flow of spent fuel, and with plutonium and uranium recycle. Representative facilities for the following fuel cycle elements are described for uranium (or uranium plus plutonium where appropriate): mining, milling, conversion, enrichment, fuel fabrication, mixed-oxide fuel waste storage, high-level waste storage, transuranic waste storage, spent fuel and high level and transuranic waste disposal, low-level and intermediate-level waste disposal, and transportation. For each representative facility the description includes: mainline process, effluent processing and waste management, facility and hardware description, safety-related information (the emphasis of this report) and potential alternative concepts for that fuel cycle element.

NUREG/CR-2874 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1-March 31,1982. BALL,S.J.; CLAPP,N.E.; CLEVELAND,J.C.; et al. Oak Ridge National Laboratory. September 1982. 19pp. 8210150564. ORNL/TM-8443/V1. 15716:113.

Continuing code development work included improvements to the BLAST steam generator dynamics simulation for the Fort St. Vrain reactor and implementation of a simplified version of the core model CORTAP. Work continued on severe accident sequence analyses, including further work on the ORECA code three-dimensional core simulation and a new effort to review the fission-product release and transport methodology. Work was also started on a siting study for a 2240-MW(t) high temperature gas-cooled reactor plant design.

NUREG/CR-2875: COMPARISON OF RADON DIFFUSION COEFFIENTS MEASURED BY TRANSIENT-DIFFUSION AND STEADY-STATE LABORATORY METHODS. NIELSON,K.K.; RICH,P.C.; ROGERS,V.C. Rogers Engineering Co.,Inc. November 1982. 43pp. 8212220341. PNL-4370. 16525:009.

A method was developed and used to determine radon diffusion coefficients in compacted soils by transient-diffusion measurements. A relative standard deviation of 12 percent was observed in repeated measurements with a dry soil by the transient-diffusion method, and a 40 percent uncertainty was determined for moistures exceeding 50 percent of saturation. Excellent agreement was also obtained between values of the diffusion coefficient for radon in air, as measured by the transient-diffusion method, and those in the published literature.

Good agreement was also obtained with diffusion coefficients measured by a steady-state method on the same soils. The agreement was best at low moistures, averaging less than ten percent difference, but differences of up to a factor of two were observed at high moistures. The comparison of the transient-diffusion and steady-state methods at low moistures provides an excellent verification of the theoretical validity and technical accuracy of these approaches, which are based on completely independent experimental conditions, measurement methods and mathematical interpretations.

NUREG/CR-2877: INVESTIGATION OF CABLE DETERIORATION IN THE CONTAINMENT BUILDING OF THE SAVANNAH RIVER NUCLEAR REACTOR. GILLEN,K.T.;

CLOUGH, R. W. ; JONES, L. H. Sandia Laboratories. September 1982. 42pp. 8210260211. SAND81-2613. 15831:048.

This report describes an investigation of the deterioration of polyethylene and polyvinylchloride cable materials which occurred in the containment building of the Savannah River nuclear reactor located in Aiken, South Carolina. Radiation dosimetry and temperature mapping data of the containment area indicated that the maximum dose experienced by the cable materials was only 2.5 Mrad at an average operating temperature of 43 degrees C. Considering this relatively moderate environment, the amount of material degradation seemed surprising. To understand these findings, we undertook an experimental program on the commercial polyethylene and polyvinylchloride materials used at the plant to investigate their degradation behavior under combined lambda - radiation and elevated temperature conditions. Several important aging effects were observed, including 1) a strong synergism between radiation and temperature, 2) large dose-rate dependent effects which occur over a wide range of dose rates, and 3) a dependence of the degradation in sequential radiation, elevated-temperature experiments on the ordering of the sequential exposure. The aging effects are discussed in terms of a chemical mechanism involving thermal breakdown of peroxides formed in reactions initiated by the radiation. Evidence for this mechanism is derived from infrared and chemiluminescence measurements and from chemical techniques.

NUREG/CR-2880: ACOUSTIC EMISSION MONITORING OF ASME SECTION III-HYDROSTATIC TEST. Watts Bar Unit 1 Nuclear Reactor. HUTTON, P. H. ; TAYLOR, T. T. ; DAWSON, J. F. ; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1982. 52pp. 8211160503. PNL-4307. 16128.173.

This report describes the acoustic emission monitoring performed during the ASME Section III hydrostatic testing of Watts Bar Nuclear Power Plant Unit 1 and the results obtained. Highlights of the results are:

1. Spontaneous AE was detected from a nozzle area during final pressurization.

2. Evaluation of the apparent source of the spontaneous AE using an empirically derived AE/fracture mechanics relationship agreed within a factor of two with an evaluation by ASME Section XI Code procedures.

3. AE was detected from a fracture specimen which was pressure coupled to the 10-inch accumulator nozzle. This provided reassurance of adequate system sensitivity.

4. High background noise was observed when all four reactor coolant pumps were operating.

Work is continuing at Watts Bar Unit 1 toward AE monitoring hot functional testing and subsequently monitoring during reactor operation.

NUREG/CR-2885: MECHANISM AND CORRELATION OF DROPLET ENTRAINMENT AND DEPOSITION IN ANNULAR TWO-PHASE FLOW. KATAOKA, I. ; ISHII, M. Argonne National Laboratory. September 1982. 74pp. 8210150556. ANL-82-44. 15725:081.

The droplet entrainment from a liquid film is important to the mass, momentum, and energy transfer processes in annular two-phase flow. For example, the amount of entrainment as well at the rate of entrainment significantly affect the occurrences of the dryout, whereas the post CHF heat transfer depends strongly on the entrainment and droplet sizes. Despite the importance of the entrainment rate, there have been no satisfactory correlations available in the literature. In view of these, correlations for entrainment rate covering both entrance region and equilibrium region have been developed from a simple model in collaboration with data. Results show that the entrainment rate varies considerably in the entrainment development region, however, at a certain distance from an inlet it attains an equilibrium value. A simple approximate correlation has been obtained for the equilibrium state where entrainment rate and deposition rate becomes equal. The result indicates that the equilibrium entrainment rate is proportional to Weber number based on the hydraulic diameter of a tube.

NUREG/CR-2888: OPERATOR ACTION EVENT TREES FOR THE ZION 1 PRESSURIZED WATER REACTOR. BROWN, R. G.; VONHERRMANN, J.; QUILLIAM, J. F. EG&G, Inc. October 1982. 204pp. 8212060471. EGG-2201. 16348:020.

Operator Action Event Trees for transient and LOCA initiated accident sequences at the Zion 1 PWR have been developed and documented. These trees logically and systematically portray the role of the operator throughout the progression of the accident. The documentation includes a delineation of the required operator response and the key symptoms exhibited by the plant at each state of the tree. These operator action event trees were based on the best-estimate computer analyses performed by EG&G Idaho, Inc. and Los Alamos National Laboratory under the NRC Severe Accident Sequence Analysis (SASA) Program.

NUREG/CR-2889: A TECHNICAL TEST DESCRIPTION OF THE NRC LONG-TERM WHOLE ROD AND CRUD PERFORMANCE TEST. EINZINGER, R. E.; FISH, R. L. Hanford Engineering Development Laboratory. September 1982. 77pp. 8210150550. HEDL/TME 82-32. 15726:248.

Westinghouse Hanford Company (WHC) and EG&G-Idaho are jointly conducting a long-term low-temperature spent fuel whole rod and crud behavior test to provide the Nuclear Regulatory Commission (NRC) with information to assist in the licensing of light water reactor (LWR) spent fuel dry storage facilities.

Readily available fuel rods from an H. B. Robinson (PWR) fuel assembly and a Peach Bottom 2 (BWR) fuel assembly were selected for use in the 50 month test. Both intact and defected rods will be tested in inert and oxidizing atmospheres. A 230 degrees C test temperature was selected for the first 10 month run. Both nondestructive and destructive exams are planned to characterize the fuel rod behavior during the five year test. Four interim exams and a final exam will be conducted. Crud spallation behavior will be investigated by sampling the crud particulate from the test capsule at each of the four interim exams and at the end of the test.

The background to whole rod testing, description of rod breach mechanisms, along with a detailed description of the test, are presented in this document.

NUREG/CR-2890: HISTORIC EXTREME WINDS FOR THE UNITED STATES-GREAT LAKES AND ADJACENT REGIONS. CHANGERY, M. J. Commerce, Dept. of, National Oceanographic & Atmospheric Administration. August 1982. 194pp. 8209270420. 15529:119.

Annual fastest mile wind data were extracted for the complete period of record for 70 locations in the Great Lakes, Ohio, and upper Mississippi valley regions. Existing models were used to standardize the data to 10 and 30 meters for airport-type exposures and 30 meters for city exposures. Selected probability estimates were developed from application of the Fisher-Tippette Type I extreme value model for all extracted data. Maps present the .99 probability level (100-year return period) for 10 and 30 meters for airport-type exposures, and 30 meters for city exposures.

NUREG/CR-2894: CONSOLIDATION THEORY AND ITS APPLICABILITY TO THE DEWATERING AND COVERING OF URANIM MILL TAILINGS. GATES, T. E. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1982. 48pp. 8212270456. PNL-4402. 16566:294.

This report is a review and evaluation of soil consolidation theories applicable for evaluating settlement during dewatering and subsequent covering of uranium mill tailings. Such theories may be used to predict both consolidation and water flow related effects in uranium mill tailings during drainage, following sluicing into burial pits. A consolidation theory to be useful must consider the effect of time-dependent loads, nonhomogeneous soil mass, nonlinear variation of soil properties with the stress-state parameters, large strain, and saturated and unsaturated flow. Constitutive relations linking the stress deformation state variables with void ratio should be adopted for predicting both consolidation and fluid flow interaction in unsaturated uranium mill tailings.

NUREG/CR-2897: CALCULATIONS OF HYDROGEN DETONATIONS IN NUCLEAR CONTAINMENTS BY RANDOM CHOICE METHOD. DELICHATSIOS, M.; GENADRY, M. B.; FARDIS, M. N. Massachusetts Institute of Technology. September 1982. 215pp. 8210050428. 15612:117.

Computer codes simulating hydrogen detonators in planar, cylindrical, spherical and two-dimensional axisymmetric geometries were developed. The computational method is based on the Random Choice Technique which can handle accurately sharp discontinuities. The detonation front is represented as a discontinuity changing the still unburnt gas to a completely burnt one, according to Chapman-Jouguet conditions. Results for one-dimensional geometries show good agreement with available analytical solutions. The one-dimensional code was modified to include coupling with an elastically deformable wall and used to demonstrate that for typical concrete containment structures interaction of the waves with wall deformations has insignificant effects on the wave properties, and can be neglected. The two-dimensional axisymmetric code was used to calculate pressure time histories at the wall of cylindrical containment capped with a semi-spherical dome, dimensionally similar to the Indian Point Nuclear Power Plant. The detonations simulated has initiation at either the center of the base mat or at a point on the axis at approximately two-thirds the cylinder height, and were for two different intensities. Computed pressures included repeated reflections at the walls and died out within a few tenths of a second.

NUREG/CR-2898: REINFORCED CONCRETE CONTAINMENT SAFETY UNDER HYDROGEN EXPLOSION LOADING. FARDIS, M. N.; NACAR, A.; DELICHATSIOS, M. Massachusetts Institute of Technology. September 1982. 222pp. 8210150574. 15722:252.

The effects of deflagration and detonation of hydrogen on a steel-lined reinforced concrete containment similar to Indian Point Unit 3 are studied. The structural response to a slow deflagration is computed by axisymmetric nonlinear finite element analysis. A static analysis, using average rebar properties and splices stronger than bars, was carried to 170 psig (3.62 x design pressure) giving wall displacements exceeding 20 inches. Introducing "natural variability" of reinforcing strengths, a second static analysis predicts containmen structural failure at 140 psig (2.98 x design pressure). For detonation, a conservative hydrogen accumulation assumption with ignition at floor center and another at containment midheight, on the containment axis, and nonlinear finite element dynamic analyses were used. Neglecting reinforcing material variability, midheight initiation resulted in calculated maximum dynamic strains well below the ultimate rebar strain, while initiation at floor center produced rebar failures around the dome apex. Using reinforcing material variability, all the postulated detonations were well beyond the dynamic containment capacity.

NUREG/CR-2899: ANALYSIS OF A PROPOSED ONE THOUSAND DOLLAR PER MAN-REM COST EFFECTIVENESS CRITERION. STRIP, D.R. Sandia Laboratories. November 1982. 12pp. 8212130110. SAND82-1870. 16417:155.

The Nuclear Regulatory Commission has proposed a one thousand dollar per man-rem averted ALARA criterion. This report examines the relationship between population dose and health effects, as well as onsite and offsite damage. Examination of the consequences of potential accidents at two sites shows that a site with lower population dose can have equal or greater consequences in terms of health effects and property damage. Using predicted consequences of potential accidents at 156 reactor-site combinations, the dependence of population dose and consequences is examined and its implications discussed.

NUREG/CR-2900: PREDICTED RATES OF FORMATION OF IODINE HYDROLYSIS SPECIES AT PH LEVELS, CONCENTRATIONS, AND TEMPERATURES ANTICIPATED IN LWR ACCIDENTS. BELL, J.T.; LIETZKE, M.H.; PALMER, D.A. Oak Ridge National Laboratory. November 1982. 70pp. 8212270441. ORNL-5876. 16571:021.

The literature was reviewed to define the kinetics for the reactions that follow dissolution of a molecular iodine source into water. Rate constants and rate expressions that had been determined for iodine reactions at temperatures below 60 degrees Centigrade were extrapolated by various procedures to 125 degees Centigrade. Thus, a kinetic model was developed, and computer program IRATE was written to calculate the concentrations of aqueous iodine species as a function of time. The concentrations of the significant iodine species in aqueous solutions were calculated as a function of time to 10 (9) s for six concentrations of total iodine that spanned the 10- (9) to 10- (3) g-atom/L range at 25, 60, 100, and 125 degrees Centigrade, and with eight pH conditions that spanned the 3 to 11 range. Corresponding partition coefficients were calculated for selected conditions, with the assumption that only the I (2) and HOI species are volatile. The results are presented in the form of 35 figures, which contain a total of 93 plots.

NUREG/CR-2903: AN INDEPENDENT ASSESSMENT OF EVALUATION TIME ESTIMATES FOR A PEAK POPULATION SCENARIO IN THE EMERGENCY PLANNING ZONE OF THE SEABROOK POWER STATION. MOELLER, M.P.; MCLEAN, M.A.; URBANIK, T.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1982. 87pp. 8212060457. PNL-4290. 16346:109.

This study comprises two major tasks: (1) an assessment of the methods and assumptions used in calculating evacuation time estimates (ETEs) applicable to the general population for a peak population scenario in the emergency planning zone of the Seabrook Nuclear Power Station, consisting of a review and analysis of previous work by Public Service of New Hampshire (PSNH) and the Federal Emergency Management Agency (FEMA), as well as an independent calculation of ETEs using the CLEAR model for the demographic data reported by PSNH; (2) independent estimations of the ETEs for the peak population scenario, developed using demographic data prepared by the U.S. Nuclear Regulatory Commission (NRC) and the CLEAR model. The results of this study reveal the importance of the assumptions used for calculating ETEs. Because traffic routings and management plans have not been prepared for the area, the CLEAR calculations utilized independently prepared traffic routings and assumptions. A detailed analysis of the results suggests that the ETEs submitted by PSNH are consistent with the methods and assumptions which provide the bases for PSNH's ETEs. Differences among ETEs stem largely from differences in the assumed size of the evacuating population and the estimated effectiveness of traffic controls.

NUREG/CR-2905: RAMAN SCATTERING TEMPERATURE MEASUREMENTS FOR WATER VAPOR IN NON-EQUILIBRIUM DISPERSED TWO-PHASE FLOW. ANASTASIA, C.M.; NETI, S.; SMITH, W.R.; et al. Lehigh Univ. September 1982. 204pp. 8210050368. 15624:061.

The objective of this investigation was to determine the feasibility of using Raman scattering to measure vapor temperatures in dispersed two-phase flow as an alternative nonintrusive technique. The Raman system developed for this investigation is described, including alignment of optics and optimization of the photodetector for photon pulse counting. Experimentally obtained Raman spectra are presented for the following single and two-phase samples: liquid water, atmospheric nitrogen, superheated steam, nitrogen and water droplets in a high void fraction air/water mist, and superheated water vapor in nonequilibrium dispersed flow.

NUREG/CR-2908: FAULTING IN SOUTHWEST INDIANA. AULT, C.H.; SULLIVAN, D.M.; MACKEY, J.C.; et al. Indiana Geological Survey. October 1982. 59pp. 8211030208. 15935:260.

This is the final report of the multi-year program of geologic and geophysical investigations of faulting in southwestern Indiana. During the first 2 years of the investigation in Indiana, mapping and characterization of the Wabash Valley Fault System were accomplished using data mostly from numerous geophysical logs of closely drilled wells in Posey and Gibson counties. Both surface and subsurface investigations were conducted in a second area of known faulting in Perry and Spencer counties. Finally, test drilling and concentrated field mapping of the Mt. Carmel Fault in southcentral Indiana completed the study. The report includes interpretation of the relationship of the faulting in southwestern Indiana to regional structure and tectonics, especially with regard to association with the New Madrid Fault Zone and its potential extension to the northwest.

NUREG/CR-2909: PALEOZOIC GEOLOGY OF THE NEW MADRID AREA. SCHWALB, H.R. Illinois, State of. * Illinois, Univ. of, Urbana-Champaign. September 1982. 69pp. 8210210051. 15785:185.

This is the final report on the geologic and geophysical study of the New Madrid area by the Illinois State Geological Survey. The area of investigation for this study encompassed portions of eight states within a two hundred mile radius around New Madrid, Missouri. The objectives of the study were to provide a better understanding of the regional structure and tectonics of the New Madrid zone through geologic time by understanding the Paleozoic geology.

NUREG/CR-2911: MULTIROD BURST TEST PROGRAM PROGRESS REPORT FOR JANUARY-JUNE 1982. CHAPMAN, R.H. Oak Ridge National Laboratory. December 1982. 82pp. 8301120087. ORNL/TM-8485. 16780:001.

The B-6 (8 x 8) array was tested, and posttest examination was completed; data reduction and analysis are in progress. Preliminary quick-look results are included in this report. All 64 rods were pressurized and burst. The average burst temperature was 931 degrees Centigrade, and the bundle average heating rate 3.5 degrees Centigrade/s during the time of deformation. Preliminary results indicate burst strains ranged from 22 to 56%, with a bundle average of 30%.

Analysis of the B-5 test results continues to provide insight to the complexity of cladding deformation in bundles, particularly for conditions conducive to large deformation and rod-to-rod interactions. Additional analyses, including re-evaluation of burst temperatures, are included in this report.

The B-6 test concluded the experimental phase of this research program. Future activities will be concerned with analysis and evaluation of experimental data produced by this and other research programs.

NUREG/CR-2912: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A FEEDWATER LINE BREAK EXPERIMENTS (TESTS S-SF-1, S-SF-2 AND S-SF-3). O'CONNELL, T.M. EG&G, Inc. October 1982. 45pp. 8211120160. EGG-2216. 16066:298.

This report presents recorded test data for Tests S-SF-1, S-SF-2, and S-SF-3 of the Semiscale Mod-2A Steam and Feedwater Line Break Experiment Series. These three tests were used to investigate the thermal-hydraulic phenomena resulting from ruptures in the feedwater line of a pressurized water reactor system, with evaluation focusing on the primary system overpressurization phenomena resulting from the sudden loss of heat sink in the affected steam generator. Tests S-SF-1, -2, and -3 were conducted using initial conditions and operating sequences duplicating, as nearly as possible, those of a typical pressurized water reactor (PWR) undergoing a feedwater line rupture with consequent scram and coincident loss-of-offsite power. This report presents, for future analysis, the uninterpreted data from each of the three tests. The data, presented as graphs in engineering units, have been analyzed only to the extent necessary to insure their validity as actual measurements of the system phenomena.

NUREG/CR-2914: CORRELATION OF LIGNITE BEDS FOR FAULT IDENTIFICATION IN THE MISSISSIPPI EMBAYMENT AREA OF WESTERN KENTUCKY. COBB, J.C.; WILLIAMS, D.A. Kentucky, State of. September 1982. 36pp. 8210210046. 15786:012.

This is the final report of geologic and geophysical investigations of faulting in western Kentucky. Structure mapping based on seismic-reflection profiling and earthquake plotting in the Jackson Purchase Region of western Kentucky indicated lineaments, faults, and earthquake epicenters in the deposits of the Mississippi Embayment. New Madrid seismic activity has been linked to these structures which have been reactivated through geologic time (Zoback and others, 1980). The area of study was selected based on existence of information, proximity to lineaments, proximity to epicenter, adequate penetration to lignite beds and access to drill sites. The report includes an interpretation of the relationship of faulting in the western Kentucky study area to the regional structure and tectonics.

NUREG/CR-2916: AN EMPIRICAL EXAMINATION OF EVALUATION METHODS FOR COMPUTER GENERATED DISPLAYS:PSYCHOPHYSICS. PETERSEN, R. J. ; SMITH, R. L. ; BANKS, W. W. EG&G, Inc. October 1982. 46pp. 8211110636. EGG-2214. 16050:105.

An investigation was performed to evaluate the perceptual aspects of safety parameter display systems (SPDSs) in nuclear power plant control rooms. Three SPDS configurations (star, bars, and meters) were evaluated in a series of four experiments. Subjects for the investigation were qualified nuclear plant operators and engineers at EG&G Idaho, Inc., at the Idaho National Engineering Laboratory. The techniques reported herein were demonstrated to be sensitive to differences in human performance which result from using different display formats to present the same safety parameter information.

NUREG/CR-2917: REVIEW OF GROUND-WATER FLOW AND TRANSPORT MODELS IN THE UNSATURATED ZONE. OSTER, C. A. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1982. 175pp. 8212270559. PNL-4427. 16546:042.

Models of partially saturated flow and transport in porous media have application in the analysis of existing as well as future low-level radioactive waste facilities located above water table. An extensive literature search along with telephone and mail correspondence with recognized leading experts in the field, was conducted to identify computer models suitable for studies of low-level radioactive waste facilities located in the unsaturated zone. Fifty-five existing models were identified as potentially useful. Ten of these models were selected for further examination.

NUREG/CR-2918: SOIL-STRUCTURE INTERACTION ANALYSIS OF A 5 KG HDR EXPLOSIVE TEST. KOT, C. A. ; SRINIVASAN, M. G. ; VAUGHAN, D. K. ; et al. Argonne National Laboratory. November 1982. 118pp. 8212130147. ANL-82-53. 16415:044.

This report summarizes a nonlinear 3-D finite-element, soil-structure interaction analysis of dynamic response to a 5 kg explosive blast test performed on the containment building of the Heissdampfreaktor (HDR) in Germany in 1979. The modeling of the explosive source, site, and building for use in the TRANAL computer code was partialy based on information obtained from previous tests. Comparison of analytical results with the test measurements shows that the analysis simulates superstructure response better than it does the foundation response. This is attributed to the uncertainties associated with the source and site modeling. Considering the complexities of the soil-structure system, the correlation between analysis and test results is reasonably good. The use of information from prior dynamic tests is seen to have enhanced the predictive capability of the analytical model.

NUREG/CR-2919: USER GUIDE FOR XOQDOQ:EVALUATING ROUTINE EFFLUENT RELEASES AT COMMERCIAL NUCLEAR POWER STATIONS. SAGENDORF, J. F. ; GOLL, S. T. ; SANDUSKY, W. F. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1982. 158pp. 8210220149.

PNL-4380. 15795:167.

Provided is a user's guide for the U. S. Nuclear Regulatory Commission's (NRC) computer program XOQDOQ which implements Regulatory Guide 1.111. This NUREG supercedes NUREG-0324 which was published as a draft in September 1977. This program is used by NRC meteorology staff in their independent meteorological evaluation of routine or anticipated intermittent releases at nuclear power stations. It operates in a batch input mode and has various options a user may select. Relative atmospheric dispersion and deposition factors are computed for 22 specific distances out to 50 miles from the site for each directional sector. From these results, values for 10 distance segments are computed. The user may also select other locations for which atmospheric dispersion deposition factors are computed. Program features, including required input data and output results, are described. A program listing and test case data input and resulting output are provided.

NUREG/CR-2920: BEHAVIOR OF REINFORCED CONCRETE SLABS SUBJECTED TO COMBINED PUNCHING SHEAR AND BIAXIAL TENSION. JAU, W. C.; WHITE, R. N.; GERGELY, P. Cornell Univ. September 1982. 103pp. 8210050412. 15624:266.

This report summarizes experimental results on peripheral (punching) shear strength of precracked, biaxially tensioned, orthogonally reinforced concrete slabs. The 6 inch thick, 4 foot square test specimens were designed to simulate the stress state in the wall of a pressurized containment vessel subjected to combined internal pressurization (which produces biaxial tension) and punching loads normal to the wall. Seven specimens were tested to study the influence of level of biaxial tension, shear span, size of loaded area, and reinforcing ratio on punching shear strength. The levels of biaxial tension, 0 and 0.8fy, were used.

The test results, supplemented by those from a similar study, are used to formulate new recommendations for punching shear design criteria that are much less conservative than existing design approaches. The report also includes a review of pertinent literature, a comparison of test results with selected predictions for the slabs with zero biaxial tension, and a finite element analysis to study the influence of edge restraint on slab behavior.

NUREG/CR-2922: COMPARISON OF QUICK PREDICTIONS WITH RESULTS OF SELECTED RECENT AEROSOL BEHAVIOR EXPERIMENTS. JORDAN, H.; SCHUMACHER, P. M.; GIESEKE, J. A. Battelle Memorial Institute, Columbus Laboratories. September 1982. 37pp. 8210150592. BMI-2099. 15726:324.

A detailed comparison of QUICK aerosol behavior code predictions with four representative experiments (AB-1, AB-3, NSPP-104, NSPP-203) is presented. It is shown that the three sodium fire expriments of this set can be used to evaluate and confirm values of the two aerosol particle shape factors for these systems (X = 1.3, Y = 5.0). In addition, it is shoun that uncertainty in prediction due to uncertainty in the two shape factors is not augmented in translation from experimental to full scale.

Whi. e single uranium experiment suggests the doublet X = 3.0, Y = 15.0, se values are more equivocal since for this system wall plate-out is significant factor. Large uncertainty in plate-out data resulted in c comitant uncertainty in the fitted values of the shape factors.

NUREG/CR-2923. MSPEC USER'S MANUAL. JORDAN, H. ; SCHUMACHER, P. M. ; GIESEKE, J. A. Battelle Memorial Institute, Columbus Laboratories. September 1982. 53pp. 8210210014. BMI-2100. 15784:324.

The MSPEC aerosol behavior code for a multiple species, singly contained aerosol is described. The model equation used is presented and the numerical solution technique -- modified finite difference coupled with the GEAR ordinary differential equation solver -- is discussed briefly. A short description of each subroutine is presented. Input required by MSPEC is given in detail, as is the result of a sample run. MSPEC is intended to provide a method for estimating aerosol behavior in a homogeneously mixed nonflowing containment atmosphere. The aerosol particles are assumed to be characterizable by representative particles in each of a finite set of size classes. Large uncertainty rests with the model employed in determining the shape factors associated with a given size class and composition. This model can, however, be readily modified.

NUREG/CR-2927: NUCLEAR POWER PLANT ELECTRICAL CABLE DAMAGEABILITY EXPERIMENTS. LUKERS, L. L. Sandia Laboratories. November 1982. 44pp. 8212220291. SAND82-0236. 16523:263.

Under the direction of the Nuclear Regulatory Commission, Sandia National Laboratories has been conducting confirmatory research in fire protection for nuclear power plants. As a part of this research, a program was developed to determine the damageability of electrical cable insulation to thermal radiation in a loaded cable tray. The critical flux or threshold level at which cable damage occurs in the form of electrical failure (short from conductor to tray) and nonpiloted ignition was determined for two types of electrical cable, one an IEEE-383 qualified cable and the other an unqualified cable. The critical flux for electrical failure was determined to be about 18 kW/m (2) for the IEEE-383 qualified cable and about 8 kW/M (2) for the unqualified cable. The critical flux for nonpiloted ignition was determined to be about 28 kW/m (2) for the IEEE-383 qualified cable and about 22 kW/m (2) for the unqualified cable. A program was also developed to determine the damageability of electrical cable insulation to constant temperature, thermal exposure. Thermal aging and radiation exposure efforts were not included in the investigation.

NUREG/CR-2930: EXPERIMENT DATA REPORT FOR LOFT ANTICIPATED TRANSIENT EXPERIMENT SERIES L6-8. JARRELL, D. B. ; DIVINE, J. M. ; MCKENNA, K. J. EG&G, Inc. November 1982. 278pp. 8212130152. EGG-2219. 16414:088.

Selected pertinent and uninterpreted data from anticipated transient Experiment Series L6-8, conducted in the Loss-of-Fluid Test (LOFT) facility, are presented. The operation of the LOFT system is typical of large [1000 MW(e)] commercial PWR operations. Experiment Series L6-8 consisted of six experiments that were independently conducted simulations of transients which have a high probability of occurrence in a commercial PWR. The type of transient and major significant events for each of the six experiments are summarized in the report. Experiment L6-8B-1 was an uncontrolled rod withdrawal with a 0.47 cent/$ reactivity insertion rate. Experiment L6-8B-2 was an uncontrolled rod withdrawal with a 5.5 cent/$ reactivity insertion rate. Experiment L6-8C-1 was recovery from a simulated steam generator tube rupture with the primary coolant pumps (PCPs) remaining on. Experiment L6-8C-2 was a recovery from a simulated steam generator tube rupture with the PCPs being tripped on loss of pressurizer liquid level indication. Experiment L6-8C-3 was a recovery from a small break with

PCPs void formation. The break flow was the same as for Experiments L6-8C-1 and L6-8C-2. Experiment L6-8D was a PCS natural circulation cooldown with void formation.

NUREG/CR-2932 V01: EQUIPMENT QUALIFICATION RESEARCH TEST OF ELECTRIC CABLE WITH FACTORY SPLICES AND INSULATION REWORK TEST NO. 2, REPORT NO. 1. MINOR, E.E.; FURGAL, D.T. Sandia Laboratories. October 1982. 79pp. 8212060486. SAND81-2027. 16344:128.

Electric cables with flame-retardant chemically crosslinked polyolefin extruded insulation containing factory-made center-conductor splices and insulation repairs manufactured by The Rockbestos Company were used in a methodology test of the IEEE Standard 383-1974. This standard is concerned with the ability of cables to function during and following exposure to aging and LOCA/MSLB environments. Cable specimens were radiation aged at a low-dose rate and then thermally aged to simulate a 40-year containment exposure. After aging, the specimens were subjected to LOCA radiation and a 33-day steam and chemical spray exposure. The cables were electrically loaded and functioned without failure during and after LOCA steam and chemical spray exposure. Insulation resistence measurements were taken during the exposure sequence. Subsequent to the exposures, hipot and mandrel bend test were conducted. Test results indicate that the methods given in IEEE 383-1974 are adequate to show that cables can function and support power and control operations during and after a LOCA/MSLB of the severity simulated by the test. Further, the presence of center-conductor splices and insulation repairs did not appear to degrade cable performance.

NUREG/CR-2933: NUCLEAR FUEL CYCLE RISK ASSESSMENT: SURVEY AND COMPUTER COMPILATION OF RISK RELATED LITERATURE. YATES, K.R.; SCHREIBER, A.M.; RUDOLPH, A.W. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1982. 457pp. 8211080011. PNL-4350. 16016:001.

The U.S. Nuclear Regulatory Commission has initiated the Fuel Cycle Risk Assessment Program to provide risk assessment methods for assistance in the regulatory process for nuclear fuel cycle facilities other than reactors. Both the once-through cycle and plutonium recycle are being considered. A previous report was generated by this program. This report, the second from the program, describes the survey and computer compilation of fuel cycle risk-related literature. Sources of available information on the design, safety, and risk associated with the defined set of fuel cycle elements were searched and documents obtained were catalogued and characterized with respect to fuel cycle elements and specific risk/safety information. Both U.S. and foreign surveys were conducted. Battelle's computer-based BASIS information management system was used to facilitate the establishment of the literature compilation. A complete listing of the literature compilation and several useful indexes are included. Future updates of the literature compilation will be published periodically.

NUREG/CR-2934: REVIEW AND EVALUATION OF THE INDIAN POINT PROBABILISTIC SAFETY STUDY. KOLB, G.J.; BERRY, D.L.; EASTERLING, R.G.; et al. Sandia Laboratories. December 1982. 480pp. 8301190377. SAND82-2929. 16856:184.

This report summarizes the review of the internal and external event portions of the Indian Point Probabilistic Safety Study (IPPSS). The review was conducted by Sandia National Laboratories and Sandia contractors over approximately a 6-month period. The purpose of the review was to search for areas in the IPPSS where omissions and critical judgments were made which could impact the quantitative results. The review identified several of these areas. This report also evaluated, in a preliminary way, some proposed Indian Point plant modifications which were based on the insights of the IPPSS but were not included in the IPPSS results. A comparison of the quantitative results in this report, which assumes the plant modifications are in place, with the IPPSS results yields less than a factor of two difference on the overall plant difference on the overall plant core melt frequency.

NUREG/CR-2942: CRT DISPLAY EVALUATION: THE MULTIDIMENSIONAL RATING OF CRT GENERATED DISPLAYS. GERTMAN, D. I.; BLACKMAN, H. S.; BANKS, W. W.; et al. EG&G, Inc. November 1982. 43pp. 8212130108. EGG-2220. 16415:209.

This report is one in a series evaluating various methods for determining the effectiveness of display formats and contains results from multidimensional rating of three cathode ray tube displays. Each display contained information regarding the status of nine safety parameters critical to safe operations at the Loss-of-Fluid Test (LOFT) reactor located at the Idaho National Engineering Laboratory (INEL). Certified LOFT operators evaluated three formats-bars, stars, and meters-as part of a multimethod display evaluation ongoing at INEL. Each of the six cognitive dimensions embedded in the rating scale are discussed in terms of their internal consistency and ability to differentiate between each of the three formats. Preliminary findings suggest that two of the six dimensions, content integration and cognitive processing, were able to discriminate between formats while satisfying the criteria of internal consistency as measured by Cronbach's alpha. Conclusions in this report must be interpreted in light of moderate test-retest reliability.

NUREG/CR-2946: THE LONG-TERM STABILITY OF EARTHEN MATERIALS IN CONTACT WITH ACIDIC TAILINGS SOLUTIONS. PETERSON, S. R.; ERIKSON, R. L.; GEE, G. W. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1982. 82pp. 8301100005. 16714:126.

The objectives of the studies documented in this report were to use experimental and geochemical computer modeling tools to assess the long-term environmental impact of leachate movement from acidic uranium mill tailings. Liner failure (i.e., an increase in the permeability of the liner material) was found to be a problem when various acidic tailings solutions leached through liner materials for periods up to 3 years. On the contrary, materials that contained over 30% clay showed a decrease in permeability with time in the laboratory columns. The high clay materials tested appear suitable for lining tailings impoundment ponds. The decreases in permeability are attributed to pore plugging resulting from the precipitation of minerals and solids. This precipitation takes place due to the increase in pH of the tailings solution brought about by the buffering capacity of the soil. Geochemical modeling predicts, and X-ray characterization confirms, that precipitation of solids from solution is occurring in the acidic tailings solution/liner interactions studied. In conclusion the same mineralogical changes and contaminant reactions predicted by geochemical modeling and observed in laboratory studies were found at a drained evaporation pond (Lucky Mc in Wyoming) with a 4 year history of acid attack.

NUREG/CR-2948 SIGNIFICANCE OF NICKEL AND COPPER CONTENT TO RADIATION SENSITIVITY AND POSTIRRADIATION HEAT TREATMENT RECOVERY OF REACTOR VESSEL STEELS. HAWTHORNE, J.R. ENSA, Inc. November 1982. 26pp. 8212270499. MEA-2006. 16548:137.

This paper describes an experimental test in confirmation of a suspected interaction between nickel alloying and copper impurities in steel radiation sensitivity development. The study was prompted by differences observed in radiation embrittlement trends for high nickel, high copper content versus low nickel, high copper content welds and for A533-B (0.4 to 0.7 percent Ni) versus A302-B (<<0.4 percent Ni) steel plates. The study also explored the influence of nickel content on notch ductility recovery attainable by a low temperature heat treatment after irradiation.

Eight steel plates with variations in nickel content from 0.002 to 0.69 percent and with variations in copper content from 0.005 to 0.28 percent were used for the investigation. The plates were produced from two (4-way split) laboratory melts. Plate radiation embrittlement sensitivities were determined from Charpy-V notch ductility changes by 288 degrees Centigrade irradiation to 2.5 X 10 (19) n/cm (2). Experimental observations show that a 0.7 percent but not a 0.3 percent nickel content can be detrimental to the radiation resistance of steel; the effect, however, is clearly dependent on steel copper content. Nickel content was also found to influence the amount of transition temperature recovery produced by a postirradiation 399 degrees Centigrade hour heat treatment.

NUREG/CR-2949: TRANSPORTATION OF RADIOACTIVE MATERIAL IN WASHINGTON STATE. September 1980-September 1981. GRONEMYER, E.L.; INGERSOLL, C.E.; MATTHEWS, S.P.; et al. Washington, State of, Dept. of Social & Health Services. October 1982. 80pp. 8211020033. 15904:082.

The receipt and survey of shipments of low-level radioactive waste at the U.S. Ecology disposal facility near Richland, Washington, continued during the second year of the joint NRC/DOT contract. Washington State Radiation Control personnel inspected essentially all incoming shipments of radioactive waste to assure compliance with appropriate DOT shipping regulations, State of Washington Radiation Control regulations, and the conditions of U.S. Ecology's radioactive materials license. Surveillance activities associated with this contract have shown that transportation violations have decreased significantly since 1979. Other than waste, essentially all other shipments or radioactive materials through and within the state of Washington are radiopharmaceuticals. Especially in the Seattle area, the largest quantity of shipments involve radiopharmaceuticals transported via the state's only licensed nuclear pharmacy. Contract studies involved inspections at all facilities or operations in the state handling radioactive materials. Radiation exposures to transportation workers were studied and it was determined that excessive radiation was not being received by individuals in the course of transporting radioactive materials.

NUREG/CR-2960: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A STEAM LINE BREAK EXPERIMENTS (TESTS S-SF-4 AND S-SF-5). LARSON, R.A. EG&G, Inc. November 1982. 68pp. 8212060468. EGG-2224. 16348:224.

This report presents test data recorded for Tests S-SF-4 and S-SF-5 of the Semiscale Mod-2A Primary Steam and Feedwater Line Break Experiment Series. These tests are part of a series of Semiscale tests that investigate the thermal-hydraulic phenomena resulting from operational transients involving rupture of the steam line piping on the secondary side of a pressurized water reactor. Experimental data from the tests are used to develop and assess the analytical capabiltiy of computer models used to predict the results of such a rupture in the steam line piping and evaluate the operational procedures involved in system recovery. The primary objectives of the tests were to determine the effects of a secondary-side transient on the the primary side of the system, with 100% (S-SF-4) and 50% (S-SF-5) ruptures of the main steam line piping, and to provide data for water reactor safety codes and scoping for future tests. This report presents the uninterpreted data from Tests S-SF-4 and -5 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-2961: EXPERIMENT DATA REPORT FOR SEMISCALE MOD-2A PRIMARY FEED AND BLEED EXPERIMENT SERIES (TESTS S-SR-1 AND S-SR-2). FOGDALL, S.P. EG&G, Inc. November 1982. 64pp. 8212270454. EGG-2225. 16572:060.

This report presents test data recorded for Tests S-SR-1 and S-SR-2 of the Semiscale Mod-2A Primary Feed and Bleed Tests. 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) or abnormal operating transient. These tests provide experimental data for assessing the analytical capability of computer codes used in LOCA and operational transient analysis. The primary objectives of Tests S-SR-1 and -2 were to provide data on primary system recovery through the use of primary feed and bleed cooling, with no heat transfer to the secondaries. Data was obtained using high- and low-head pump curves for the safety injection (SI) pumps. This report presents the uninterpreted data from Tests S-SR-1 and -2 for analysis.

NUREG/CR-2966: BUCKLING INVESTIGATION OF RING-STIFFENED CYLINDRICAL SHELLS UNDER UNSYMMETRICAL AXIAL LOADS. BAKER, W.; BENNETT, J.; BABCOCK, C. Los Alamos Scientific Laboratory. December 1982. 82pp. 8302040468. LA-9536-MS. 17044:001.

Two buckling experiments are described in detail. The first purpose of these experiments is to establish baseline values for the buckling loads and modes for ring-stiffened cylindrical shells that have geometric parameters characteristic of the ring-stiffened cylindrical section of a typical nuclear steel containment vessel. A series of follow-on experiments on nominally identical ring-stiffened cylinders that have framed and reinforced penetrations typical of nuclear industry practice are planned and will be described in a separate report. This report is issued separately because of the volume of detail included on methods, measurements, and data reduction techniques. In addition, a second purpose of these experiments is to serve as a set of benchmark experiments for computer codes that can predict buckling loads for axisymmetric geometries, and in this respect this work stands alone. Complete material stress-strain curves are described and complete geometric imperfection data for the shells are given. These data are also available in digitized form on tape. Typical load-strain information is reported, as well as the distribution of load around the cylinder at buckling. A method for taking and reducing chord gage measurements to characterize the out-of-roundness imperfections is also examined in detail and comparisons to actual imperfection sweep measurements are made.

NUREG/CR-2970 V01: MATERIALS SCIENCE DIVISION LIGHT-WATER REACTOR SAFETY RESEARCH PROGRAM: Quarterly Progress Report. * Argonne National Laboratory. November 1982. 131pp. 8212270468. ANL-82-41. 16570:102.

This progress report summarizes the Argonne National Laboratory work performed during January, February, and March 1982 on water-reactor-safety problems. The research and development areas covered are Environmentally Assisted Cracking in Light Water Reactors, Transient Fuel Response and Fission Product Release, and Clad Properties for Code Verification.

NUREG/CR-2971: AN EXPERIMENTAL INVESTIGATION OF BOILING WATER NUCLEAR REACTOR PARALLEL CHANNEL EFFECTS DURING A POSTULATED LOSS-OF-COOLANT ACCIDENT. CONLON, W. M.; LAHEY, R. T. Rensselaer Polytechnic Inst. December 1982. 877pp. 8301120080. 16787:001.

This report describes an experimental study of the influence of parallel channel effects (PCE) on the distribution of emergency core spray cooling water in a Boiling Water Nuclear Reactor (BWR) following a postulated design basis loss of coolant accident (LOCA). The experiments were conducted in a scaled test section in which the reactor coolant was simulated by Freon-114 at conditions similar to those postulated to occur in the reactor vessel shortly after a LOCA.

NUREG/CR-2975: SOIL-TO-PLANT CONCENTRATION FACTORS FOR RADIOLOGICAL ASSESSMENTS. NG, Y. C.; COLSHER, C. S.; THOMPSON, S. E. Lawrence Livermore Laboratory. November 1982. 140pp. 8212160773. UCID-19463. 16464:286.

This report presents the results of a literature review to derive soil-to-plant concentration factors to predict the concentration of a radionuclide in plants from that in soil. The concentration factor, B (v) is defined as the ratio of the concentration of a nuclide in the edible plant part to that in dry soil. CR (the concentration ratio) is similarly defined to denote the concentration factor for dry feed consumed by livestock. B (v) and CR values are used to assess the dose from radionuclides deposited onto soil and transferred into crop plants via roots. Approaches for deriving B (v) and CR values are described, and values for food and feed are tabulated for individual elements. The sources of uncertainty are described, and the factors that contribute to the inherent variability of the B (v) and CR values are discussed. Summary tables of elemental B (v) and CR values and statistical parameters that characterize their distributions provide a basis for a systematic updating of many of the B (v) values in Regulatory Guide 1.109. They also provide a basis for selecting B (v) and CR values for other applications that involve the use of equilibrium models to predict the concentration of radionuclides in plants from that in soil.

NUREG/CR-2976: TRANSFER COEFFICIENTS FOR ASSESSING THE DOSE FROM RADIONUCLIDES IN MEAT AND EGGS. NG, Y. C.; COLSHER, C. S.; THOMPSON, S. E. Lawrence Livermore Laboratory. November 1982. 116pp. 8212140464. UCID-19464. 16420:251.

This report presents the results of a literature review to derive transfer factors to predict the transfer of radionulides from contaminated vegetation to meat and eggs. Updated estimates of transfer coefficients from feed to beef, pork, lamb, chicken and hens' eggs are presented for a large number of radionuclides associated with the nuclear fuel cycle. The transfer coefficient (F(f)) meat is defined as the fraction of the nuclide ingested daily by an animal that is found in a kilogram of muscle from the animal under steady-state or equilibrium conditions or at the time of slaughter. The (F(f)) to eggs is similarly defined. Both direct approaches and indirect approaches using collateral data were used to derive transfer coefficients. The sources of uncertainty are described, and where sufficient data exist, the inherent variability of (F(f)) values is characterized by statistical parameters of the distributions. Summary tables of elemental transfer coefficients from feed to beef, pork, lamb, chicken and hens' eggs are presented. These estimates, along with a discussion of the associated uncertainties, provide a basis for systematically updating the values of (F(f)) (beef) in Regulatory Guide 1.109 and for deriving (F(f)) values for other applications that involve the use of equilibrium models to predict concentrations of radionuclides in animal products.

NUREG/CR-2978: EXPERIMENT DATA REPORT FOR LOFT ANTICIPATED TRANSIENT WITHOUT SCRAM EXPERIMENT 19-4. BATT, D. L.; DIVINE, J. M.; MCKENNA, K. J. EG&G, Inc. November 1982. 87pp. 8212270145. EGG-2227. 16550:001.

Selected pertinent and uninterpreted data from the fourth anticipated transient with multiple failures experiment (Experiment L9-4) conducted on September 24, 1982, 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's thermal-hydraulic and nuclear conditions. The operation of the LOFT system is typical of large [1000 MW (e)], commercial PWR operations. Experiment L9-4 simulated a loss-of-offsite-power anticipated transient without reactor scram. The loss-of-offsite-power accident led to an increase in the primary coolant system temperature and pressure. The experiment safety relief valve opened and was able to limit and control the pressure transient. In addition, subsequent heat generation was dissipated by the auxiliary feedwater flow in the secondary coolant system until the reactor was scrammed at experiment termination.

NUREG/CR-2982: BUOYANCY, TRANSPORT, AND HEAD LOSS OF FIBROUS REACTOR INSULATION. BROCARD, D. N. Alden Research Laboratory. November 1982. 59pp. 8212270520. SAND82-7205. 16548:017.

In the event of a loss of coolant accident (LOCA) in a nuclear power plant, it is possible that insulation for pipes or other items inside the containment building could be dislodged by the high energy break jet. This insulation debris could affect the recirculation of water from the sump of the emergency core cooling system (ECCS) by collecting on the screen surrounding this sump. To help in assessing the possible effect of detached insulation on the ECCS, bouyancy, transport, and head loss characteristics of the insulation were studied experimentally. Three types of insulation pillows were tested in undamaged, opened, broken up in pieces and shredded conditions. Small samples of reflective metallic and closed cell insulations were also tested for transport and buoyancy. This report contains experimental data which can be used to assess insulation debris transport and screen blockage effects.

NUREG/CR-2985: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Project Summary Report, Elk River Reactor. MILLER, R. L.; ADAMS, J. A. United Nuclear Corp. December 1982. 49pp. 8301100006. 16714:206.

This report summarizes information concerning the decommissioning of the Elk River Reactor. Decommissioning data from available documents were input into a computerized data-handling system in a manner that permits specific information to be readily retrieved. The information is in a form that assists the Nuclear Regulatory Commission in its assessment of decommissioning alternatives and ALARA methods for future decommissioning projects. Samples of computer reports are included in the report. Decommissioning of other reactors, including NRC reference decommissioning studies, will be described in similar reports.

NUREG/CR-2988: AN APPROACH TO MODELING SUPERVISORY CONTROL OF A NUCLEAR POWER PLANT. BARON, S. ; FEEHRER, C. ; MURALIDHARAN, R. ; et al. Bolt, Beranck & Newman, Inc. December 1982. 156pp. 8302040521. ORNL/SUB/81-705. 17046:001.

This report describes the results of a study aimed at determining the feasibility of applying a supervisory control modelling technology to the study of critical operator-machine problems in the operation of a nuclear power plant. The report includes brief overviews of various alternative approaches to the modelling of human performance, and different perspectives on the roles of operators in process control activities like those represented in a power plant. The result of the study is a conceptual model that incorporates the major elements of the operator and of the plant to be controlled. The operator portion of the model is developed at a block diagram level and includes several algorithms that are considered suitable for use with various model components. The plant portion of the model is developed from literature available on plant dynamics and is of the "first-principles" type. Both models are presented in detail sufficient for demonstrating the feasibility of developing a quantitative supervisory control model, for outlining the requirements for data to build and operate such a model, and for discussing its potential applications.

NUREG/CR-2991: EVALUATION OF POTENTIAL SURFACE FAULTING AND OTHER TECTONIC DEFORMATION. BONILLA, M. G. Interior, Dept. of, Geological Survey. October 1982. 64pp. 8211190354. 16167:305.

This report summarizes and provides references to much of what is known about tectonic deformation associated with earthquakes and describes current approaches and procedures for evaluating the hazards of surface faulting and other earthquake-related tectonic deformation. Emphasis is placed on surface faulting because it is the more significant hazard for most construction.

The deformation discussed here is the permanent deformation of the ground arising from the sudden displacement of buried rock masses that generates an earthquake. Such deformation includes both faulting that ruptures the surface of the earth and permanent distributed deformation of the rocks surrounding the earthquake-generating fault.

Surface faulting also results from slow movement of large sedimentary deposits (for example, the "growth faults" of the Gulf Coast), from withdrawal of subsurface fluids, or from movement of salt, gypsum, or anhydrite deposits.

NUREG/CR-2995: CANADIAN SEISMIC AGREEMENT. Annual Report (1 July 1981-30 June 1982). LYANS, J. A. ; WETMILLER, R. J. ; ANDERSON, F. Canadian Commercial Corp. November 1982. 32pp. 8212220249. 16526:152.

This is the fourth annual progress report under terms of the Contract No. NRC-04-79-180 for a Research Agreement entitled "Canadian Seismic Agreement" between the U.S. Nuclear Regulatory Commission and the Canadian Commercial Corporation. Activities undertaken by the Earth Physics Branch during the past year, and supported in part by the Agreement, are described in this report.

During the past year, the resources were used for telephone line rentals, capital expenditures and non-recurring research and development costs. Data concentrators were established at Hauterive and Riviere-du-Loup in Quebec and St. John in New Brunswick to handle new stations in eastern Quebec and the Maritimes. Six new stations have been installed at Hauterive and Grosses-Roches, Quebec and Edmunston, Caledonia Mountain, St. George and Mount McKendrick, New Brunswick.

A new version of the ECTN MKIII software capable of handling up to 26 seismic components went into full production in the Ottawa central computer site in October, 1981. The ECTN is gradually becoming a mature and stable network. In the coming year most of the resources will be devoted to improving the performance of the network by installing the new concentrator software at remote sites, improving the signal margin in the radio links and further developing the data analysis and maintenance software.

NUREG/CR-2997 V01: APPLICATIONS OF ENERGY RELEASE RATE TECHNIQUES TO PART-THROUGH CRACK IN PLATES AND CYLINDERS. BASS, B. R.; BRYSON, J. W. Oak Ridge National Laboratory. December 1982. 71pp. 8302040502. ORNL/TM-8527/V1. 17046:220.

This report (Volume 1) describes the ORMGEN-3D (Oak Ridge Mesh GENerator - 3D) finite element mesh generator program for computational fracture mechanics anaylsis. The program automatically generates a three-dimensional (3-D) finite element model for six different crack geometries. These geometries include flat plates with straight or curved surface cracks and cylinders with part-through cracks on the outer or inner surface. Mathematical or user-defined crack shapes may be considered. ORMGEN-3D generates a core of special wedge or collapsed prism elements at the crack front to introduce the appropriate stress singularity at the crack tip. Regular 20-node isoparametric brick elements are used elsewhere in the modeling. Also, a cladding option is available that allows for an embedded or penetrating crack in the clad material. As few as four input cards are required to execute the program. ORMGEN-3D is part of a three-program system, ORMGEN-ADINA-ORVIRT, which addresses linear or nonlinear fracture in 2- or 3-D crack geometries. ORMGEN-3D creates data files for nodal point coordinates and element connectivities which have formats compatible with the ADINA structural analysis program. Program ORVIRT, described in Volume 2 of this report, functions as a post-processor to ADINA and employs a virtual crack extension technique to compute energy release rates at specified positions along the crack front.

NUREG/CR-2999: FINAL REPORT USNRC ANCHOR BOLT STUDY DATA SURVEY AND DYNAMIC TESTING. LINDQUIST, M. R. Hanford Engineering Development Laboratory. December 1982. 84pp. 8301190436. HEDL-MISC-7246. 16857:304.

A survey was performed to determine the adequacy of existing concrete expansion anchor test data. Based upon the survey findings, additional dynamic testing to assess the benefits of preload was undertaken. Exploratory testing was performed on typical wedge and shell anchors. It was found that, providing the installation torque is properly applied, residual preload does not significantly affect anchor load-displacement characteristics until the preload drops to less than

50% of the full installation preload. It was concluded that this must be considered in design situations where support stiffness is an important factor.

NUREG/CR-3001: FUEL PERFORMANCE ANNUAL REPORT FOR 1981. BAILEY, W. J. Battelle Memorial Institute, Pacific Northwest Laboratory. TOKAR, M. NRC - No Detailed Affiliation Given. December 1982. 82pp. 8301100026. PNL-4342. 16749:212.

This annual report, the fourth in a series, provides a brief description of fuel performance during 1981 in commercial nuclear power plants. Brief summaries of fuel operating experience, fuel problems, fuel design changes and fuel surveillance programs, and high burnup fuel experience are provided. References to additional, more detailed information and related NRC evaluations are included.

NUREG/CR-3004: EVALUATION OF EXPERIENCES IN LONG-TERM RADON AND RADON DAUGHTER MEASUREMENTS. YOUNG, J. A.; JACKSON, P. O.; THOMAS, V. W. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1982. 43pp. 8212220193. PNL-4509. 16527:019.

Pacific Northwest Laboratory (PNL) is performing side-by-side measurements of radon and radon daughter concentrations using several instruments and techniques, and is comparing these measurements with side-by-side measurements made by other investigators at other locations. The standard deviation of the differences between the (natural) logarithms of the Terradex Track Etch radon concentrations and the logarithms of the RPISU radon daughter concentrations (S.D.-1n) measured in 50 buildings in Edgemont, South Dakota, was 0.37. Using this S.D.-1n, it can be calculated that if the Track Etch radon daughter concentration is 0.010 WL there should be only a 14% probability that the RPISU average would be greater than 0.015 WL, and only a 3% probability that the RPISU average would be greater than 0.020 WL. If buildings had been cleared from remedial action when the Track Etch averages were less than 0.010 WL, then about 61% of the buildings would have been cleared from remedial action, and only a few percent of these buildings would have actually had average RPISU concentrations greater than 0.015 WL. The S.D.-1n between (1) the Track Etch radon measurements and the RPISU radon daughter measurements made by ALARA at Grand Junction, (2) the PERM radon measurements and the MOD-225 radon daughter measurements made by Mound Facility at Canonsburg and Middlesex, and (3) the PERM and Track Etch radon measurements made by Mound Facility at Salt Lake City were similar to the S.D.-1n between the Track Etch radon measurements and the RPISU radon daughter measurements at Edgemont.

NUREG/CR-3020: SELECTED REVIEW OF REGULATORY STANDARDS AND LICENSING ISSUES FOR NUCLEAR POWER PLANTS. STEVENSON, J. D.; THOMAS, F. A. Stevenson & Associates. November 1982. 488pp. 8212270221. 16552:001.

This report provides encyclopedic information on nuclear power regulatory standards for the United States, Canada, the United Kingdom, France, Sweden, the Federal Republic of Germany and Japan. Current licensing issues in these same countries are also treated briefly. A comparative evaluation of Standards in the different countries is presented. Technical disciplines covered include mechanical engineering, structural engineering, materials and metallurgy, seismology and geology.

NUREG/CR-3021: REGIONAL TECTONICS AND SEISMICITY OF SOUTHWESTERN IOWA. Final Report: 1978-1982. VAN ECK, O. J. ; ANDERSON, R. R. ; CUMERLATO, C. L. ; et al. Iowa Geological Survey. November 1982. 70pp. 8301100011. 16719:268.

Utilizing gravity and aeromagnetic data, a series of geophysical profiles were constructed across the Midcontinent Geophysical Anomaly (MGA) which extends across southwestern Iowa. By combining the information provided by modeling techniques with limited deep well data, a map of the Precambrian basement was generated. The interpretation that emerged includes a central horst of igneous material, extensively faulted, and overlain in some areas by Keweenawan clastics. The horst is flanked by a series of high angle faults, with the majority of the faults displaying vertical displacement along two structural zones, the Thurman-Redfield Structural Zone, southeast of the MGA, and the Northern Boundary Fault Zone to the northwest. The total vertical displacement present along these fault zones is about maximum of 9 Km. Two clastic-filled basins flank the horst, one of which has an interpreted depth of 10 Km. Seismic profiling indicates extensive folding and faulting of Paleozoic rock units, although the scale of deformation is minor when compared to Precambrian features in the same area. Although the Paleozoic Era was the most tectonically active of the post-Precambrian eras there are indications of post-Cretaceous movement. Recent tectonic activity is suggested by the numerous microearthquakes that were recorded by the monitoring array that was installed for this study.

NUREG/CR-3022: PARAMETERS THAT INFLUENCE DAMPING IN NUCLEAR POWER PLANT PIPING SYSTEMS. WARE, A. G. EG&G, Inc. December 1982. 43pp. 8301100033. EGG-2232. 16749:292.

This report gives the present status of the guidelines for structural damping in dynamic analyses of nuclear power plant piping systems. A brief description of the present state of knowledge of piping system damping in the U.S. is included, as are gaps in the overall understanding of the phenomenon. The report concludes with proposed EG&G Idaho efforts to contribute to the satisfactory establishment of reasonable damping values to be used in structural analyses.

NUREG/CR-3029: GRAVITATIONAL AGGLOMERATION OF POST-HCDA LMFBR AEROSOLS: NONSPHERICAL PARTICLES. TUTTLE, R. F. ; LOYALKA, S. K. Missouri, Univ. of, Columbia. December 1982. 248pp. 8301100039. 16753:084.

Aerosol behavior analysis computer programs have shown that temporal aerosol size distributions in nuclear reactor containments are sensitive to "shape" factors. This research investigates shape factors by a detailed theoretical analysis of hydrodynamic interactions between a nonspherical particle and a spherical particle undergoing gravitational collisions in an LMFBR environment. First, basic definitions and expressions for settling speeds and collisional efficiencies of nonspherical particles are developed. These are then related to corresponding quantities for spherical particles through shape factors. Using volume equivalent diameter as the defining length in the gravitational collison kernel, the aerodynamic shape factor, the density correction factor, and the gravitational collision shape factor, are introduced to describe the collision kernel for collisions between aerosol agglomerates. The Navier-Stokes equation in oblate spheroidal coordinates is solved to model a nonspherical particle and then the dynamic equations for two particle motions are developed. A computer program (NGCEFF) is constructed, and the dynamical equations are solved by Gear's method. Results are obtained for several cases and are compared with previous work of atmospheric sciences. Finally, results for the collisional shape factors, for a range of particle sizes and environments pertinent to LMFBR aerosols are provided.

NUREG/CR-3035: RADIOLOGICAL SURVEY OF THE REED-KEPPLER PARK SITE WEST CHICAGO, ILLINOIS. BOOTH, L. F.; MCDOWELL, G. S.; PECK, S. I.; et al. Radiation Management Corp. November 1982. 104pp. 8301100014. 16716:228.

This report presents the results of radiological survey of the Reed-Keppler Park, West Chicago, Illinois, performed by Radiation Management Corporation during the fall of 1981 and the spring of 1982. Measurements were made to determine external radiation levels, concentrations of water and airborne containements and the identity and concentrations of subsurface deposits. Results show that materials containing Th-232 and daughters are present in surface subsurface locations, comprising a total volume of about 15,000 cubic yards, with concentrations as high as 11,000 pCi/g. These contaminants are a source of radon and daughter radionuclides which may produce slightly elevated airborne radioactivity levels off-site. There is no evidence that materials are moving off-site through ground water, although small subsurface deposits exceeding 5 pCi/g exist north of the fenced site in a landfill area, and to the southeast of the site near the tennis courts. These off-site deposits do not present a significant radiological hazard to the public at this time.

NUREG/CR-3038: TESTS FOR EVALUATING SITES FOR DISPOSAL OF LOW-LEVEL RADIOACTIVE WASTE. LUTTON, R. J.; BUTLER, D. K.; MEADE, R. B.; et al. Army, Dept. of, Army Engineer Waterways Experiment Station. December 1982. 176pp. 8301190415. 16859:087.

This report, the second of a series, identifies the tests and other means of evaluating or documenting the important characteristics of sites for disposal of low-level radioactive waste. The specific parameters were identified and explained in regard to their importance in characterizing disposal facilities in the previous report. More than half of the tests and procedures are standard methods recognized and used nationwide, most conspicuously the numerous chemical tests. Other tests are commonly used methods recognized widely as state-of-the-art, e.g., geological and geophysical methods. The basis for choosing these state-of-the-art methods is discussed, and the concepts and procedures themselves are reviewed in the absence of standards for ready reference. Besides standards and state-of-the-art practices a third category of methods involves the use of existing data sources or recognized correlations in place of new testing or documentation. It is particularly important that mapping, logging, sampling, testing, interpretation, and analysis be conducted by technically qualified and professionally motivated personnel using appropriate equipment and facilities, and general guidance is provided in this direction.

NUREG/CR-3040: SELECTED REVIEW OF FOREIGN SAFETY RESEARCH FOR NUCLEAR POWER PLANTS. STEVENSON, J. D.; THOMAS, F. A. Stevenson & Associates. November 1982. 123pp. 8212060458. 16347:001.

A compilation and description of current foreign research related to regulatory standards and licensing issues in areas of interest associated with Siting, Structural Engineering, Metallurgy and Materials, and Mechanical Engineering are presented. Also included in this report and summary is a discussion of those research areas in which there exists a potential for joint sponsorship by the U.S. NRC. The particular foreign countries surveyed are Canada, France, Japan, Sweden, United Kingdom of Great Britain & Northern Ireland, and the Federal Republic of Germany.

NUREG/CR-3055: REVIEW AND EVALUATION OF PALEOHYDROLOGIC METHODOLOGIES. FOLEY, M. G. ; ZIMMERMAN, D. A. ; DOESBURG, J. M. ; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1982. 94pp. 8212220238. PNL-4346. 16526:058.

Pacific Northwest Laboratory conducted a literature review to identify methodologies that could be used to interpret paleohydrologic environments. Paleohydrology is the study of past hydrologic systems or of the past behavior of an existing hydrologic system. The purpose of the review was to evaluate how well these methodologies could be applied to the siting of low-level radioactive waste facilities.

Paleohydrologic interpretations are uncertain because of the effects of time hydrologic and geologic systems and because of the complexity of fluvial systems. Paleoflow determinations appear in many cases to be order-of-magnitude estimates. However, the methodologies identified in this report mitigate this uncertainty when used collectively as well as independently. That is, the data from individual methodologies can be compared or combined to corroborate hydrologic predictions. In this manner, paleohydrologic methodologies are viable tools to assist in evaluating the likely future hydrology of low-level radioactive waste sites.

NUREG/CR-3066: PARAMETERS AND VARIABLES APPEARING IN REPOSITORY SITING MODELS. MERCER, J. W. ; THOMAS, S. D. ; ROSS, B. Teknekron Research, Inc. December 1982. 262pp. 8301100029. 16718:049.

Included in this report is a summary of data characterizing the parameters and variables appearing in repository siting models. These data cover the processes of saturated flow, unsaturated flow, surface water flow, geochemistry, heat transport, solute transport, and geomechanical response. Definitions and ranges of values are provided for equation parameters, source terms, dependent variables, boundary conditions, and initial conditions for the equations that are solved in the repository siting models. The data were compiled to help guide the selection of values of parameters and variables to be used in benchmark problems.

NUREG/CR-3077: HAARM-3 CODE VERIFICATION PROCEDURE. GIESEKE, J. A. ; LEE, K. W. ; JORDAN, H. ; et al. Battelle Memorial Institute, Columbus Laboratories. November 1982. 61pp. 8212270420. BMI-2030. 16572:001.

A procedure is provided for verifying experimentally the HAARM-3 aerosol behavior code. Methodology is obtained for selecting experimental conditions under which important aerosol agglomeration and deposition mechanisms can be separately evaluated. It is additionally suggested that a statistical analysis be employed to assess the level of significance with which the HAARM-3 model can predict experimental data.

NUREG/CR-3090: EVALUATION OF WATER HAMMER POTENTIAL IN PREHEAT STEAM GENERATORS. SEXTON, D. E. ; KASAHARA, M. ; UFFER, R. A. Quadrex Corp. December 1982. 42pp. 8302030259. 17013:179.

An evaluation of the design and the operation of preheat steam generators was performed to determine the potential risk for damaging water hammer during auxiliary feedwater (AFW) operation. It was found that the potential for such a water hammer event is extremely low because multiple component failures and operational errors are required for the water hammer to occur.

NUREG/CR-3097: BENCHMARK PROBLEMS FOR REPOSITORY SITING MODELS.
ROSS, B.; MERCER, J. W.; THOMAS, S. D.; et al. Geotrans, Inc. December 1982. 151pp. 8302040479. 17044:191.

This report describes benchmark problems to test computer codes used in siting nuclear waste repositories. Analytical solutions, field problems and hypothetical problems are included. Problems are included for the following types of codes: ground-water flow in saturated porous media, ground-water flow in saturated fractured media, heat and solute transport in saturated porous media, solute transport in saturated porous media, solute transport in saturated fractured media, and solute transport in unsaturated porous media.

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.

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IMAGE EVALUATION TEST TARGET (MT-3)

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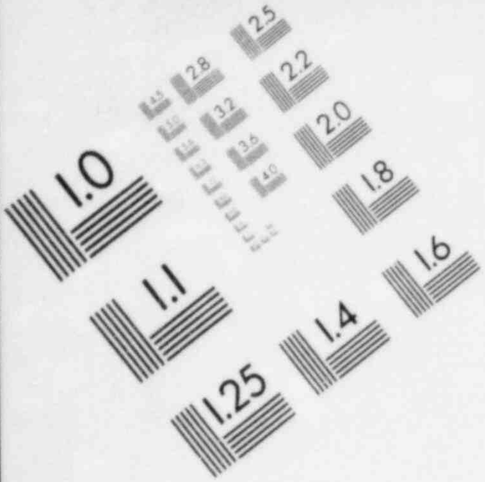

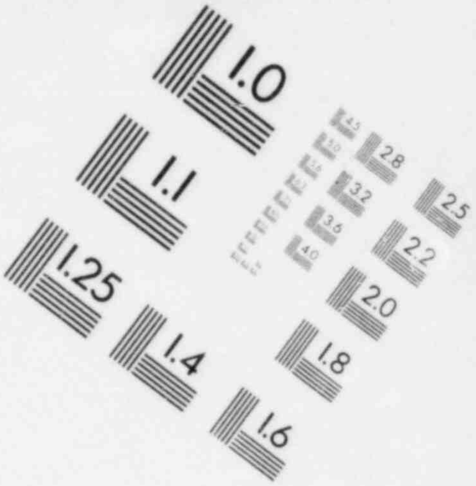

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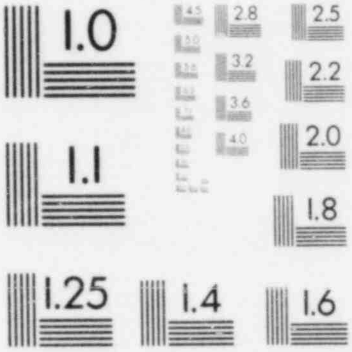

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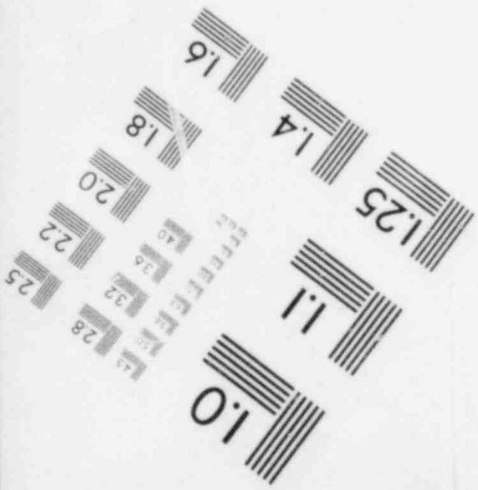

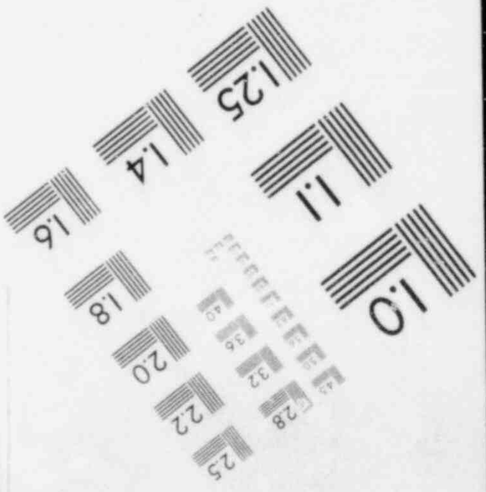

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NRC Originating Organization Index (Staff Reports)

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.

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