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Vol. 10, No. 4

Regulatory and Technical Reports (Abstract Index Journal)

Annual Compilation for 1985

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Division of Technical Information and Document Control
Office of Administration
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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
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Policy and Publications Management Branch
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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 staff of the Publishing and Translations Section of 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 this report.

NUREG-0017 R01: CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM PRESSURIZED WATER REACTORS (PWR-GALE CODE). CHANDRASEKARAN; LEE,J.Y.; WILLIS,C.A. Division of Systems Integration (811005-851124). April 1985. 208pp. 8505280361. 30603:161.

This report revises the original issuances of NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Pressurized Water Reactors (PWR-GALE Code)" (April 1976), to incorporate more recent operating data now available as well as the results of a number of in-plant measurement programs at operating pressurized water reactors. The PWR-GALE Code is a computerized mathematical model for calculating the releases of radioactive material in gaseous and liquid effluents (i.e., the gaseous and liquid source terms). The U.S. Nuclear Regulatory Commission uses the PWR-GALE Code to determine conformance with the requirements of Appendix I to 10 CFR Part 50.

NUREG-0020 V08 N12: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of November 30,1984.(Gray Book I) * Division of Budget & Analysis. February 1985. 427pp. 8502210264. 29047:289.

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 Resource Management 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 V09 N01: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of December 31,1984.(Gray Book I) * Division of Budget & Analysis. February 1985. 396pp. 8503220010. 29487:265.

See NUREG-0020,V08,N12 abstract.

NUREG-0020 V09 N02: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of January 31,1985.(Gray Book I) * Division of Budget & Analysis. March 1985. 421pp. 8504090008. 29740:311.

See NUREG-0020,V08,N12 abstract.

NUREG-0020 V09 N03: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of Febuary 28,1985.(Gray Book I) * Division of Budget & Analysis. April 1985. 403pp. 8505100053. 30269:082.

See NUREG-0020,V08,N12 abstract.

NUREG-0020 V09 N04: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of March 31,1985.(Gray Book I) ROSS,P.A.; BEEBE,M.R. Division of Budget & Analysis. May 1985. 440pp. 8506170349. 30959:072.

See NUREG-0020,V08,N12 abstract.

NUREG-0020 V09 N05: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of April 30, 1985.(Gray Book I) * Division of Budget & Analysis. June 1985. 441pp. 8507080197. 31396:267.

See NUREG-0020,V08,N12 abstract.

NUREG-0020 V09 N06: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of May 31,1985.(Gray Book I) ROSS,P.A.; BEEBE,M.R. Division of Budget & Analysis. July 1985. 437pp. 8508190629. 32261:001.

See NUREG-0020,V08,N12 abstract.

NUREG-0020 V09 N07: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of June 30,1985.(Gray Book I) ROSS,P.A.; BEEBE,M.R. Division of Budget & Analysis. August 1985. 426pp. 8509130022. 32620:001.

See NUREG-0020,V08,N12 abstract.

NUREG-0020 V09 N08: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of July 31,1985.(Gray Book I) * Division of Budget & Analysis. September 1985. 440pp. 8510070175. 32902:037.

See NUREG-0020,V08,N12 abstract.

NUREG-0020 V09 N09: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of August 31,1985.(Gray Book I) ROSS,P.A.; BEEBE,M.R. Division of Budget & Analysis. October 1985. 471pp. 8511210285. 33556:216.

See NUREG-0020,V08,N12 abstract.

NUREG-0020 V09 N10: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of September 30,1985.(Gray Book I) * Division of Budget & Analysis. November 1985. 455pp. 8512190244. 34015:345.

See NUREG-0020,V08,N12 abstract.

NUREG-0020 V09 N11: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of October 31,1985.(Gray Book I) ROSS,P.A.; BEEBE,M.R. Division of Budget & Analysis. December 1985. 200pp. 8601070520. 34184:313.

See NUREG-0020,V08,N12 abstract.

NUREG-0040 V08 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1984. (White Book) * Division of QA, Safeguards & Insp Programs (830103-850212). January 1985. 238pp. 8502210401. 29056:014.

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 October 1984 through December 1984. Also included in this issue are the results of certain inspections performed prior to October 1984 that were not included in previous issues of NUREG-0040.

NUREG-0040 V09 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1985.(White Book) * Division of QA, Vendor & Technical Training Center Programs (Post 850212). May 1985. 219pp. 8506030069. 30707:001.

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 1985 through March 1985. Also included in this issue are the results of certain inspections performed prior to January 1985 that were not included in previous issues of NUREG-0040.

NUREG-0040 V09 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,April-June 1985. (White Book) * Division of QA, Vendor & Technical Training Center Programs (Post 850212). August 1985. 245pp. 8509060260. 32503:078.

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 1985 through June 1985. Also included in this issue are the results of certain inspections performed prior to April 1985 that were not included in previous issues of NUREG-0040.

NUREG-0040 V09 N03: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,July 1985-September 1985.(White Book) * Division of QA, Vendor & Technical Training Center Programs (Post 850212). December 1985. 137pp. 8601070495. 34210:127.

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 1985 through September 1985. Also included in this issue are the results of certain inspections performed prior to July 1985 that were not included in previous issues of NUREG-0040.

NUREG-0090 V07 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1984. * AEOD, Director's Office. April 1985. 70pp. 8505160182. 30456:325.

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 and safety and requires a quarterly report of such events to be made to Congress. This report covers the period July 1 to September 30, 1984. During the report period, there were four abnormal occurrences at the nuclear power plants licensed to operate. These involved degraded isolation valves in emergency core cooling systems, degraded shutdown systems, a loss of offsite and onsite AC electrical power, and a refueling cavity water seal failure, respectively. There was one abnormal occurrence at a fuel cycle facility; the event involved degraded material access area barriers. There were four abnormal occurrences at the other NRC licensees. One involved contaminated radiopharmaceuticals used in several diagnostic administrations. Two involved therapeutic medical misadministrations. The other involved significant internal exposure to Iodine-125 to a hospital employee. There was one abnormal occurrence reported by an Agreement State; the event involved contaminated radiopharmaceuticals used in several diagnostic administrations. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0090 V07 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1984. * AEOD, Director's Office. May 1985. 40pp. 8506180402. 30986:013.

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 and safety and requires a quarterly report of such events to be made to Congress. This report covers the period October 1 to December 31, 1984. During the report period, there were two abnormal occurrences at the nuclear power plants licensed to operate. One involved four control rods failing to insert during testing and the other involved degraded upper head injection system accumulator isolation valves. There was one abnormal occurrence at a fuel cycle facility; the event involved buildup of uranium in a ventilation system. There was one abnormal occurrence reported by an Agreement State; the event involved an overexposure of a radiographer trainee. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0090 V08 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.Janurary-March 1985. * AEOD, Director's Office. August 1985. 46pp. 8509060189. 32505:273.

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 and safety and requires a quarterly report of such events to be made to Congress. This report covers the period January 1 to March 31, 1985. During the report period, there was one abnormal occurrence at the nuclear power plants licensed to operate; the event involved a premature criticality during reactor startup. There were three abnormal occurrences at the other NRC licensees. Two events involved diagnostic medical misadministrations and the other event involved unlawful possession of radioactive material. There were four abnormal occurrences reported by an Agreement State (Texas). Three events involved radiation overexposures; the other event involved a well logging source which was apparently stolen, but later was recovered. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0090 V08 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1985. * AEOD, Director's Office. November 1985. 54pp. 8512190002. 33966:012.

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 and safety and requires a quarterly report of such events to be made to Congress. This report covers the period April 1 to June 30, 1985. During the report period, there were three abnormal occurrences at the nuclear power plants licensed to operate. These events involved, respectively, (1) inoperable safety injection pumps, (2) significant deficiencies in reactor operator training and material false statements, and (3) loss of main and auxiliary feedwater system. There were four abnormal occurrences at other NRC licensees. Three events involved diagnostic or therapeutic medical misadministrations; the other involved a breakdown in management controls. There was one abnormal occurrence reported by an Agreement State; the event involved overexposures of a radiographer and an assistant radiographer. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0304 V09 N04: REGULATORY AND TECHNICAL REPORTS.Annual Compilation For 1984. * Division of Technical Information & Document Control. January 1985. 501pp. 8502210096. 29049:001.

This journal includes all formal reports in the NUREG series prepared by the NRC staff and contractors, as well as proceedings of conferences and workshops. The entries in the compilation are indexed for access by title and abstract, contractor report number, personal author, subject, NRC organization, contractor, and licensed facility.

NUREG-0304 V10 N01: REGULATORY AND TECHNICAL REPORTS.Compilation For First Quarter 1985,January-March. * Division of Technical Information & Document Control. April 1985. 129pp. 8505240207. 30564:195.

See NUREG-0304,V09,N04 abstract.

NUREG-0304 V10 N02: REGULATORY AND TECHNICAL REPORTS.Compilation For Second Quarter 1985,April-June. * Division of Technical Information & Document Control. July 1985. 88pp. 8508150008. 32198:129.
See NUREG-0304,V09,N04 abstract.

NUREG-0304 V10 N03: REGULATORY AND TECHNICAL REPORTS.Compilation For Third Quarter 1985,July - September. * Division of Technical Information & Document Control. October 1985. 72pp. 8511260053. 33639:193.
See NUREG-0304,V09,N04 abstract.

NUREG-0325 R07: U.S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS. * Office of Resource Management, Director. January 1985. 56pp. 8501180528. 28471:117.
Funtional organization charts for the NRC Commission Offices, Divisions, and Branches are presented.

NUREG-0383 V01 R08: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Packages. * Division of Fuel Cycle & Material Safety. October 1985. 477pp. 8511110217. 33408:043.
This directory contains a Summary Report of NRC Approved Packages (Volume 1), Certificates of Compliance (Volume 2), and a Summary Report of NRC Approved Quality Assurance Programs for Radioactive Material Packages (Volume 3). The purpose of this directory is to make available a convenient source of information on packagings which have been approved by the U.S. Nuclear Regulatory Commission. To assist in identifying packaging, an index by Model Number and corresponding Certificate of Compliance number is included at the back of each volume of the directory. The Summary Report includes a listing of all users of each package design prior to the publication date of the directory. Shipments of radioactive material utilizing these packagings must be in accordance with the provisions of 49 CFR 173.471 and 10 CFR Part 71, as applicable. In satisfying the requirements of Section 71.12, it is the responsibility of the licensees to insure them that they have a copy of the current approval and conduct their transportation activities in accordance with an NRC approved quality assurance program. Copies of the current approval may be obtained from the U.S. Nuclear Regulatory Commission Public Document Room files (see Docket No. listed on each certificate) at 1717 H Street, Washington, DC 20555. Note that the general license of 10 CFR 71.12 does not authorize the receipt, possession, use or transfer of byproduct source, or special nuclear material; such authorization must be obtained pursuant to 10 CFR Parts 30 to 36, 40, 50, or 70.

NUREG-0383 V02 R08: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Certificates of Compliance. * Division of Fuel Cycle & Material Safety. October 1985. 633pp. 8511110338. 33409:160.
See NUREG-0383,V01,R08 abstract.

NUREG-0383 V03 R05: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Quality Assurance Programs For Radioactive Material Packages. * Division of Fuel Cycle & Material Safety. October 1985. 128pp. 8511070490. 33383:243.
See NUREG-0383,V01,R08 abstract.

NUREG-0386 D03: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.JULY 1972 - SEPTEMBER 1983. * Office of the Executive Legal Director. * Aspen Systems, Inc. July 1985. 800pp. 8508210006. 32303:334.
This edition of the NRC Staff Practice and Procedure Digest contains a digest of a number of Commission, Atomic Safety and Licensing Appeal Board, and Atomic Safety and Licensing Board decisions issued during the period from July 1, 1972 to September 30, 1983 interpreting the NRC's Rules of Practice in 10 CFR Part 2. This edition replaces earlier editions and supplements and includes appropriate changes reflecting the amendment to the Rules of Practice effective September 30, 1983.

NUREG-0420 S09: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SHOREHAM NUCLEAR POWER STATION,UNIT 1.Docket No. 50-322. (Long Island Lighting Company) * Office of Nuclear Reactor Regulation, Director (post 851125). December 1985. 171pp. 8601070479. 34187:150.
Supplement 9 (SSER 9) to the Safety Evaluation Report on 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 addresses several items that have been reviewed by the staff since the previous supplement was issued.

NUREG-0430 V05 N01: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.January 1984 - June 1984.(Gray Book II) * Director's Office, Office of Inspection and Enforcement. April 1985. 18pp. 8504290008. 30056:348.
NRC is committed to the periodic publication of licensed facilities inventory difference data, following agency review of the information and completion of any related NRC investigations. Information 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-0430 V05 N02: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.July 1984 - December 1984.(Gray Book II) * Director's Office, Office of Inspection and Enforcement. October 1985. 17pp. 8511220456. 33601:030.
See NUREG-0430,V05,N01 abstract.

NUREG-0525 R10: SAFEGUARDS SUMMARY EVENT LIST (SSEL),REVISION 10. * Licensing Policy & Programs Branch (Pre 850707). May 1985. 59pp. 8506140072. 30907:130.
The Safeguards Summary Event List (SSEL) provides brief summaries of several hundred safeguards-related events involving nuclear material of facilities regulated by the U.S. Nuclear Regulatory Commission (NRC). Events are described under the categories of bomb-related, intrusion, missing/allegedly stolen, transportation, tampering/vandalism, arson, firearms-related, radiological sabotage and miscellaneous. The information contained in the event descriptions is derived primarily from official NRC reporting channels.

NUREG-0540 V06 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30,1984. * Division of Technical Information & Document Control. January 1985. 569pp. 8502060305. 28754:136.
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 V06 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.December 1-31,1984. * Division of Technical Information & Document Control. February 1985. 614PP. 8503200131. 29469:001.
See NUREG-0540,V06,N11 abstract.

NUREG-0540 V07 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.January 1-31,1985. * Division of Technical Information & Document Control. March 1985. 665pp. 8504090020. 29742:015.
See NUREG-0540,V06,N11 abstract.

NUREG-0540 V07 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.February 1-28,1985. * Division of Technical Information & Document Control. April 1985. 699pp. 8505160179. 30457:296.
See NUREG-0540,V06,N11 abstract.

NUREG-0540 V07 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.March 1-31,1985. * Division of Technical Information & Document Control. April 1985. 430pp. 8505210419. 30522:018.
See NUREG-0540,V06,N11 abstract.

NUREG-0540 V07 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30, 1985. * Division of Technical Information & Document Control. June 1985. 484pp. 8507080219. 31395:143.
See NUREG-0540,V06,N11 abstract.

NUREG-0540 V07 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.May 1-31, 1985. * Division of Technical Information & Document Control. July 1985. 489pp. 8507250203. 31790:011.
See NUREG-0540,V06,N11 abstract.

NUREG-0540 V07 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.June 1-30, 1985. * Division of Technical Information & Document Control. July 1985. 577pp. 8508150439. 32218:236.
See NUREG-0540,V06,N11 abstract.

NUREG-0540 V07 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.July 1-31, 1985. * Division of Technical Information & Document Control. August 1985. 638pp. 8509190103. 32670:122.
See NUREG-0540,V06,N11 abstract.

NUREG-0540 V07 N08: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.August 1-31, 1985. * Division of Technical Information & Document Control. September 1985. 656pp. 8510070182. 32903:117.
See NUREG-0540,V06,N11 abstract.

NUREG-0540 V07 N09: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.September 1-30,1985. * Division of Technical Information & Document Control. October 1985. 406pp. 8511220014. 33604:248.
See NUREG-0540,V06,N11 abstract.

NUREG-0540 V07 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.October 1-31,1985. * Division of Technical Information & Document Control. December 1985. 452pp. 8512270359. 34089:109.
See NUREG-0540,V06,N11 abstract.

NUREG-0544 R02: A HANDBOOK OF ACRONYMS AND INITIALISMS. * Division of Technical Information & Document Control. January 1985. 131pp. 8502150692. 28960:175.
This Handbook records in alphabetical order abbreviations (acronyms, initialisms, and other condensed forms) that have been used in the nuclear industry, both foreign and domestic. The present volume is an attempt by the editorial staff of the Division of Technical Information and Document Control to compile in one updated publication the abbreviations used in NRC staff and consultant reports. This issue, although not all inclusive, is to be treated as a compilation of available information.

NUREG-0606 V07 N01: UNRESOLVED SAFETY ISSUES SUMMARY.Data As Of February 15, 1985.(Aqua Book) * Division of Safety Technology (800428-851124). March 1985. 55pp. 8504040429. 29627:321.
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 V07 N02: UNRESOLVED SAFETY ISSUES SUMMARY.Data As Of May 17, 1985. (Aqua Book) * Division of Safety Technology (800428-851124). June 1985. 61pp. 8507080200. 31390:169.
See NUREG-0606,V07,N01 abstract.

NUREG-0606 V07 N03: UNRESOLVED SAFETY ISSUES SUMMARY.Data As Of August 16,1985. (Aqua Book) * Division of Safety Technology (800428-851124). August 1985. 55pp. 8509260507. 11631:363.
See NUREG-0606,V07,N01 abstract.

NUREG-0675 S28: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company) * Division of Licensing (800428-851124). April 1985. 635pp. 8505100069. 30286:156.
Supplement No. 28 to the Safety Evaluation Report for the application by the Pacific Gas and Electric Company (PG&E) to operate the Diablo Canyon Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-275 and 50-323) has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement reports on the status of the staff's investigation, inspection and evaluation of those allegations or concerns that have been identified to the NRC as of March 1, 1985.

NUREG-0675 S29: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company) * Division of Licensing (800428-851124). March 1985. 64pp. 8503280011. 29548:075.
Supplement No. 29 to the Safety Evaluation Report for Pacific Gas and Electric Company's application for licenses to operate Diablo Canyon Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-275 and 50-323), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement presents the staff evaluation of the licensee's Internal Review Program for Diablo Unit 2 applicability and resolution of concerns that had been raised during the Diablo Unit 1 design verification by the Independent Design Verification Program, the licensee's internal Technical Program and the NRC staff.

NUREG-0675 S30: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company) * Division of Licensing (800428-851124). April 1985. 137pp. 8504220336. 29943:202.
Supplement No. 30 to the Safety Evaluation Report for the application by the Pacific Gas and Electric Company (PG&E) to operate the Diablo Canyon Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-275 and 50-323) has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement reports on the staff's technical review and evaluation of the design and analysis of piping systems and pipe supports for Unit 2.

NUREG-0675 S31: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company) * Division of Licensing (800428-851124). April 1985. 171pp. 8505240223. 30564:022.
Supplement 31 to the Safety Evaluation Report for the application by Pacific Gas and Electric Company for licenses to operate Diablo Canyon Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-275/323) has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement addresses a number of matters related to the issuance of an operating license for Diablo

Canyon Unit 2, in particular those issues identified by the NRC staff in earlier supplements, commitments made by the applicant, and certain license conditions included in Facility Operating License No. DPR-81 for Diablo Canyon Unit 2.

NUREG-0675 S32: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos.50-275 And 50-323.(Pacific Gas and Electric Company) * Division of Licensing (800428-851124). August 1985. 50pp. 8508210398. 32337:324.

Supplement 32 to the Safety Evaluation Report for the application by Pacific Gas and Electric Company for licenses to operate Diablo Canyon Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-275/323) has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement provides the staff evaluation of those matters that require an appropriate resolution prior to full-power operation of Unit 2 and updates previous supplements to the Safety Evaluation Report.

NUREG-0713 V05: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS - 1983 ANNUAL REPORT. BROOKS,B.G. Division of Radiation Programs & Earth Sciences (post 840429). March 1985. 128pp. 8504050287. 29674:001.

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 1983. 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 license 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 76 nuclear power plants that had completed at least one full year of operation as of December 31, 1983, indicated that the number of personnel monitored during 1983 was 136,700 persons and the annual collective dose incurred by these individuals was 56,500 man-rems (man-cSv). The average annual dose for each worker that received a measurable dose was 0.66 rems(cSv), and the average collective dose per reactor was 753 man-rems (man-cSv). The termination reports revealed that some 56,500 individuals completed their employment with one or more reactor facilities during 1982.* Approximately 4,500 of these workers could be considered transients and they received an average dose of 1.11 rems(cSv). *The most recent year for which most of the termination data are available for analysis.

NUREG-0714 V04-05: OCCUPATIONAL RADIATION EXPOSURE.Fifteenth And Sixteenth Annual Reports,1982 And 1983. BROOKS,B.G.; MCDONALD,S.; RICHARDSON,E. Division of Radiation Programs & Earth Sciences (post 840429). October 1985. 53pp. 8511220012. 33603:356.

This report summarizes the occupational radiation exposure information that has been reported to the NRC by certain categories of NRC licensees during the years 1973 through 1983. 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. Data on workers terminating their employment at certain NRC licensed facilities were obtained from reports submitted pursuant to 10 CFR 20.408. The annual reports submitted by nearly 500 licensees indicated that approximately 154,000 individuals were monitored in 1982 and about 173,000 individuals were monitored in 1983. They incurred average annual doses of 0.37 rem (cSv) and 0.35 rem (cSv) respectively. Termination radiation exposure reports required to be submitted pursuant to 10 CFR 20.408 were analyzed to reveal that about 59,000 individuals completed their employment with one or more of the 500 covered licensees during 1982*. Some 56,000 of these individuals terminated from power reactor facilities, when about 4,500 of them were considered to be transient workers who received an average dose of 1.11 rems (cSv). * The most recent year for which most of the termination data are available for analysis.

NUREG-0725 R05: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL. * Division of Safeguards. June 1985. 55pp. 8507080177. 31394:330.

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 June 1, 1985. Spent fuel shipment routes, primarily for road transportation, but also including one rail route, are indicated on reproductions of DOT road maps. Also included are the amounts of material shipped during the approximate three year period that safeguards regulations for spent fuel shipments have been effective. In addition, the Commission has chosen to provide information in this document regarding the NRC's safety and safeguards regulations for spent fuel shipments as well as safeguards incidents regarding spent fuel shipments (of which none have been reported to date). This additional information is furnished by the Commission in order to convey to the public a more complete picture of NRC regulatory practices concerning the shipment of spent fuel than could be obtained by the publication of the shipment routes and quantities alone.

NUREG-0748 V04 N12: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of December 31,1984.(Orange Book) * Management Support Branch. January 1985. 320pp. 8502210263. 29056:355.

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 the operating power and nonpower reactors.

NUREG-0748 V05 N01: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of January 31,1985.(Orange Book) * Management Support Branch. March 1985. 332pp. 8504030070. 29598:326.

See NUREG-0748,V04,N12 abstract.

NUREG-0748 V05 N02: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of February 28,1985.(Orange Book) * Management Support Branch. April 1985. 335pp. 8505070582. 30207:295.

See NUREG-0748,V04,N12 abstract.

NUREG-0748 V05 N03: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of March 31,1985. (Orange Book) * Management Support Branch. May 1985. 342pp. 8506030192. 30688:001.

See NUREG-0748,V04,N12 abstract.

NUREG-0748 V05 N04: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of April 30,1985.(Orange Book) * Management Support Branch. June 1985. 344pp. 8507020440. 31310:077.

See NUREG-0748,V04,N12 abstract.

NUREG-0748 V05 N05: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of May 31,1985. (Orange Book) * Management Support Branch. July 1985. 357pp. 8507250167. 31792:028.

See NUREG-0748,V04,N12 abstract.

NUREG-0748 V05 N06: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of June 30,1985. (Orange Book) * Management Support Branch. August 1985. 358pp. 8508210039. 32302:336.

See NUREG-0748,V04,N12 abstract.

NUREG-0748 V05 N07: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of July 31,1985.(Orange Book) * Management Support Branch. September 1985. 362pp. 8510030438. 32846:287.
See NUREG-0748,V04,N12 abstract.

NUREG-0748 V05 N08: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of August 31,1985. (Orange Book) * Management Support Branch. October 1985. 365pp. 8511070084. 33377:161.
See NUREG-0748,V04,N12 abstract.

NUREG-0748 V05 N09: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of September 30,1985.(Orange Book) * Management Support Branch. November 1985. 385pp. 8511220465. 33601:046.
See NUREG-0748,V04,N12 abstract.

NUREG-0748 V05 N10: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of October 31,1985. (Orange Book) * Management Support Branch. December 1985. 225pp. 8512270422. 34073:081.
See NUREG-0748,V04,N12 abstract.

NUREG-0750 V20 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY-SEPTEMBER 1984. * Division of Technical Information & Document Control. January 1985. 78pp. 8503270298. 29540:261.
Digests and indexes for issuances of the Commission, the Atomic Safety and Licensing Appeal Panel, the Atomic Safety and Licensing Board Panel, the Administrative Law Judge, the Director's Decisions, and the Denials of Petitions for Rulemaking are presented.

NUREG-0750 V20 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY-DECEMBER 1984. * Division of Technical Information & Document Control. March 1985. 105pp. 8504040416. 29618:193.
See NUREG-0750,V20,I01 abstract.

NUREG-0750 V20 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1984. Pages 1,055-1,435. * Division of Technical Information & Document Control. January 1985. 388pp. 8502060353. 28748:001. Legal issuances of the Commission, the Atomic Safety and Licensing Appeal Panel, the Atomic Safety and Licensing Board Panel, the Administrative Law Judge, and NRC Program Offices.

NUREG-0750 V20 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1984. Pages 1,437-1,572. * Division of Technical Information & Document Control. January 1985. 143pp. 8502190320. 29012:226.
See NUREG-0750,V20,N04 abstract.

NUREG-0750 V20 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1984. Pages 1,573-1,706. * Division of Technical Information & Document Control. February 1985. 134pp. 8503110591. 29307:239.
See NUREG-0750,V20,N04 abstract.

NUREG-0750 V21 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY-MARCH 1985. * Division of Technical Information & Document Control. June 1985. 59pp. 8507080203. 31380:272.
See NUREG-0750,V20,I01 abstract.

NUREG-0750 V21 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY-JUNE 1985. * Division of Technical Information & Document Control. September 1985. 108pp. 8510030126. 32855:009.
See NUREG-0750,V20,I01 abstract.

NUREG-0750 V21 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1985. Pages 1-273. * Division of Technical Information & Document Control. March 1985. 281pp. 8504030299. 29624:081.
See NUREG-0750,V20,N04 abstract.

NUREG-0750 V21 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1985. Pages 275-469. * Division of Technical Information & Document Control. April 1985. 203pp. 8504290238. 30056:001.
See NUREG-0750,V20,N04 abstract.

NUREG-0750 V21 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1985. Pages 471-559. * Division of Technical Information & Document Control. May 1985. 78pp. 8505280104. 30606:279.
See NUREG-0750,V20,N04 abstract.

NUREG-0750 V21 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1985. Pages 561-1,041. * Division of Technical Information & Document Control. June 1985. 490pp. 8507080179. 31380:323.
See NUREG-0750,V20,N04 abstract.

NUREG-0750 V21 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1985. Pages 1,043-1,567. * Division of Technical Information & Document Control. July 1985. 524pp. 8508260300. 32370:304.
See NUREG-0750,V20,N04 abstract.

NUREG-0750 V21 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1985. Pages 1,569-1,786. * Division of Technical Information & Document Control. August 1985. 215pp. 8509230739. 32702:248.
See NUREG-0750,V20,N04 abstract.

NUREG-0750 V22 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-September 1985. * Commissioners. December 1985. 69pp. 8601090315. 34239:016.
See NUREG-0750,V20,I01 abstract.

NUREG-0750 V22 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1985. Pages 1-176. * Division of Technical Information & Document Control. August 1985. 184pp. 8509300527. 32787:308.
See NUREG-0750,V20,N04 abstract.

NUREG-0750 V22 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1985. Pages 177-457. * Division of Technical Information & Document Control. August 1985. 286pp. 8511010269. 33305:048.
See NUREG-0750,V20,N04 abstract.

NUREG-0750 V22 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1985. Pages 459-649. * Division of Technical Information & Document Control. November 1985. 201pp. 8512100721. 33830:086.
See NUREG-0750,V20,N04 abstract.

NUREG-0750 V22 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1985. Pages 651-769. * Division of Technical Information & Document Control. December 1985. 125pp. 8601070494. 34190:094.
See NUREG-0750,V20,N04 abstract.

NUREG-0787 S10: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATERFORD STEAM ELECTRIC STATION,UNIT 3.Docket No. 50-382. (Louisiana Power And Light Company) * Division of Licensing (800428-851124). March 1985. 35pp. 8503270297. 29541:279.
Supplement 10 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 U.S. 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 licensee since the Safety Evaluation Report and its nine supplements were issued.

NUREG-0797 S07: 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) * Division of Licensing (800428-851124). January 1985. 100pp. 8502150150. 28958:071.

Supplement No. 7 to the Safety Evaluation Report for the Texas Utilities Generating Company application for a license to operate the Comanche Peak Steam Electric Station located in Somervell County, Texas has been jointly prepared by the Office of Nuclear Reactor Regulation and the Technical Review Team of the U.S. Nuclear Regulatory Commission. This Supplement provides the results of the staff's evaluation and resolution of approximately 80 technical concerns and allegations in the areas of Electrical/Instrumentation and Test Program regarding construction practices at the Comanche Peak facility.

NUREG-0797 S08: 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) * Division of Licensing (800428-851124). February 1985. 196pp. 8503110247. 29327:151.

Supplement No. 8 to the Safety Evaluation Report for the Texas Utilities Generating Company application for a license to operate the Comanche Peak Steam Electric Station located in Somervell County, Texas has been jointly prepared by the Office of Nuclear Reactor Regulation and the Technical Review Team of the U.S. Nuclear Regulatory Commission. This Supplement provides the results of the staff's evaluation and resolution of approximately 80 technical concerns and allegations relating to civil/structural and miscellaneous issues regarding construction and plant readiness testing practices at the Comanche Peak facility.

NUREG-0797 S09: 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) * Division of Licensing (800428-851124). March 1985. 170pp. 8504090015. 29730:116.

Supplement 9 to the Safety Evaluation Report for the Texas Utilities Electric Company's application for a license to operate Comanche Peak Steam Electric Station, Units 1 and 2 located in Somervell County, Texas, has been prepared jointly by the Office of Reactor Regulation and the Comanche Peak Technical Review Team of the U.S. Nuclear Regulatory Commission. This supplement addresses Texas Utilities analyses in support of its request to amend the Comanche Peak Final Safety Analysis Report to eliminate the commitment that coatings inside the reactor Containment Building be qualified for Units 1 and 2. In addition, this supplement provides the results of the staff's evaluation and resolution of 62 technical concerns and allegations in the coating area for Unit 1. Because of the favorable resolution of the items discussed in this report, the staff concludes for the issues considered herein, that there is reasonable assurance that the facility can be operated without endangering the health and safety of the public.

NUREG-0797 S10: 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 Electric Company, et al) * Division of Licensing (800428-851124). April 1985. 326pp. 8506030061. 30706:008.

Supplement No. 10 to the Safety Evaluation Report for the Texas Utilities Electric Company application for a license to operate Comanche Peak Steam Electric Station, Units 1 and 2 (Docket Nos. 50-445 and 50-446), located in Somervell County, Texas, has been jointly prepared by the Office of Nuclear Reactor Regulation and the Comanche Peak Technical Review Team of the U.S. Nuclear Regulatory Commission. This supplement provides the results of the staff's evaluation and resolution of approximately 400 technical concerns and allegations in the mechanical and piping area regarding construction practices at the Comanche Peak facility. This report does not address the Walsh/Doyle allegations regarding deficiencies in the pipe support design process. Issues raised by the Walsh/Doyle allegations as well as issues raised during recent Atomic Safety and Licensing Board hearings will be dealt with in future supplements to the Safety Evaluation Report as needed.

NUREG-0797 S11: 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) * Division of Licensing (800428-851124). May 1985. 349pp. 8506190054. 31018:014.

Supplement No. 11 to the Safety Evaluation Report for the Texas Utilities Electric Company application for a license to operate the Comanche Peak Steam Electric Station, Units 1 and 2 (Docket Nos. 50-445 and 50-446), located in Somervell County, Texas, has been jointly prepared by the Office of Nuclear Reactor Regulation and the Comanche Peak Technical Review Team of the U.S. Nuclear Regulatory Commission (NRC) and is in two parts. Part 1 (Appendix O) of this supplement provides the results of the TRT's evaluation of approximately 125 concerns and allegations relating specifically to quality assurance and quality control (QA/QC) issues regarding construction practices at the Comanche Peak facility. Part 2 (Appendix P) contains overall summary and conclusion of the QA/QC aspects of the NRC Technical Review Team efforts as reported in Safety Evaluation Report (SER) Supplements 7, 8, 9 and 10. Issues raised during recent Atomic Safety and Licensing Board hearings will be dealt with in future supplements to the SER as needed.

NUREG-0797 S12: 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) * Comanche Peak Project (Technical Review Team). October 1985. 283pp. 8511040056. 33338:200.

Supplement No. 12 to the Safety Evaluation Report related to the operation of the Comanche Peak Steam Electric Station, Units 1 and 2 (NUREG-0797), 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, approximately 40 miles southwest of Fort Worth, Texas. This supplement reports the status of certain issues that had not been resolved at the time of publication of the Safety Evaluation Report and Supplements 1, 2, 3, 4 and 6 to that report. This supplement also lists the new issues that have been identified since Supplement 6 was issued and includes the evaluations for licensing items resolved in this interim period. Supplement 5 has not been issued. Supplements 7, 8, 9, 10 and 11 were limited to the staff evaluations of allegations investigated by the NRC Technical Review Team, and items identified therein are not included in this supplement.

NUREG-0798 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF ENRICO FERMI ATOMIC POWER PLANT,UNIT NO. 2.Docket No. 50-341. (Detroit Edison Company) * Division of Licensing (800428-851124). March 1985. 200pp. 8504040427. 29628:243.

Supplement No. 5 to the Safety Evaluation Report (SER) related to the operation of the Fermi-2 facility, provides the NRC staff's evaluation of additional information submitted by the applicant regarding the outstanding review issues identified in Supplement No. 4 to the SER dated September 1984. This supplement contains the staff's conclusion that there are no outstanding issues which must be resolved prior to issuance of a low-power operating license (i.e., less than five percent of full rated power) for the Fermi-2 facility. Supplement No. 5 to the SER also summarizes the conditions which are placed in the Fermi-2 operating license. The Fermi-2 facility is located on Lake Erie in

Monroe County, almost 8 miles east-northeast of Monroe, Michigan.

NUREG-0798 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF FERMI-2.Docket No. 50-341.(Detroit Edison Company) * Division of Licensing (800428-851124). July 1985. 56pp. 8508210034. 32302:259.

Supplement No. 6 to the Safety Evaluation Report (SER) related to operation of the Fermi-2 facility addresses items pertinent to the issuance of the full power license for Fermi-2. The Fermi-2 facility is located on Lake Erie in Monroe County, almost 8 miles east-northeast of Monroe, Michigan.

NUREG-0800 06.2.2 R4: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 4 To Section 6.2.2, "Containment Heat Removal System." * Office of Nuclear Reactor Regulation, Director (pre-851125). October 1985. 11pp. 8512110128. 33845:301.

Revision 4 to SRP Section 6.2.2 incorporates guidelines developed from the technical resolution of Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance."

NUREG-0800 13.5.2 R1: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 1 To Section 13.5.2, "Operating And Maintenance Procedures," and Revision 0 of Appendix A to Section 13.5.2, "Review...." * Office of Nuclear Reactor Regulation, Director (pre-851125). July 1985. 29pp. 8508150055. 32195:180.

Revision No. 1 to Section 13.5.2 and Revision 0 to Appendix A of Section 13.5.2 of the Standard Review Plan incorporates changes that have been developed since the original issuance in July 1981. This revision incorporates guidelines of Task Action Plan Items I.C.1 and I.C.9 of NUREG-0660 as clarified in Supplement 1 of NUREG-0737. Appendix A to SRP Section 13.5.2 was formerly NUREG-0899.

NUREG-0800 18.2 R0: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 0 To SRP Section 18.2, "Safety Parameter Display System (SPDS)." * Office of Nuclear Reactor Regulation, Director (pre-851125). January 1985. 11pp. 8502150068. 28962:310.

This revision incorporates the guideline of Task Action Plan Item 1.D.2 of NUREG-0660 as clarified in Supplement 1 to NUREG-0737, "Safety Parameter Display System."

NUREG-0800 18.2A1 R0: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 0 To Appendix A To SRP Section 18.2, "Human Factors Review Guidelines For The Safety Parameter Display System (SPDS)." * Office of Nuclear Reactor Regulation, Director (pre-851125). January 1985. 46pp. 8502140040. 28943:317.

This revision incorporates the guideline of Task Action Plan Item 1.D.2 of NUREG-0660 as clarified in Supplement 1 of NUREG-0737. Appendix A to SRP Section 18.2 was formerly draft NUREG-0835, "Human Factors Acceptance Criteria for the Safety Parameter Display System." Draft Report issued for Comment.

NUREG-0800 R05: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 5 To SRP Table Of Contents. * Office of Nuclear Reactor Regulation, Director (pre-851125). January 1985. 24pp. 8502140198. 28955:140.

Revision 5 to SRP Table of Contents.

NUREG-0824 S01: INTEGRATED PLANT SAFETY ASSESSMENT REPORT,SYSTEMATIC EVALUATION PROGRAM-MILLSTONE NUCLEAR POWER STATION,UNIT 1. Docket No. 50-245. (Northeast Nuclear Energy Company) * Office of Nuclear Reactor Regulation, Director (post 851125). November 1985. 58pp. 8512120114. 33874:202.

The Nuclear Regulatory Commission (NRC) has published its Supplement No. 1 to the Final Integrated Plant Safety Assessment Report (IPSAR) (NUREG-0824) under the scope of the Systematic Evaluation Program (SEP), for Northeast Nuclear Energy Company's Millstone 1 Plant located in Waterford, New London County, Connecticut. The SEP was initiated by the NRC to review the design of older operating nuclear power plants to reconfirm and document their safety. This report documents the review completed under the SEP for those issues that required refined engineering evaluations after the Final IPSAR for Millstone 1 Plant was issued. 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 Millstone 1 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 the supplement 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 Final IPSAR and its supplements will form part of the bases for considering the conversion of the license.

NUREG-0829 DRFT: INTEGRATED PLANT SAFETY ASSESSMENT REPORT,SYSTEMATIC EVALUATION PROGRAM - SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1.Docket No. 50-206.(Southern California Edison Company) * Division of Licensing (800428-851124). April 1985. 558pp. 8505240058. 30565:001.

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 confirm 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 San Onofre Nuclear Generating Station, Unit 1, operated by Southern California Edison Company. The San Onofre 1 facility is one of 10 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.

NUREG-0837 V04 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, July-September 1984. JANG,J.; KRAMARIC,M.; COHEN,L. Region 1, Office of Director. January 1985. 146pp. 8502040043. 28727:001.

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 third quarter of 1984.

NUREG-0837 V04 N04: NRC TLD DIRECT RADIATION MONITORING REPORT.Progress Report, October-December 1984. JANG,J.; KRAMARIC,M.; COHEN,L. Region 1, Office of Director. July 1985. 316pp. 8508010750. 31924:148.

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 1984.

NUREG-0837 V05 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, January-March 1985. JANG,J.; KRAMARIC,M.; COHEN,L. Region 1, Office of Director. July 1985. 152pp. 8508020373. 31961:073.

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 1985.

NUREG-0837 V05 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, April-June 1985. JANG,J.; KRAMARIC,M.; COHEN,L. Region 1, Office of Director. September 1985. 225pp. 8510020232. 32828:336.

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 quarter of 1985.

NUREG-0844 DRFT FC: NRC INTEGRATED PROGRAM FOR RESOLUTION OF UNRESOLVED SAFETY ISSUES A-3,A-4 AND A-5 REGARDING STEAM GENERATOR TUBE INTEGRITY.Draft Report For Comment. MURPHY,E. Division of Licensing (800428-851124). April 1985. 166pp. 8505310663. 30666:007.

This report presents the results of the NRC integrated program for the resolution of Unresolved Safety Issues A-3, A-4, and A-5 regarding steam generator tube integrity. The report addresses issues within the areas of steam generator integrity, plant systems response, human factors, radiological consequences and the response of various organizations to a steam generator tube rupture. A generic risk assessment is provided and indicates that risk from steam generator tube rupture events is not a significant contributor to total risk at a given site, nor to the total risk to which the general public is routinely exposed. However, the report identifies a number of actions which the staff finds as a group would be effective in significantly reducing the incidence of steam generator tube degradation, the frequency of tube ruptures and the corresponding potential for significant non-core melt radiological releases, and occupational radiological exposures and which would be effective in mitigating the consequences of SGTR events. The actions would also further reduce risk and have been designated as "staff recommended actions." Final publication of the report herein, following a 90-day period for public comment, will constitute technical resolution of Unresolved Safety Issues A-3, A-4, and A-5.

NUREG-0847 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT,UNITS 1 AND 2.Docket Nos. 50-390 And 50-391.(Tennessee Valley Authority) * Division of Licensing (800428-851124). January 1985. 77pp. 8502060553. 28745:199.

This report supplements the Safety Evaluation Report, NUREG-0847 (June 1982), Supplement No. 1 (September 1982), and Supplement No. 2 (January 1984) issued 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 confirmatory items and license conditions identified in the Safety Evaluation Report.

NUREG-0847 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT,UNITS 1 AND 2.Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority) * Division of Licensing (800428-851124). March 1985. 45pp. 8504050283. 29673:180.

This report supplements the Safety Evaluation Report, NUREG-0847 (June 1982), Supplement No. 1 (September 1982), Supplement No. 2 (January 1984), and Supplement No. 3 (January 1985) issued 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 and confirmatory items and license conditions identified in the Safety Evaluation Report.

NUREG-0853 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF CLINTON POWER STATION,UNIT 1.Docket No. 50-461.(Illinois Power Company,et al) * Division of Licensing (800428-851124). February 1985. 70pp. 8503210295. 29480:158.

Supplement No. 4 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 items that have been resolved by the staff since Supplement No. 3 was issued.

NUREG-0856 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment). SILBERBERG,M.; PASEDAG,W.F.; RYDER,C.P.; et al. Accident Source Term Program Office. July 1985. 265pp. 8508190634. 32262:085.

NUREG-0956 describes the NRC staff and contractor efforts to reassess and update the agency's analytical procedures for estimating accident source terms for nuclear power plants. The effort included development of a new source term analytical procedure -- a set of computer codes -- that is intended to replace the methodology of the Reactor Safety Study (WASH-1400) and to be used in reassessing the use of TID-14844 assumptions (10 CFR 100). NUREG-0956 describes the development of these codes, the demonstration of the codes to calculate source terms for specific cases, the peer review of this work, some perspectives on the overall impact of new source terms on plant risks, the plans for related research projects, and the conclusions and recommendations resulting from the effort.

NUREG-0857 S08: 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) * Division of Licensing (800428-851124). May 1985. 37pp. 8506240081. 31177:312.

Supplement No. 8 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 U.S. Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation Report by providing an evaluation of (1) additional information submitted by the applicants since Supplement No. 7 was issued and (2) matters that the staff had under review when Supplement No. 7 was issued, specifically those issues which required resolution prior to plant operation of Unit 1 above 5% full power.

NUREG-0857 S09: 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) * Division of Pressurized Water Reactor Licensing - B (post 851125). December 1985. 63pp. 8601070472. 34187:319.

Supplement No. 9 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 Regulations of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation Report by providing an evaluation of (1) additional information submitted by the applicants since Supplement No. 8 was issued and (2) matters that the staff had under review when Supplement No. 8 was issued, specifically those issues which required resolution prior to Unit 2 fuel loading and testing up to 5% of full power.

NUREG-0869 R01: USI A-43 REGULATORY ANALYSIS. SERKIZ,A.W. Division of Safety Technology (800428-851124). October 1985. 130pp. 8512100503. 33833:001.

This report consists of (1) the regulatory analysis for Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance"; (2) the proposed resolution; (3) a summary of public comments received and action taken; (4) the Committee to Review Generic Requirements (CRGR) minutes related to this USI; and (5) appendices that summarize assumptions, calculational methods, consequence analyses, and cost estimates used in this regulatory analysis.

NUREG-0871 V04 N01: SUMMARY INFORMATION REPORT.Data As Of June 30,1985. (Brown Book) * Division of Budget & Analysis. October 1985. 51pp. 8511010465. 33333:094.

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-0876 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BYRON STATION,UNITS 1 AND 2.Docket Nos. 50-454 And 50-455. (Commonwealth Edison Company) * Division of Licensing (800428-851124). February 1985. 52pp. 8503050075. 29243:290.

Supplement No. 6 to the Safety Evaluation related to 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 provides recent information regarding resolution of the license conditions identified in the SER. Because of the favorable resolution of the items discussed in this report, the staff concludes that the Byron Station, Unit 1 can be operated by the licensee at power levels greater than 5% without endangering the health and safety of the public.

NUREG-0881 S05: 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) * Division of Licensing (800428-851124). March 1985. 240pp. 8504030055. 29599:298.

Supplement No. 5 to the Safety Evaluation Report related to the operation of the Wolf Creek Generating Station, Unit No. 1 updates the information contained in the Safety Evaluation Report, dated April 1982 and Supplements 1, 2, 3 and 4, dated August 1982, June 1983, August 1983, and December 1983, respectively. Supplement No. 5 also addresses open issues and items concerning the issuance of a five percent low power license. The Safety Evaluation and its supplements pertains to the application for a license to operate the Wolf Creek Generating Station, Unit No. 1 filed by Kansas Gas and Electric Company on February 18, 1980. The Construction Permit No. CPPR-147 was issued on May 17, 1977. The facility is located in Coffey County, Kansas.

NUREG-0881 S06: 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) * Division of Licensing (800428-851124). June 1985. 22pp. 8506240149. 31152:311.

Supplement No. 6 to the Safety Evaluation Report related to the operation of the Wolf Creek Generating Station, Unit No. 1 updates the information contained in the Safety Evaluation Report, dated April 1982 and Supplements 1, 2, 3, 4, and 5, dated August 1982, June 1983, August 1983, December 1983, and March 1985 respectively. Supplement No. 6 concludes that the facility can be operated by the licensee at power levels greater than 5% without endangering the health and safety of the public. The Safety Evaluation and its supplements pertains to the application for a license to operate the Wolf Creek Generating Station, Unit No. 1 filed by Kansas Gas and Electric Company on February 18, 1980. The Construction Permit No. CPPR-147 was issued on May 17, 1977 and a low power 5% license issued on March 11, 1985. The facility is located in Coffey County, Kansas.

NUREG-0885 I04: U.S. NUCLEAR REGULATORY COMMISSION POLICY AND PLANNING GUIDANCE 1985. * Commissioners. February 1985. 25pp. 8503220005. 29495:096.

The purposes of the Policy and Planning Guidance document are to set forth the regulatory approach of the Nuclear Regulatory Commission and to provide the supporting principles to that approach; to state the major policies and planning objectives of the Commission; and to provide a common basis for the development of programs, for the establishment of priorities, and for the allocation of resources.

NUREG-0887 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF THE PERRY NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-440 And 50-441.(Cleveland Electric Illuminating Company) * Division of Licensing (800428-851124). February 1985. 212pp. 8503050551. 29220:135.

Supplement No. 5 to the Safety Evaluation Report (NUREG-0887) pertains to the application filed by the Cleveland Electric Illuminating Company, the Ohio Edison Company, the Pennsylvania Power Company, and the Toledo Edison Company (the Central Area Power Coordination Group or 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 report has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Lake County, Ohio, approximately 35 miles northeast of Cleveland, Ohio. This supplement reports the status of certain issues that had not been resolved at the time of publication of the Safety Evaluation Report and Supplement Nos. 1 through 4.

NUREG-0887 S06: 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) * Division of Licensing (800428-851124). April 1985. 150pp. 8505010117. 30114:204.

Safety Evaluation Report, NUREG-0887, pertains to 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 or 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 report has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Lake County, Ohio, approximately 35 miles northeast of Cleveland, Ohio. This Supplement, No. 6 addresses the remaining unresolved Atomic Safety and Licensing Board contention issues; TDI diesel generator reliability in Section 9.6.3.1; hydrogen control system design per the new hydrogen rule in Section 6.2.7; and several issues related to Emergency Plans in Section 13.3.

NUREG-0887 S07: 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 (post 851125). November 1985. 243pp. 8512120154. 33875:280.

This Supplement No. 7 reports the status of certain issues that had not been resolved at the time of publication of the Safety Evaluation and Supplement Nos. 1 through 6 to that report. The Perry Nuclear Power Plant facility is located in Lake County, Ohio, approximately 35 miles northeast of Cleveland, Ohio. This report relates to the application for licenses to operate the Perry Nuclear Power Plant, Units 1 and 2 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 as applicants and owners.

NUREG-0896 S03: SAFETY EVALUATION REPORT 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) * Division of Licensing (800428-851124). July 1985. 94pp. 8508090565. 32112:094.

Supplement No. 3 to the Safety Evaluation Report for the application filed by Public Service Company of New Hampshire, et al. for licenses to operate the Seabrook Station, Units 1 and 2, located in Rockingham County, New Hampshire, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulation Commission. This supplement provides information to update the status of the NRC review of the application.

NUREG-0897 R01: CONTAINMENT EMERGENCY SUMP PERFORMANCE.(Technical Findings Related To Unresolved Safety Issues). SERKIZ,A.W. Division of Safety Technology (800428-851124). October 1985. 240pp. 8512110209. 33847:235.

This report summarizes key technical findings related to Unresolved Safey Issue (USI) A-43, Containment Emergency Sump Performance. Although this issue was formulated considering pressurized water reactor (PWR) sumps, the generic safety questions apply to both boiling water reactors (BWRs) and PWRs. Hence, both BWRs and PWRs are considered in this report. The technical findings in this report provide information on post-LOCA recirculation performance. These findings have been derived from extensive experimental studies, generic plant studies, and assessments of sumps and pumps used for long-term cooling. The results of hydraulic tests have shown that the potential for air ingestion is less severe than previously hypothesized. The effects of debris blockage on NPSH margin must be dealt with on a plant-specific basis. These findings have been used to develop revisions to Regulatory Guide 1.82 and Standard Review Plan Section 6.2.2. (NUREG-0800).

NUREG-0905: CLOSEOUT OF IE BULLETIN 79-12:SHORT-PERIOD SCRAMS AT BOILING-WATER REACTORS. DEBEVEC,C.J.; HOLLAND,R.A. Emergency Preparedness Branch. March 1985. 21pp. 8504050280. IEB-79-12. 29673:144.

IE Circular 77-07 was issued on April 14, 1977 because of the occurrence of short period scram events at Dresden Unit 2 on December 28, 1976 and at Monticello on February 23, 1977. The circular advised BWR plants to revise their control rod withdrawal sequences and operating procedures to reduce the likelihood of future short period scrams. However, similar events continued to occur. These included events at Oyster Creek on December 14, 1978; at Browns Ferry Unit 1 on January 18, 1979; and at Hatch Unit 1 on January 31, 1979. As a result of these events, IE Bulletin 79-12 was issued on May 31, 1979. This bulletin required a written response from licensees of GE-designed BWRs regarding specific actions listed in the bulletin. All of the licensees responded in a satisfactory manner. No similar events have been reported since IE Bulletin 79-12 was issued.

NUREG-0910 R01 S01: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE. * Division of Technical Information & Document Control. January 1985. 18pp. 8502060493. 28749:329.

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-0910 R01 S02: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE. * Division of Technical Information & Document Control. February 1985. 24pp. 8503010298. 29188:287.

See NUREG-0910,R01,S01 abstract.

NUREG-0910 R01 S03: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE. * Division of Technical Information & Document Control. April 1985. 22pp. 8505100062. 30286:119.

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 has published and maintains "NRC Comprehensive Records Disposition Schedule," (NUREG-0910) for records created or maintained by the NRC. Supplement 3 forwards changes to the General Records Schedules as made by the National Archives & Records Administration (NARA) and General Schedule 20 for inclusion.

NUREG-0910 R01 S04: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE. * Division of Technical Information & Document Control. October 1985. 9pp. 8511040062. 33337:091.

In compliance with statutory requirements set forth in Title 44 U.S. Code, "Public Printing and Documents," and in applicable regulation cited in 36 CFR Subchapter XII, National Archives and Records Administration (NARA). The U.S. Nuclear Regulatory Commission has published and maintains "NRC Comprehensive Records Disposition Schedule," (NUREG-0910 Rev. 1) for records created or maintained by the NRC. Supplement 4 forwards schedules to the General Records Schedules and approved NRC schedules as made by the National Archives and Records Administration for inclusion.

NUREG-0933 S02: A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R.; MINNERS,W.; VANDER MOLEN,H.; et al. Division of Safety Technology (800428-851124). January 1985. 288pp. 8502210083. 29053:003.

The report presents the priority rankings for generic safety issues related to nuclear power plants. The purpose of these rankings is to assist in the timely and efficient allocation of NRC resources for the resolution of those safety issues that have a significant potential for reducing risk. The safety priority rankings are HIGH, MEDIUM, LOW, and DROP and have been assigned on the basis of risk significance estimates, the ratio of risk to costs and other impacts estimated to result if resolutions of the safety issues were implemented, and the consideration of uncertainties and other quantitative or qualitative factors. To the extent practical, estimates are quantitative.

NUREG-0933 S03: A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R.; MINNERS,W.; VANDERMOLEN,H.; et al. Division of Safety Technology (800428-851124). July 1985. 269pp. 8508150024. 32194:252.

See NUREG-0933,S02 abstract.

NUREG-0936 V03 N04: NRC REGULATORY AGENDA.Quarterly Report,October-December 1984. * Division of Rules and Records. February 1985. 207pp. 8502250788. 29094:043.

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 V04 N01: NRC REGULATORY AGENDA.Quarterly Report,January-March 1985. * Division of Rules and Records. May 1985. 201pp. 8505310669. 30666:173.

See NUREG-0936,V03,N04 abstract.

NUREG-0936 V04 N02: NRC REGULATORY AGENDA.Quarterly Report,April-June 1985. * Division of Rules and Records. July 1985. 216pp. 8508090725. 32101:305.

See NUREG-0936,V03,N04 abstract.

NUREG-0936 V04 N03: NRC REGULATORY AGENDA.Quarterly Report,July-September 1985. * Division of Rules and Records. October 1985. 210pp. 8511070469. 33384:292.

See NUREG-0936,V03,N04 abstract.

NUREG-0940 V03 N04: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,October-December 1984. * Enforcement Staff. February 1985. 402pp. 8502250240. 29093:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (October - December 1984) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions and the licensees' responses. 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 pubic health and safety as well as common defense and security.

NUREG-0940 V04 N01: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,January-March 1985. * Enforcement Staff. April 1985. 541pp. 8505100072. 30288:071.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January - March 1985) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions and the licensees' responses. 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.

NUREG-0940 V04 N02: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June,1985. * Enforcement Staff. July 1985. 341pp. 8508190616. 32260:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1985) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions and the licensees' responses. 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.

NUREG-0940 V04 N03: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,July-September 1985. * Enforcement Staff. November 1985. 243pp. 8512120125. 33875:023.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (July - September 1985) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions and the licensees' responses. 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.

NUREG-0946: AN EVALUATION OF RADIONUCLIDE CONCENTRATIONS IN HIGH-LEVEL RADIOACTIVE WASTES. FEHRINGER,D.J. Division of Waste Management. October 1985. 34pp. 8510310448. 33278:231.

This report describes a possible approach for development of a numerical definition of the term "high-level radioactive waste." Five wastes are identified which are recognized as being high-level wastes under current, non-numerical definitions. The constituents of these wastes are examined and the most hazardous component radionuclides are identified. This report suggests that other wastes with similar concentrations of these radionuclides could also be defined as high-level wastes.

NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment). SILBERBERG,M.; MITCHELL,J.A.; MEYER,R.O.; et al. Accident Source Term Program Office. July 1985. 265pp. 8508190634. 32262:085.

NUREG-0956 describes the NRC staff and contractor efforts to reassess and update the agency's analytical procedures for estimating accident source terms for nuclear power plants. The effort included development of a new source term analytical procedure -- a set of computer codes -- that is intended to replace the methodology of the Reactor Safety Study (WASH-1400) and to be used in reassessing the use of TID-14844 assumptions (10 CFR 100). NUREG-0956 describes the development of these codes, the demonstration of the codes to calculate source terms for specific cases, the peer review of this work, some perspectives on the overall impact of new source terms on plant risks, the plans for related research projects, and the conclusions and recommendations resulting from the effort.

NUREG-0970: PROCEDURES FOR MEETING NRC ANTITRUST RESPONSIBILITIES. TOALSTON,A.L.; MESSIER,M.E.; LAMBE,W.M.; et al. Site Analysis Branch. May 1985. 29pp. 8506070360. 30798:013.

This report describes the procedures used by NRC staff to implement the antitrust review and enforcement prescribed in Sections 105 and 186 of the Atomic Energy Act of 1954, as amended (the Act), as covered largely by the Commission's Rules and Regulations in 10 CFR Parts 2.101, 2.102, 2.200, 50.33a, 50.80, and 50.90. These procedures set forth the steps and criteria the staff applies in the antitrust review of construction permit and operating license applications and the amendments to those applications that deal with changes in ownership. In addition, the procedures describe how the staff enforces compliance by licensees when antitrust conditions have been appended to construction permits and operating licenses.

NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984. * Division of Engineering Technology. April 1985. 400pp. 8506040260. 30711:262.

This report presents summaries of the research work performed during Fiscal Year 1984 by laboratories and organizations under contracts administered by the NRC's Materials Engineering Branch, Office of Nuclear Regulatory Research. Each contractor has written a more complete and detailed annual report of their work which can be obtained by writing to NRC; however, we believe it is useful to have a summary of each contractor's efforts for the year combined into one volume.

NUREG-0979 S03: SAFETY EVALUATION REPORT RELATED TO THE FINAL DESIGN APPROVAL OF THE GESSAR II BWR/6 NUCLEAR ISLAND DESIGN. Docket No. 50-447. (General Electric Company) * Division of Licensing (800428-851124). January 1985. 33pp. 8502110615. 28860:291.

Supplement 3 to the Safety Evaluation Report (SER) for the application filed by General Electric Company for the final design approval for the GE BWR/6 nuclear island design has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. This report supplements the GESSAR II SER (NUREG-0979), issued in April 1983, summarizing the results of the staff's safety review of the GESSAR II BWR/6 nuclear island design. Subject to favorable resolution of the items discussed in this supplement, the staff concludes that the GESSAR II design satisfactorily addresses the severe-accident concerns described in draft NUREG-1070.

NUREG-0979 S04: SAFETY EVALUATION REPORT RELATED TO FINAL DESIGN APPROVAL OF THE GESSAR II BWR/6 NUCLEAR ISLAND DESIGN.Docket No. 50-447.(General Electric Company) * Division of Licensing (800428-851124). July 1985. 91pp. 8508020356. 31961:226.

Supplement 4 to the Safety Evaluation Report (SER) for the application filed by General Electric Company for the final design approval for the GE BWR/6 nuclear island design (GESSAR II) has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. This report supplements the GESSAR II SER (NUREG-0979) issued in April 1983 summarizing the results of the staff's safety review of the GESSAR II BWR/6 nuclear island design; Supplement 1, issued in July 1983; Supplement 2, issued in November 1984; and Supplement 3, issued in January 1985. Subject to favorable resolution of the items discussed in this supplement, the staff concludes that the GESSAR II design satisfactorily addresses the severe-accident concerns described in the Commission's Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants.

NUREG-0981 R01: NRC/FEMA OPERATIONAL RESPONSE PROCEDURES FOR RESPONSE TO A COMMERCIAL NUCLEAR REACTOR ACCIDENT. * Director's Office, Office of Inspection and Enforcement. * Federal Emergency Management Agency. February 1985. 34pp. 8503060141. FEMA-51. 29263:287.

Procedures have been developed by the U.S. Nuclear Regulatory Commission (NRC) and the Federal Emergency Management Agency (FEMA) which provide the response teams of both agencies with the steps to be taken in responding to an emergency at a commercial nuclear power plant. The emphasis of these procedures is mainly on the interface between NRC and FEMA at their respective Headquarters and Regional Offices and at the various sites at which such an emergency could occur. Detailed procedures are presented that cover for both agencies, notification schemes and manner of activation, organizations at Headquarters and the site, interface procedures, coordination of onsite and offsite operations, the role of the Senior FEMA official, and the cooperative efforts of each agency's public information staff.

NUREG-0989 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF RIVER BEND STATION.Docket No. 50-458.(Gulf States Utilities Company,Cajun Electric Power Cooperative) * Division of Licensing (800428-851124). August 1985. 247pp. 8508210406. 32338:015.

Supplement No. 2 to the Safety Evaluation Report for the application filed by Gulf States Utilities Company as applicant and for itself and Cajun Electric Power Cooperative, as owners, for a license to operate River Bend Station has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in West Feliciana Parish, near St. Francisville, Louisiana. 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-0989 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF RIVER BEND STATION.Docket No. 50-458.(Gulf States Utilities Company,Cajun Electric Power Cooperative) * Division of Licensing (800428-851124). August 1985. 305pp. 8509100337. 32529:148.

Supplement No. 3 to the Safety Evaluation Report for the application filed by Gulf States Utilities Company as applicant and for itself and Cajun Electric Power Cooperative, as owners, for a license to operate River Bend Station has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in West Feliciana Parish, near St Francisville, Louisiana. The supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report and Supplement Nos. 1 and 2.

NUREG-0989 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF RIVER BEND STATION.Docket No. 50-458.(Gulf States Utilities Company,Cajun Electric Power Cooperative) * Division of Licensing (800428-851124). September 1985. 39pp. 8509260506. 32758:043.

Supplement No. 4 to the Safety Evaluation Report for the application filed by Gulf States Utilities Company as applicant and for itself and Cajun Electric Power Cooperative, as owners, for a license to operate River Bend Station has been prepared by the Office of the Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in West Feliciana Parish, near St. Francisville, Louisiana.

NUREG-0989 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF RIVER BEND STATION.Docket No. 50-458.(Gulf States Utilities Company,Cajun Electric Power Cooperative) * Office of Nuclear Reactor Regulation, Director (post 851125). November 1985. 58pp. 8512100373. 33833:224.

Supplement No. 5 to the Safety Evaluation Report for the application filed by Gulf States Utilities Company as applicant and for itself and Cajun Electric Power Cooperative, as owners for a license to operate River Bend Station has been prepared by the Office of the Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in West Feliciana Parish, near St. Francisville, Louisiana.

NUREG-0991 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF LIMERICK GENERATING STATION,UNITS 1 AND 2.Docket Nos. 50-352 And 50-353.(Philadelphia Electric Company) * Division of Licensing (800428-851124). May 1985. 51pp. 8506060596. 30776:256.

In August 1983 the NRC Staff issued its Safety Evaluation Report regarding the application for licenses to operate the Limerick Generating Station, Units 1 & 2 located on a site in Montgomery and Chester Counties, Pennsylvania. Supplement 1 was issued in December 1983 and addressed several outstanding issues. It also contains the comments made by the Advisory Committee on Reactor Safeguards in its interim report dated October 18, 1983. Supplement 2 was issued in October 1984. Supplement 3 was issued in October 1984 and addressed issues that required resolution before issuance of the operating license for Unit 1. On October 26, 1984 a license (NPF-27) for Unit 1 was issued which was restricted to a five percent power level and contained conditions which required resolution prior to proceeding beyond the five percent power level. This Supplement 4 addresses some of those technical issues and their associated license conditions which require resolution prior to proceeding beyond the five percent power level. The remaining issues will be addressed in a later supplement to this report. This Supplement 4 also contains the comments made by the Advisory Committee on Reactor Safeguards in its report dated

November 6, 1984, regarding full power operation of Limerick Unit 1.

NUREG-0991 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF LIMERICK GENERATING STATION,UNITS 1 AND 2.Docket Nos. 50-352 And 50-353. (Philadelphia Electric Company) * Division of Licensing (800428-851124). July 1985. 52pp. 8508020362. 31962:001.

In August 1983 the NRC issued its Safety Evaluation Report regarding the application for licenses to operate the Limerick Generating Station, Units 1 and 2 located on a site in Montgomery and Chester Counties, Pennsylvania. Supplement 1 was issued in December 1983 and addressed several outstanding issues. SSER 1 also contains the comments made by the Advisory Committee on Reactor Safeguards in its interim report dated October 18, 1983. Supplement 2 was issued in October 1984. Supplement 3 was issued in October 1984 and addressed the remaining issues that required resolution before issuance of the operating license for Unit 1. On October 26, 1984 a license (NPF-27) for Unit 1 was restricted to a five percent power level and contained conditions which required resolution prior to proceeding beyond the five percent power level. Supplement 4 issued in May 1985 addressed some of the technical issues and their associated license conditions, which required resolution prior to proceeding beyond the five percent power level. SSER 4 also contained the comments made by the Advisory Committee on Reactor Safeguards in its report dated November 6, 1984. This Supplement 5 to the SER addresses further issues that require resolution prior to proceeding beyond the five percent level.

NUREG-0991 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF LIMERICK GENERATING STATION,UNITS 1 AND 2.Docket Nos. 50-352 And 50-353.(Philadelphia Electric Company) * Division of Licensing (800428-851124). August 1985. 19pp. 8508210037. 32302:317.

In August 1983 the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-0991) regarding the application of the Philadelphia Electric Company (the licensee) for licenses to operate the Limerick Generating Station, Units 1 and 2, located on a site in Montgomery and Chester Counties, Pennsylvania. Supplement 1 was issued in December 1983. Supplement 2 was issued in October 1984. Supplement 3 was issued October 1984. Supplement 4 was issued in May 1985. Supplement 5 was issued in July 1985 and Supplement 6 issued in August 1985. This supplement 6 addresses further issues, principally the status of offsite emergency planning, that require resolution prior to proceeding beyond the five percent power level.

NUREG-1021 R01: OPERATOR LICENSING EXAMINER STANDARDS. * Division of Human Factors Safety (800428-851124). February 1985. 195pp. 8503200229. 29473:058.

The Operator Licensing Examiner Standards provide policy and guidance to NRC examiners and establish the procedures and practices for examining and licensing of Title 10 of the CODE OF FEDERAL REGULATIONS (10 CFR 55). They are intended to assist NRC examiners and facility licensees to understand the examination process better and to provide for equitable and consistent administration of examinations to all applicants by NRC examiners. These standards are not a substitute for the operator licensing regulations and are subject to revision or other internal operator examination licensing policy changes. As appropriate, these standards will be revised periodically to accommodate comments and reflect new information or experience.

NUREG-1022 S02: LICENSEE EVENT REPORT SYSTEM.Evaluation Of First Year Results And Recommendations For Improvements. HEBDON,F.J. AEOD, Director's Office. September 1985. 84pp. 8509230665. 32701:299.

This report describes an evaluation of an industry-wide sample of Licensee Event Reports (LERs) that was conducted to determine whether or not these LERs were prepared in accordance with the requirements set forth in 10 CFR 50.73, which became effective on January 1, 1984. The study was performed at the Idaho National Engineering Laboratory (INEL) by EG&G, Inc. The evaluation (NUREG/CR-4178) indicated that although the overall quality of the LERs was good, many LERs failed to meet all of the requirements. This supplementary report presents the methodology that was used to evaluate the LERs, the conclusions reached concerning problem areas in the reports, and suggestions as to how the overall quality and completeness of reports can be improved.

NUREG-1030 DFT FC: SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING NUCLEAR POWER PLANTS.Unresolved Safety Issue A-46.Draft Report For Comment. CHANG,T.Y. Division of Safety Technology (800428-851124). August 1985. 215pp. 8509180444. 32665:158.

The margin of safety provided in existing nuclear power plant equipment to resist seismically induced loads and perform their intended safety functions may vary considerably, because of significant changes in design criteria and method for the seismic qualification of equipment over the years. Therefore, the seismic qualification of equipment in operating plants should be reassessed to determine whether requalification is necessary. The objective of technical studies performed under the Task Action Plan A-46 was to establish an explicit set of guidelines and acceptance criteria to judge the adequacy of equipment under seismic loading at all operating plants, in lieu of requiring qualification to the current criteria that are applied to new plants.

NUREG-1031 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION,UNIT 3.Docket No. 50-423. (Northeast Nuclear Energy Company) * Division of Licensing (800428-851124). March 1985. 75pp. 8504090018. 29753:001.

The Safety Evaluation Report provides the results of the NRC staff review of Northeast Nuclear Energy Company's application for a license to operate the Millstone Nuclear Power Plant, Unit No. 3. The facility is located in Waterford Township, New London, Connecticut. This Supplement No. 1 updates the information contained in the Safety Evaluation Report, dated July 1984. This supplement also addresses the ACRS Report issued September 10, 1984.

NUREG-1031 S02: SAFETY EVALUATION RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION,UNIT 3.Docket No 50-423.(Northeast Nuclear Energy Company) * Division of Licensing (800428-851124). September 1985. 100pp. 8510010227. 32836:195.

The Safety Evaluation Report issued in August 1984 provided the results of the NRC staff review of Northeast Nuclear Energy Company's application for a license to operate the Millstone Nuclear Power Station, Unit No. 3. Supplement No. 1 to that report, issued in March 1985 updated the information contained in the Safety Evaluation Report and addressed the ACRS Report issued on September 10, 1984. This Report, Supplement No. 2 updates the information contained in the Safety Evaluation Report and Supplement No. 1 and addresses prior unresolved items. The facility is located in Waterford Township, New London, Connecticut.

NUREG-1031 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION,UNIT NO. 3.Docket No. 50-423.(Northeast Nuclear Energy Company) * Office of Nuclear Reactor Regulation, Director (post 851125). November 1985. 83pp. 8512030475. 33724:290.

The Safety Evaluation Report issued in August 1984 provided the results of the NRC staff review of Northeast Nuclear Energy Company's application for a license to operate the Millstone Nuclear Power Station, Unit No. 3. Supplement No. 1 to that report, issued in March 1985 updated the information contained in the Safety Evaluation Report and addressed the ACRS Report issued on September 10, 1984. Supplement No. 2

issued in September 1985 updated the information contained in the Safety Evaluation Report and Supplement No. 1 and addressed prior unresolved items. This Supplement, No. 3, provides more recent information regarding resolution or updating of some of the open and confirmatory items and license conditions identified in the Safety Evaluation Report. The facility is located in Waterford Township, New Condon County, Connecticut.

NUREG-1031 S04: SAFETY EVALUATION RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION,UNIT 3.Docket No. 50-423. (Northeast Nuclear Energy Company) * Office of Nuclear Reactor Regulation, Director (post 851125). November 1985. 232pp. 8512260100. 34063:073.

The Safety Evaluation Report issued in August 1984 provided the results of the NRC staff review of Northeast Nuclear Energy Company's application for a license to operate the Millstone Nuclear Power Station, Unit No. 3. Supplement No. 1 to that report, issued in March 1985 updated the information contained in the Safety Evaluation Report and addressed the ACRS Report issued on September 10, 1984. Supplement Nos. 2 and 3 dated September 1985 and November 1985, respectively, updated the information contained in the Safety Evaluation Report and Supplement No. 1 and addressed prior unresolved items. This Supplement No. 4, addresses the items concerning the issuance of a low power license (5%). The facility is located in Waterford Township, New London County, Connecticut.

NUREG-1032 DRFT FC: EVALUATION OF STATION BLACKOUT ACCIDENTS AT NUCLEAR POWER PLANTS.Technical Findings Related To Unresolved Safety Issue A-44.Draft Report For Comment. BARANOWSKI,P.W. Office of Nuclear Regulatory Research, Director. * Office of Nuclear Reactor Regulation, Director (pre-851125). May 1985. 200pp. 8506250217. 31212:301.

"Station Blackout," which is the complete loss of alternating current (AC) electrical power in a nuclear power plant, has been designated as Unresolved Safety Issue A-44. Because many safety systems required for reactor core decay heat removal and containment heat removal depend on AC power, the consequences of a station blackout could be severe. This report documents the findings of technical studies performed as part of the program to resolve this issue. The important factors analyzed include: the frequency of loss of offsite power; the probability that emergency or onsite AC power supplies would be unavailable; the capability and reliability of decay heat removal systems independent of AC power; and the liklihood that offsite power would be restored before systems that cannot operate for extended periods without AC power fail, thus resulting in core damage. This report also addresses effects of different designs, locations, and operational features on the estimated frequency of core damage resulting from station blackout events.

NUREG-1033: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF WPPSS NUCLEAR PROJECT NO. 3.Docket No. 50-508.(Washington Public Power Supply System) * Division of Licensing (800428-851124). May 1985. 247pp. 8505310065. 30671:001.

The Final Environmental Statement related to the operation of Washington Nuclear Project No. 3 by Washington Public Power Supply System, et al (Docket No. 50-508), located in Grays Harbor County, Washington, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This 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 an operating license could be granted.

NUREG-1037 DRFT FC: CONTAINMENT PERFORMANCE WORKING GROUP REPORT.Draft Report For Comment. * Division of Engineering (pre-851125). May 1985. 322pp. 8506140588. 30935:257.

Containment buildings for power reactors have been studied to estimate their leak rate as a function of increasing internal pressure and temperature associated with severe accident sequences involving significant core damage. Potential leak paths through containment penetration assemblies (such as equipment hatches, airlocks, purge and vent valves, and electrical penetrations) have been identified and their contributions to leak area for the containment are incorporated into containment leak rate and pressure temperature response as a function of time. Because of lack of reliable experimental data on the leakage behavior of containment penetrations and isolation barriers at pressure beyond their design conditions, an analytical approach has been used to estimate the leakage behavior of components found in specific reference plants that approximately characterize the various containment types.

NUREG-1038 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SHEARON HARRIS NUCLEAR POWER PLANT,UNIT 1.Docket No. 50-400. (Carolina Power And Light Company And North Carolina Eastern Municipal Power Agency) * Division of Licensing (800428-851124). June 1985. 65pp. 8506270137. 31258:277.

Supplement No. 2 to the Safety Evaluation Report for the application filed by Carolina Power and Light Company and North Carolina Eastern Municipal Power Agency for a license to operate the Shearon Harris Nuclear Power Plant, Unit 1 (Docket No. 50-400), located in Wake and Chatham Counties, North Carolina, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement provides more recent information regarding resolution of some of the open items identified in the Safety Evalutation Report and in Supplement No. 1. It also addresses one of the recommendations of the Advisory Committee on Reactor Safeguards in its report on the Shearon Harris Plant, dated January 16, 1984, which was inadvertently omitted in Supplement No. 1.

NUREG-1046: DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN UNSATURATED ZONE:TECHNICAL CONSIDERATIONS AND RESPONSE TO COMMENTS. HACKBARTH,C.J.; NICHOLSON,T.J.; EVANS,D.D. Division of Radiation Programs & Earth Sciences (post 840429). October 1985. 49pp. 8510290415. 33259:295.

On July 22, 1985, the U.S. Nuclear Regulatory Commission (NRC) promulgated amendments to 10 CFR Part 60 concerning disposal of high-level radioactive waste (HLW) in geologic repositories in the unsaturated zone (50 FR 29641). This report contains a discussion of the principal technical issues considered by the NRC staff during the development of these amendments. It expands or revises certain technical discussions originally presented in draft NUREG-1046 (February 1984) based on public comment letters and an increasing understanding of the physical, geochemical, and hydrogeologic processes operative in unsaturated geologic media. The following issues related to disposal of HLW within the unsaturated zone are discussed: hydrogeologic properties and conditions, heat dissipation and temperature, geochemistry, retrievability, potential for exhumation of the radioactive waste by natural causes and by human intrusion, the effects of future climatic changes on the level of the regional water table, and transport of radionuclides in the gaseous state. The changes to 10 CFR Part 60 in definitions, siting criteria, and design criteria for the geologic repository operations area are discussed. Other criteria examined by the NRC staff but which were not changed in the rule are the minimum 300-meter depth for waste emplacement, limitations on exploratory boreholes, backfill requirements, waste package design criteria, and provisions for ventilation.

NUREG-1047: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION,UNIT NO. 2.Docket No. 50-410. (Niagara Mohawk Power Corporation,et al) * Division of Licensing (800428-851124). February 1985. 652pp. 8502210335. 29054:001.

The Safety Evaluation Report for the application filed by the Niagara Mohawk Power Corporation, as applicant and co-owner, for a license to operate the Nine Mile Point Nuclear Station, Unit No. 2 (Docket No. 50-410), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located near Oswego, New York. Subject to favorable resolution of the items discussed in this report, the NRC staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

NUREG-1047 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION,UNIT NO. 2.Docket No. 50-410. (Niagara Mohawk Power Corporation,et al) * Division of Licensing (800428-851124). June 1985. 35pp. 8507050411. 31372:268.

This report supplements the Safety Evaluation Report (NUREG-1047, February 1985) for the application filed by Niagara Mohawk Power Corporation, as applicant and co-owner, for a license to operate the Nine Mile Point Nuclear Station Unit 2 (Docket No. 50-410). It has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located near Oswego, New York. Subject to favorable resolution of the items discussed in this report, the NRC Staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

NUREG-1047 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION,UNIT NO.2.Docket No. 50-410. (Niagara Mohawk Power Corporation) * Office of Nuclear Reactor Regulation, Director (post 851125). November 1985. 97pp. 8512190253. 34011:023.

This report supplements the Safety Evaluation Report (NUREG-1047, February 1985) for the application filed by Niagara Mohawk Power Corporation, as applicant and co-owner, for a license to operate the Nine Mile Point Nuclear Station Unit 2 (Docket No. 50-410). It has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located near Oswego, New York. Supplement 1 to the Safety Evaluation Report was published in June 1985 and contained the report from the Advisory Committee on Reactor Safeguards as well as the resolution to a number of outstanding issues from the Safety Evaluation Report. Subject to favorable resolution of the issues discussed in this report, the NRC staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

NUREG-1048 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF HOPE CREEK GENERATING STATION.Docket No. 50-354.(Public Service Electric And Gas Company,Atlantic City Electric Company) * Division of Licensing (800428-851124). March 1985. 77pp. 8503270534. 29542:275.

Supplement No. 1 to the Safety Evaluation Report on the application filed by Public Service Electric and Gas Company as applicant for itself and Atlantic City Electric Company, as owners, for a license to operate Hope Creek Generating Station has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Lower Alloways Creek Township in Salem County, New Jersey. This supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report.

NUREG-1048 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF HOPE CREEK GENERATING STATION.Docket No. 50-354.(Public Service Electric And Gas Company,Atlantic City Electric Company) * Division of Licensing (800428-851124). August 1985. 92pp. 8508190624. 32302:143.

Supplement No. 2 to the Safety Evaluation Report on the application filed by Public Service Electric and Gas Company as applicant for itself and Atlantic City Electric Company, as owners, for a license to operate Hope Creek Generating Station has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Lower Alloways Creek Township in Salem County, New Jersey. This supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report.

NUREG-1048 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF HOPE CREEK GENERATING STATION.Docket No. 50-354. (Public Service Electric And Gas Company,Atlantic City Electric Company) * Division of Licensing (800428-851124). October 1985. 100pp. 8511110437. 33416:273.

Supplement No. 3 to the Safety Evaluation Report on the application filed by Public Service Electric and Gas Company as applicant for itself and Atlantic City Electric Company, as owners, for a license to operate Hope Creek Generating Station has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Lower Alloways Creek Township in Salem County, New Jersey. This supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report and Supplements 1 and 2.

NUREG-1048 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF HOPE CREEK GENERATING STATION.Docket No. 50-354.(Public Service Electric And Gas Company,Atlantic City Electric Company) * Office of Nuclear Reactor Regulation, Director (post 851125). December 1985. 53pp. 8601070492. 34197:329.

Supplement No. 4 to the Safety Evaluation Report on the application filed by Public Service Electric and Gas Company on its own behalf as co-owner and as agent for the other co-owner, the Atlantic City Electric Company, for a license to operate Hope Creek Generating Station has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Lower Alloways Creek Township in Salem County, New Jersey. This supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report.

NUREG-1057: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BEAVER VALLEY POWER STATION,UNIT 2.Docket No. 50-412.(Duquesne Light Company,et al) * Division of Licensing (800428-851124). October 1985. 639pp. 8510290574. 33260:001.

The Safety Evaluation Report for the application filed by Duquesne Light Company, et al for a license to operate the Beaver Valley Power Station, Unit 2 in Beaver County, Pennsylvania, 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 Duquesne Light Company without endangering the health and safety of the public.

NUREG-1061 V02: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2:Evaluation Of Seismic Designs - A Review Of Seismic Design Requirements For Nuclear Power Plant Piping. * Piping Review Committee. April 1985. 213pp. 8505090016. 30254:043.

This document reports the position and recommendations of the NRC Piping Review Committee, Task Group on Seismic Design. The Task Group considered overlapping conservatism in the various steps of seismic design, the effects of using two levels of earthquake as a design criterion, and current industry practices. Issues such as damping values, spectra modification, multiple response spectra methods, nozzle and support design, design margins, inelastic piping response, and the use of snubbers are addressed. Effects of current regulatory requirements for piping design are evaluated, and recommendations for immediate licensing action, changes in existing requirements, and research programs are presented. Additional background information and suggestions given by consultants are also presented.

NUREG-1061 V02 ADD: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2 Addendum:Summary And Evaluation Of Historical Strong-Motion Earthquake Seismic Response And Damage To Aboveground Industrial Piping. * Piping Review Committee. * Stevenson & Associates. April 1985. 211pp. 8505100057. 30268:005.

Earthquake experience data for industrial piping has been summarized in this report. Conclusions and recommendations for improving the design of nuclear plant piping are made by the author. Input from R. L. Cloud, P. Yaner, and H. Shibata has been included. The material in this report served as background information for the NRC Piping Review Committee Seismic Design Task Group (and their consultants) in the development of the positions given in NUREG-1061 Volume 2.

NUREG-1061 V05: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 5:Summary - Piping Review Committee Conclusions and Recommendations. * Piping Review Committee. April 1985. 55pp. 8505070580. 30209:240.

This document summarizes a comprehensive review of NRC requirements for Nuclear Piping by the U.S. NRC Piping Review Committee. Four topical areas, addressed in greater detail in Volumes 1 through 4 of this report, are included: (1) Stress Corrosion Cracking in Piping of Boiling Water Reactor Plants, (2) Evaluation of Seismic Design, (3) Evaluation of Potential for Pipe Breaks, and (4) Evaluation of Other Dynamic Loads and Load Combinations. This volume summarizes the major issues, reviews the interfaces, and presents the Committee's conclusions and recommendations for updating NRC requirments on these issues. This report also suggests research or other work that may be required to respond to issues not amendable to resolution at this time.

NUREG-1065 R01: ACCEPTANCE CRITERIA FOR THE LOW ENRICHED URANIUM REFORM AMENDMENTS. EMEIGH,C.W.; GUNDERSEN,G.E.; WITHEE,C.J. Division of Safeguards. April 1985. 53pp. 8504240693. 29988:183.

This report documents a standard format suggested by the NRC for use in preparing fundamental nuclear material control plans as required by the Low Enriched Uranium Reform Amendments (portions of 10 CFR Part 74). The report also describes the necessary contents of a comprehensive plan and provides example acceptance criteria which are intended to communicate acceptable means of achieving the performance capabilities of the Reform Amendments. By using the suggested format, the license applicant will minimize administrative problems associated with the submittal, review and approval of the FNMC plan. Preparation of the plan in accordance with this format will assist the NRC in evaluating the plan and in standardizing the review and licensing process. However, conformance with this guidance is not required by the NRC. A license applicant who employs a format that provides an equal level of completeness and detail may use their own format.

NUREG-1070: NRC POLICY ON FUTURE REACTOR DESIGNS.Decisions On Severe Accident Issues In Nuclear Power Plant Regulation. * Office of Nuclear Reactor Regulation, Director (pre-851125). July 1985. 147pp. 8508150036. 32197:342.

On April 13, 1983, the U.S. Nuclear Regulatory Commission issued for public comment a "Proposed Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation" (48 FR 16014). This report presents and discusses the Commission's final version of that policy statement now entitled, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants." It provides an overview of comments received from the public and the Advisory Committee on Reactor Safeguards and the staff response to these. In addition to the Policy Statement, the report discusses how the policies of this statement relate to other NRC programs, including the Severe Accident Research Program; the implementation of safety measures resulting from lessons learned in the accident at Three Mile Island; safety goal development; the resolution of Unresolved Safety Issues and other Generic Safety Issues; and possible revisions of rules or regulatory requirements resulting from the Severe Accident Source Term Program. Also discussed are the main features of a generic decision strategy for resolving Regulatory Questions and Technical Issues relating to severe accidents; the development and regulatory use of new safety information; the treatment of uncertainty in severe accident decision making; and the development and implementation of a Systems Reliability Program for both existing and future plants to ensure that the realized level of safety is commensurate with the safety analyses used in regulatory decisions.

NUREG-1073: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF RIVER BEND STATION.Docket No. 50-458.(Gulf States Utilities And Cajun Electric Power Cooperative) * Division of Licensing (800428-851124). January 1985. 140pp. 8501240053. 28559:174.

This Final Environmental Statement contains the second assessment of the environmental impact associated with the operation of River Bend Station, pursuant to the National Environmental Policy Act of 1969 (NEPA) and Title 10 of the Code of Federal Regulations, Part 51, as amended, of the Nuclear Regulatory Commission regulations. This statement examines the environment, environmental consequences and mitigating actions, and environmental and economic benefits and costs.

NUREG-1079 DRFT FC: ESTIMATES OF EARLY CONTAINMENT FROM CORE MELT ACCIDENTS. Draft Report for Comment. * Office of Nuclear Reactor Regulation, Director (post 851125). December 1985. 255pp. 8601070488. 34184:024.

The thermal-hydraulic processes and corium debris-material interactions that can result from core melting in a severe accident have been studied to evaluate the potential effect of such phenomena on containment integrity. Pressure and temperature loads associated with representative accident sequences have been estimated for the six various LWR containment types used within the United States. Summaries distilling the analyses are presented and an interpretation of the results provided.

NUREG-1080 V02: LONG-RANGE RESEARCH PLAN FY 1986-FY 1990. * Office of Nuclear Regulatory Research, Director. August 1985. 157pp. 8509130047. 32621:067.

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. The plan will be updated annually.

NUREG-1085: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION,UNIT NO. 2.Docket No. 50-410.(Niagara Mohawk Power Corporation,et al) * Division of Licensing (800428-851124). May 1985. 373pp. 8505230685. 30548:001.

This Final Environmental Statement contains the assessment of the environmental impact associated with the operation of the Nine Mile Point Nuclear Station, Unit 2, pursuant to the National Environmental Policy Act of 1969 (NEPA) and Title 10 of the Code of Federal Regulations, Part 51, as amended, of the Nuclear Regulatory Commission regulations. This statement examines the environment, environmental consequences and mitigating actions, and environmental and economic benefits and costs.

NUREG-1087: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF VOGTLE ELECTRIC GENERATING PLANT,UNITS 1 AND 2.Docket Nos. 50-424 And 50-425.(Georgia Power Company) * Division of Licensing (800428-851124). March 1985. 461pp. 8504090235. 29748:035.

This Final Environmental Statement contains an assessment of the environmental impact associated with the operation of the Vogtle Electric Generating Plant, Units 1 and 2, pursuant to the National Environmental Policy Act of 1969 (NEPA) and Title 10 of the Code of Federal Regulations, Part 51 (10 CFR 51), as amended, of the Nuclear Regulatory Commission regulations. This statement examines the environmental impacts, environmental consequences and mitigating actions, and environmental and economic benefits and costs associated with station operation.

NUREG-1089: TECHNICAL SPECIFICATIONS FOR FERMI-2.Docket No. 50-341. (Detroit Edison Company) * Division of Licensing (800428-851124). March 1985. 501pp. 8504050281. 29672:001.

The Fermi-2 Facility 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-1094: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF BEAVER VALLEY POWER STATION,UNIT 2.Docket No. 50-412. (Duquesne Light Company) * Division of Licensing (800428-851124). September 1985. 300pp. 8509300559. 32792:017.

The Final Environmental Statement related to the operation of Beaver Valley Power Station, Unit 2 by Duquesne Light Company, et al (Docket No. 50-412), located in Beaver County, Pennsylvania, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This 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 weighing of environmental, technical and other factors, that an operating license could be granted.

NUREG-1095: EVALUATION OF RESPONSES TO IE BULLETIN 82-02.Degradation Of Threaded Fasteners In Reactor Coolant Pressure Boundary Of Pressurized Water-Reactor Plants. ANDERSON,W.; STERNER,P. Division of Emergency Preparedness & Engineering Response (Post 830103). May 1985. 75pp. 8506240221. IEB-82-02. 31177:219.

IE Bulletin 82-02 was issued by the NRC on June 2, 1982 to notify licensees about incidents of severe degradation of threaded fasteners. Responses to the Bulletin from 41 PWR licensees included data from recent regular inspections of reactor coolant pressure boundary components connections of six-inch size and larger. Statistical analysis is used to determine significant factors related to frequency of leakage incidents in connections, occurrence of degradation of bolts and studs, and the need for bolt replacement. Factors examined include the age of the plant, types of components, use of lubricants and sealants, and differences between plants. The compiled data indicate that, on the average, 10% of the bolted connections which were inspected show evidence of leaking and 80% of those undergo some degradation of the bolting. A significant decrease in the occurrence of bolting degradation events as the age of the plant increases is observed. Valves appear to be less subject to bolting corrosion. A group of 5 of the 41 plants accounted for about one-half of the reported leakage and corrosion events. The common characteristic found for 4 of these 5 plants was the lubricant used. The use of nickel-graphite based lubricants appears to offer a significantly reduced incidence of leakage and corrosion; while use of molybdenum disulfide-based lubricants and graphite-based lubricants appears to result in a significantly increased incidence of leakage and corrosion.

NUREG-1096: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE TRIGA TRAINING AND RESEARCH REACTOR AT THE UNIVERSITY OF UTAH.Docket No. 50-407. (University of Utah) * Division of Licensing (800428-851124). March 1985. 70pp. 8504090013. 29754:045.

This Safety Evaluation Report for the application filed by the University of Utah (UU) for a renewal of Operating License R-126 to continue to operate a training and research reactor facility 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 Utah and is located on its campus in Salt Lake City, Salt Lake County, Utah. The staff concludes that this training reactor facility can continue to be operated by UU without endangering the health and safety of the public.

NUREG-1098: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF OPERATING LICENSE FOR THE RESEARCH REACTOR AT MANHATTAN COLLEGE. Docket No. 50-199. (Manhattan College) * Division of Licensing (800428-851124). February 1985. 58pp. 8503130281. 29360:064.

This Safety Evaluation Report for the application filed by Manhattan College (MC) for a renewal of Operating License R-94 to continue to operate the MC 0.1 W 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 MC and is located two blocks away from the MC main campus in the Riverdale area of New York City, New York. The staff concludes that the reactor facility can continue to be operated by MC without endangering the health and safety of the public.

NUREG-1100 V01: FY 1986 BUDGET ESTIMATES. * Division of Budget & Analysis. January 1985. 87pp. 8502190023. 29013:211.

This report contains the fiscal year budget justifications to Congress. The budget estimates for salaries and expenses for fiscal year 1986-87 provide for obligations of $429,000,000 to be funded in total by a new appropriation.

NUREG-1103: CONTAMINATED MEXICAN STEEL.Importation Of Steel Into The United States That Had Been Inadvertently Contaminated With Cobalt-60 As A Result Of Scrapping Of A Teletherapy Unit. * Safeguards & Materials Program Branch. January 1985. 84pp. 8502110623. 28905:023.

This report documents the circumstances contributing to the inadvertent melting of Co-60 contaminated scrap metal in two Mexican steel foundries and the subsequent distribution of contaminated steel products into the United States. The report covers the tracing of the source to its origin, response actions to recover radioactive steel in the United States, and return of the contaminated materials to Mexico. Information outside of this scope is recounted as necessary, e.g., details of the incident on the Mexican side of the border. The incident resulted in very significant exposure to citizens of the United States.

NUREG-1104: TECHNICAL SPECIFICATIONS FOR WOLF CREEK GENERATING STATION, UNIT 1.Docket No. 50-482.(Kansas Gas And Electric Company) * Division of Licensing (800428-851124). March 1985. 500pp. 8504030425. 29601:354.

The Wolf Creek Generating 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-1105: REVIEW AND EVALUATION OF THE NUCLEAR REGULATORY COMMISSION SAFETY RESEARCH PROGRAM FOR FISCAL YEARS 1986 AND 1987. * ACRS - Advisory Committee on Reactor Safeguards. February 1985. 59pp. 8503010055. 29186:001.

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 Years 1986 and 1987. The report contains a number of comments and recommendations.

NUREG-1106: TECHNICAL SPECIFICATIONS FOR CATAWBA NUCLEAR STATION,UNIT 1. Docket No. 50-413.(Duke Power Company) * Division of Licensing (800428-851124). January 1985. 525pp. 8502060481. 28746:001.

The Catawba 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-1108: RADIOACTIVITY TRANSPORT FOLLOWING STEAM GENERATOR TUBE RUPTURE. HOPENFELD,J. Division of Accident Evaluation. March 1985. 45pp. 8504040001. 29630:269.

A review of the capabilities of the CITADEL computer code as well as plant experience to project radioactivity releases following steam generator tube rupture in PWR's shows that certain experimental data is needed for reliable offsite dose predictions. This article defines five parameters which are the key for such predictions and discusses the functional dependence of these parameters on various operational variables. A joint Westinghouse, Electric Power Research Institute, and the Nuclear Regulatory Commission program aimed at obtaining the five parameters empirically is described. Present status of the CITADEL code is also reviewed.

NUREG-1110: COMPARISON OF LICENSING ACTIVITIES FOR OPERATING PLANTS DESIGNED BY BABCOCK & WILCOX. THOMA,J.O. Division of Licensing (800428-851124). January 1985. 29pp. 8502070590. 28807:175.

This report provides a comparison of a number of licensing activities for the operating Babcock & Wilcox (B&W) plants with emphasis on Rancho Seco. The factors selected were a comparison of staff resources expended in FY84, active licensing action reviews, implementation of NUREG-0737 modifications, exemptions to regulations, SALP reports, enforcement actions, and Licensee Event Reports (LERs). The eight licensed operating plants examined are as follows: Arkansas Nuclear One Unit 1 (ANO-1), Crystal River Unit 3, Davis Besse, Oconee Units 1, 2, and 3, Rancho Seco, and Three Mile Island Unit 1 (TMI-1).

NUREG-1112: ENVIRONMENTAL ASSESSMENT FOR RENEWAL OF SPECIAL NUCLEAR MATERIAL LICENSE NO. SNM-368.(UNC Naval Products Division Of UNC Resources,Inc) * Division of Fuel Cycle & Material Safety. January 1985. 74pp. 8502120056. 28871:266.

This Environmental Assessment is issued by the U.S. Nuclear Regulatory Commission (NRC) in response to an application by UNC Naval Products, Division of UNC Resources, Inc., for the renewal of Special Nuclear Material (SNM) License No. SNM-368 for the operation of the existing fuel fabrication facility.

NUREG-1113: TECHNICAL SPECIFICATIONS FOR BYRON STATION UNITS 1 AND 2. Docket Nos. 50-454 And 50-455. (Commomwealth Edison Company) * Division of Licensing (800428-851124). February 1985. 510pp. 8503110132. 29326:001.

The Byron Station, Unit 1 and 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-1115: CATEGORIZATION OF REACTOR SAFETY ISSUES FROM A RISK PERSPECTIVE. * Division of Risk Analysis & Operations (post 840429). March 1985. 167pp. 8504030427. 29598:072.

This report presents the results of an effort to identify and rank reactor safety and risk issues identified from past Probabilistic Risk Assessments (PRAs) and other safety analyses. Because of the varied scope of these analyses, the list of issues may be incomplete. Nevertheless, those studies comprised ordered analyses to whatever their respective depths; hence, they warranted scrutiny for whatever insights they could reveal with respect to issue importance. The top ranked issues in terms of their contribution to the uncertainty in risk are described in some detail. All of these risk issues are compared to the "generic safety issues" for completeness and omission.

NUREG-1116: A REVIEW OF THE CURRENT UNDERSTANDING OF THE POTENTIAL FOR CONTAINMENT FAILURE FROM IN-VESSEL STEAM EXPLOSIONS. * Steam Explosion Review Group. June 1985. 521PP. 8507030716. 31335:001.

A group of experts was convened to review the current understanding of the potential for containment failure from in-vessel steam explosions during core meltdown accidents in LWRs. The Steam Explosion Review Group (SERG) was requested to provide assessments of: (i) the conditional probability of containment failure due to a steam explosion, (ii) a Sandia National Laboratory (SNL) report entitled "An Uncertainty Study of PWR Steam Explosions," NUREG/CR-3369, (iii) a SNL proposed steam explosion research program. This report summarizes the results of the deliberations of the review group. It also presents the detailed response of each individual member to each of the issues. The consensus of the SERG is that the occurrence of a steam explosion of sufficient energetics which could lead to alpha-mode containment failure has a low probability. The SERG members disagreed with the methodology used in NUREG/CR-3369 for the purpose of establishing the uncertainty in the probability of containment failure by a steam explosion. A consensus was reached among SERG members on the need for a continuing steam explosion research program which would improve our understanding of certain aspects of steam explosion phenomenology.

NUREG-1117: TECHNICAL SPECIFICATIONS FOR WATERFORD STEAM ELECTRIC STATION UNIT 3.Docket No. 50-382.(Louisiana Power And Light Company) * Office of Nuclear Reactor Regulation, Director (pre-851125). March 1985. 504pp. 8504030439. 29600:178.

The Waterford, 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-1118: ENVIRONMENTAL ASSESSMENT FOR RENEWAL OF SPECIAL NUCLEAR MATERIAL LICENSE NO. SNM-1107.Docket No. 70-1151. (Westinghouse Electric Corporation) * Division of Fuel Cycle & Material Safety. May 1985. 140pp. 8505240050. 30566:199.

This Environmental Assessment is issued by the U.S. Nuclear Regulatory Commission (NRC) in response to an application by the Westinghouse Electric Corporation for the renewal of Special Nuclear Material License No. SNM-1107 which covers the operations of the Columbia plant.

NUREG-1119: SAFETY EVALUATION REPORT RELATED TO

THE RENEWAL OF THE OPERATING LICENSE FOR THE CAVALIER TRAINING REACTOR AT THE UNIVERSITY OF VIRGINIA.Docket No. 50-396.(University Of Virginia) * Division of Licensing (800428-851124). May 1985. 62pp. 8506060716. 30780:214.

This Safety Evaluation Report for the application filed by the University of Virginia for a renewal of operating license number R-123 to continue to operate a training and research reactor (CAVALIER) 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 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-1122: KNOWLEDGES AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS.Pressurized Water Reactors. * Division of Human Factors Safety (800428-851124). July 1985. 400pp. 8508090488. 32120:352.

This document catalogs roughly 5300 knowledges and abilities of reactor operators and senior reactor operators. It results from a reanylsis of a much larger job-task analysis data base compiled by the Institute of Nuclear Power Operations (INPO). Knowledges and abilities are cataloged for 45 major power plant systems and 38 emergency evolutions, grouped according to 11 fundamental safety functions (e.g., reactivity control and reactor coolant system inventory control). With appropriate sampling from this catalog, operator licensing examinations having content validity can be developed. A structural sampling procedure for this catalog is under development by the Nuclear Regulatory Commission (NRC) and will be published as a companion document, "Examiners' Handbook for Developing Operator Licensing Examinations" (NUREG-1121). The examinations developed by using the catalog and handbook will cover those topics listed under Title 10, Code of Federal Regulations, Part 55.

NUREG-1125 V01: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 1,Part 1:ACRS Reports On Project Reviews (A-F). * ACRS - Advisory Committee on Reactor Safeguards. April 1985. 658pp. 8504220393. 29956:167.

This six-volume compilation contains over 1000 reports prepared by the Advisory Committee on Reactor Safeguards from September 1957 through December 1984. The reports are divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports alphabetized by project name and within project name by chronological order. Part 2 categorizes the reports by the most appropriate generic subject area and within subject area by chronological order.

NUREG-1125 V02: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 2,Part 1:ACRS Reports On Project Reviews (G-P). * ACRS - Advisory Committee on Reactor Safeguards. April 1985. 720pp. 8504220344. 29944:001.

See NUREG-1125,V01 abstract.

NUREG-1125 V03: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 3,Part 1:ACRS Reports On Project Reviews (Q-Z). * ACRS - Advisory Committee on Reactor Safeguards. April 1985. 563pp. 8504220389. 29954:329.

See NUREG-1125,V01 abstract.

NUREG-1125 V04: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 4,Part 2:ACRS Reports On Generic Subjects (Accident Analysis - Generic Items). * ACRS - Advisory Committee on Reactor Safeguards. April 1985. 627pp. 8504220406. 29961:225.

See NUREG-1125,V01 abstract.

NUREG-1125 V05: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 5,Part 2:ACRS Reports On Generic Subjects (HTGR - Regulatory Guides). * ACRS - Advisory Committee on Reactor Safeguards. April 1985. 630pp. 8504220396. 29958:102.

See NUREG-1125,V01 abstract.

NUREG-1125 V06: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 6,Part 2:ACRS Reports On Generic Subjects (RPA - Appendix C). * ACRS - Advisory Committee on Reactor Safeguards. April 1985. 567pp. 8504220402. 29960:015.

See NUREG-1125,V01 abstract.

NUREG-1126: TECHNICAL SPECIFICATIONS FOR SHOREHAM NUCLEAR POWER STATION, UNIT NO. 1.Docket No. 50-322.(Long Island Lighting Company) * Division of Licensing (800428-851124). July 1985. 489pp. 8507250207. 31784:001.

The Shoreham, 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-1127: RADIATION PROTECTION TRAINING AT URANIUM HEXAFLUORIDE AND FUEL FABRICATION PLANTS. BRODSKY,A.; SOONG,A.L.; BELL,J. Division of Radiation Programs & Earth Sciences (post 840429). May 1985. 33pp. 8506190048. 31017:173.

This report provides general information and references useful for establishing or operating radiation safety training programs in plants that manufacture nuclear fuels or process uranium compounds that are used in the manufacture of nuclear fuels. In addition to a brief summary of the principles of effective management of radiation safety training, the report also contains an appendix that provides a comprehensive checklist of scientific, safety, and management topics, from which appropriate topics may be selected in preparing training outlines for various job categories or tasks pertaining to the uranium nuclear fuels industry. The report is designed for use by radiation safety training professionals who have the experience to utilize the report to not only select the appropriate topics, but also to tailor the specific details and depth of coverage of each training session to match both employee and management needs of a particular industrial operation.

NUREG-1128: TRIAL EVALUATIONS IN COMPARISON WITH THE 1983 SAFETY GOALS. RIGGS,R.; SEGE,G. Division of Safety Technology (800428-851124). June 1985. 200pp. 8507080209. 31402:041.

This report provides retrospective comparisons of selected generic regulatory actions to the 1983 NRC safety goals, which had been issued for evaluation during a two-year period. The issues covered are those analyzed by the Office of Nuclear Reactor Regulation (NRR) (assisted in some cases by the Battelle Pacific Northwest Laboratory). The issues include auxiliary feedwater reliability, pressurized thermal shock, power-operated relief valve isolation, asymmetric blowdown loads on PWR primary systems, pool dynamic loads for BWR containments, and steam generator tube rupture. Calculated core-melt frequencies, mortality risks, and cost-benefit ratios are compared with the corresponding safety-goal quantitative design objectives. Considerations that should influence interpretation of the comparisons are discussed. Comments are included on whether and how the safety goals may help in the regulatory decision process and on problems encountered.

NUREG-1130: ENVIRONMENTAL ASSESSMENT FOR RENEWAL AND CONSOLIDATION OF MATERIALS LICENSE NOS SNM-362,SMB-405,08-00566-05, 08-00566-10,AND 08-00566-12. * Division of Fuel Cycle & Material Safety. March 1985. 45pp. 8504080548. 29713:038.

This Environmental Assessment is issued by the U.S. Nuclear Regulatory Commission (NRC) in response to an application by the U.S. Department of Commerce, National Bureau of Standards, for the renewal and consolidation of five Materials Licenses for radiological activities at the National Bureau of Standards site.

NUREG-1131: FINANCIAL ANALYSIS OF POTENTIAL RETROSPECTIVE PREMIUM ASSESSMENTS UNDER THE PRICE-ANDERSON SYSTEM. WOOD,R.S. Office of State Programs, Director. April 1985. 17pp. 8505080348. 30218:171.

Ten representative nuclear utilities have been analyzed over the period 1981-1983 to evaluate the effects of three levels of retrospective premiums on various financial indicators. This analysis continues and expands on earlier analyses prepared as background for deliberations by the U.S. Congress for possible extension or modification of the Price-Anderson Act.

NUREG-1132: TECHNICAL SPECIFICATIONS FOR DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2.Docket No. 50-323.(Pacific Gas and Electric Company) * Division of Licensing (800428-851124). April 1985. 466pp. 8505280011. 30605:007.

The Diablo 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-1133: TECHNICAL SPECIFICATIONS FOR PALO VERDE NUCLEAR GENERATING STATION,UNIT 1.Docket No. 50-528. (Arizona Public Service Company) * Division of Licensing (800428-851124). May 1985. 515pp. 8506240646. 31151:001.

The Palo Verde 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-1134: RADIATION PROTECTION TRAINING FOR PERSONNEL EMPLOYED IN MEDICAL FACILITIES. MCELROY,N.L.; BRODSKY,A. Division of Radiation Programs & Earth Sciences (post 840429). May 1985. 61pp. 8506130363. 30868:116.

This report provides information useful for planning and conducting radiation safety training in medical facilities to keep exposures as low as reasonably achievable, and to meet other regulatory, safety and loss prevention requirements in today's hospitals. A brief discussion of the elements and basic considerations of radiation safety training programs is followed by a short bibliography of selected references and sample lecture (or session) outlines for various job categories. This information is intended for use by a professional who is thoroughly acquainted with the science and practice of radiation protection as well as the specific procedures and circumstances of the particular hospital's operations. Topics can be added or subtracted, amplified or condensed as appropriate. This document does not set forth specific training program requirements for any particular hospital or type of medical institution or group of employees.

NUREG-1135: SAFETY EVALUATION REPORT RELATED TO THE CONSTRUCTION PERMIT AND OPERATING LICENSE FOR THE RESEARCH REACTOR AT THE UNIVERSITY OF TEXAS.Docket No. 50-602. (University of Texas) * Division of Licensing (800428-851124). May 1985. 88pp. 8506240665. 31152:223.

This Safety Evaluation Report for the application filed by the University of Texas for a construction permit and operating license to construct and operate a TRIGA 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 Texas and is located at the Univeristy's Balcones Research Center, about 7 miles (11.6 kilometers) north of the main campus in Austin, Texas. The staff concludes that the TRIGA reactor facility can be constructed and operated by the University of Texas without endangering the health and safety of the public.

NUREG-1136: TECHNICAL SPECIFICATIONS FOR WOLF CREEK GENERATING STATION, UNIT 1.Docket No. 50-482.(Kansas Gas And Electric Company) * Division of Licensing (800428-851124). June 1985. 498pp. 8506270251. 31257:142.

The Wolf Creek Generating 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-1137: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF VOGTLE ELECTRIC GENERATING PLANT,UNITS 1 AND 2.Docket Nos. 50-424 And 50-425.(Georgia Power Company,et al) * Division of Licensing (800428-851124). June 1985. 747pp. 8507030707. 31314:265.

The Safety Evaluation Report for the application filed by Georgia Power Company, Municipal Electric Authority of Georgia, Oglethorpe Power Corporation, and City of Dalton, Georgia, as applicants and owners, for licenses to operate the Vogtle Electric Generating Plant, Units 1 and 2 (Docket Nos. 50-424 and 50-425), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Burke County, Georgia, approximately 41.5 km (26 mi) south-southeast of Augusta, and on the Savannah River. Subject to favorable resolution of the items discussed in this report, the staff concludes that the applicant can operate the facility without endangering the health and safety of the public.

NUREG-1137 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF VOGTLE ELECTRIC GENERATING PLANT,UNITS 1 AND 2.DOCKET Nos. 50-424 And 50-425.(Georgia Power Company,et al) * Division of Licensing (800428-851124). October 1985. 56pp. 8511210589. 33586:265.

In June 1985, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1137) regarding the application of Georgia Power Company, Municipal Electric Authority of Georgia, Oglethorpe Power Corporation, and City of Dalton, Georgia, for a license to operate the Vogtle Electric Generating Plant, Units 1 and 2 (Docket Nos. 50-424 and 50-425). The facility is located in Burke County, Georgia, approximately 26 miles south-southeast of Augusta, Georgia, and on the Savannah River. This first supplement to NUREG-1137 provides recent information regarding resolution of some of the open and confirmatory items that remained unresolved at the time the Safety Evaluation Report was issued and provides the Advisory Committee on Reactor Safeguards letter dated August 13, 1985.

NUREG-1138: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE TRAINING AND RESEARCH REACTOR AT THE UNIVERSITY OF MICHIGAN.Docket No. 50-2.(University of Michigan) * Division of Licensing (800428-851124). July 1985. 73pp. 8508010299. 31928:095.

This Safety Evaluation Report for the application filed by the University of Michigan (UM) for a renewal of the Ford Nuclear Reactor operating license R-28 to continue to operate a training and research reactor facility 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 Michigan and is located at the North Campus of the University in Ann Arbor, Michigan. The staff concludes that this training reactor facility can continue to be operated by UM without endangering the health and safety of the public.

NUREG-1139: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE TRAINING & RESEARCH REACTOR AT THE UNIVERSITY OF LOWELL.Docket No. 50-223.(University of Lowell) * Division of Licensing (800428-851124). November 1985. 84pp. 8512030668. 33734:089.

This Safety Evaluation Report for the application filed by the University of Lowell for renewal of operating license number R-125 to confine to operate the open-pool type training and research reactor has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is owned and operated by the University of Lowell and is located on the university campus in Lowell, Massachussetts. The staff concludes that the open-pool type reactor facility can continue to be operated by the University of Lowell without endangering the health and safety of the public.

NUREG-1140 DRFT FC: A REGULATORY ANALYSIS ON EMERGENCY PREPAREDNESS FOR FUEL CYCLE AND OTHER RADIOACTIVE MATERIAL LICENSEES.Draft Report For Comment. MCGUIRE,S.A. Division of Risk Analysis & Operations (post 840429). June 1985. 125pp. 8507020410. 31309:033.

Potential accidents for 15 types of fuel cycle and other radioactive material licensees were analyzed. The most potentially hazardous accident, by a large margin, was determined to be the sudden rupture of a heated multi-ton cylinder of UF6. Acute fatalities offsite are probably not credible. Acute permanent injuries may be possible for many hundreds of meters, and clinically observable transient effects of unknown long term consequences may be possible for distances up to a few miles. These effects would be caused by the chemical toxicity of the UF6. Radiation doses would not be significant. The most potentially hazardous accident due to radiation exposure was determined to be a large fire at certain facilities handling large quantities of alpha-emitting radionuclides (i.e., Po-210, Pu-238, Pu-239, Am-241, Cm-242, Cm-244) or radioiodines (I-125 and I-131). However, acute fatalities or injuries to people offsite due to accidental releases of these materials do not seem plausible. The only other significant accident was identified as a long-term pulsating criticality at fuel cycle facilities handling high-enriched uranium or plutonium. An important feature of the most serious accidents is that releases are likely to start without prior warning. The releases would usually end within about half an hour. Thus protection actions would have to be taken quickly to be effective. There is not likely to be enough time for dose projections, complicated decisionmaking during the accident, or the participation of personnel not in the immediate vicinity of the site. The appropriate response by the facility is to immediately notify local fire, police, and other emergency personnel and give them a brief predetermined message recommending protective actions. Emergency personnel are generally well qualified to respond effectively to small accidents of these types.

NUREG-1141: TECHNICAL SPECIFICATIONS FOR FERMI-2 FACILITY.Docket No. 50-341. (Detroit Edison Company) * Division of Licensing (800428-851124). July 1985. 490pp. 8508070371. 32058:001.

The Fermi-2 facility 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-1142: TECHNICAL SPECIFICATIONS FOR RIVER BEND STATION.Docket No. 50-458. (Gulf States Utilities Company) BENEDICT,R. Division of Licensing (800428-851124). August 30, 1985. 541pp. 8509180507. 32663:337.

The River Bend Station 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-1143: SAFETY EVALUATION REPORT RELATED TO THE FULL-TERM OPERATING LICENSE FOR MILLSTONE NUCLEAR POWER STATION,UNIT NO. 1.Docket No. 50-245.(Northeast Nuclear Energy Company) * Division of Licensing (800428-851124). October 1985. 280pp. 8511070476. 33384:012.

The Safety Evaluation Report for the full-term operating license application filed by Northeast Nuclear Energy Company for Millstone Nuclear 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 New London County, Waterford, Connecticut. The staff concludes that the facility can continue to be operated without endangering the health and safety of the public.

NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN. MORRIS,B.M.; VORA,J.P. Division of Engineering Technology. July 1985. 48pp. 8508210443. 32337:277.

The nuclear plant aging research described in this plan is intended to resolve issues related to the aging and service wear of equipment and systems at commercial reactor facilities and their possible impact on plant safety. Emphasis has been placed on identification and characterization of the mechanisms of material and component degradation during sevice and evaluation of methods of inspection, surveillance, condition monitoring and maintenance as means of mitigating such effects. Specifically, the goals of the program are as follows: (1) To identify and characterize aging and service wear effects which, if unchecked, could cause degradation of structures, components, and systems and thereby impair plant safety. (2) To identify methods of inspection, surveillance and monitoring, or of evaluating residual life of structures, components, and systems, which will assure timely detection of significant aging effects prior to loss of safety function, and (3) To evaluate the effectiveness of storage, maintenance, repair and replacement practices in mitigating the rate and extent of degradation caused by aging and service wear.

NUREG-1145 V01: U.S. NUCLEAR REGULATORY COMMISSION 1984 ANNUAL REPORT. * Office of Resource Management, Director. June 1985. 234pp. 8506260386. 31246:057.

This report covers the major activities, events, decisions and planning that took place during fiscal year 1984 within the NRC or involving the NRC.

NUREG-1147: SEISMIC SAFETY RESEARCH PROGRAM PLAN. * Division of Engineering Technology. June 1985. 195pp. 8507080215. 31392:150.

This plan describes the safety issues, regulatory needs, and the research necessary to address these needs. The plan also discusses the relationship between current and proposed research within the NRC and research sponsored by other government agencies, universities, industry groups, professional societies, and foreign sources.

NUREG-1148: NUCLEAR POWER PLANT FIRE PROTECTION RESEARCH PROGRAM. DATTA,A. Division of Engineering Technology. July 1985. 32pp. 8508080059. 32072:263.

A program plan for nuclear power plant fire protection research has been presented in this report. The principal objective of the program is to create a data base that would reduce the uncertainties in fire probabilistic risk assessment of plants. A three-pronged approach of characterization of potential fires, determination of the ensuing environment, and determination of failure thresholds of safety-related equipment in that environment is described. The techniques are to be applied to estimating the fire safety margin available in a control room.

NUREG-1149: TECHNICAL SPECIFICATIONS FOR LIMERICK GENERATING STATION, UNIT 1.Docket No. 50-352. (Philadelphia Electric Company) MARTIN,R.E. Office of Nuclear Reactor Regulation, Director (pre-851125). June 1985. 500pp. 8508270346. 32380:272.

The Limerick Generating 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-1151: TECHNICAL SPECIFICATIONS FOR DIABLO CANYON NUCLEAR POWER PLANT UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company) * Division of Licensing (800428-851124). August 1985. 465pp. 8509100521. 32535:001.

The Diablo Canyon 1 and 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-1153: INSPECTION REPORT OF UNAUTHORIZED POSSESSION AND USE OF UNSEALED AMERICIUM-241 AND SUBSEQUENT CONFISCATION. J.C. Haynes Company,Newark,Ohio. CANIANO,R.J. Region 3, Office of Director. November 1985. 100pp. 8512190248. 34010:162.

This U.S. Nuclear Regulatory Commission report documents the circumstances surrounding the March 26, 1985, confiscation and subsequent decontamination activities related to the use of unauthorized quantities of americium-241 at the John C. Haynes Company (licensee) of Newark, Ohio. It focuses on the period from early February to July 26, 1985. The incident started when NRC Region III received information that John C. Haynes possessed unauthorized quantities of americium-241 and was conducting unauthorized activities (diamond irradiation). By July 26, 1985, the decontamination activities at the licensee's laboratory were concluded. The licensee's actions with diamond irradiation resulted in contamination in restricted and unrestricted areas of the facility. The confiscation and decontamination activities required the combined efforts of NRC, Federal Bureau of Investigation, U.S. Department of Energy, Oak Ridge Associated Universities, the State of Ohio, and the U.S. Environmental Protection Agency. The report describes the factual information and significant findings associated with the confiscation and decontamination activities.

NUREG-1154: LOSS OF MAIN AND AUXILIARY FEEDWATER EVENT AT THE DAVIS-BESSE PLANT ON JUNE 9, 1985. * Office of the Executive Director for Operations. July 1985. 103pp. 8508150428. 32220:150.

On June 9, 1985, Toledo Edison Company's Davis-Besse Nuclear Power Plant, located in Ottawa County, Ohio, experienced a partial loss of feedwater while the plant was operating at 90% power. Following a reactor trip, a loss of all feedwater occurred. The event involved a number of equipment malfuntions and extensive operator actions, including operator actions outside the control room. Several operator errors also occurred during the event. This report documents the findings of an NRC Team sent to Davis-Besse by the NRC Executive Director for Operations in conformance with the staff-proposed Incident Investigation Program.

NUREG-1155 V01: RESEARCH PROGRAM PLAN.Reactor Vessels. VAGINS,M. Division of Engineering Technology. July 1985. 41pp. 8508150435. 32220:110.

This document presents a plan for research in Reactor Vessels to be performed by the Materials Engineering Branch, MEBR, Division of Engineering Technology, (DET), Office of Nuclear Regulatory Research. It is one of four plans describing the ongoing research in the corresponding areas of MEBR activity, which are being published simultaneously in four volumes as follows: Vol. 1 Reactor Vessels, Vol. 2 Steam Generators, Vol. 3 Piping and Vol. 4 Non-Destructive Examination. These plans have been updated and are more detailed expansions of those originally published as part of the Long Range Research Plan for the Office of Nuclear Regulatory Research in NUREG-1080 Vol. 1.

NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators. MUSCARA,J.; SERPAN,C.Z. Division of Engineering Technology. July 1985. 18pp. 8508150425. 32220:093.

This report describes the NRC's research program related to steam generators. Mainly it discusses the program for evaluation of a removed-from-service degraded steam generator. Also discussed are projects to evaluate the vibration and wear that could result from chemical cleaning and NDE tasks for inservice inspection of steam generators.

NUREG-1155 V03: RESEARCH PROGRAM PLAN.Piping. VAGINS,M.; STROSNIDER,J. Division of Engineering Technology. July 1985. 16pp. 8508160080. 32230:056.

This document presents a plan for research in Piping to be performed by the Materials Engineering Branch, MEBR, Division of Engineering Technology, (DET), Office of Nuclear Regulatory Research. It is one of four plans describing the ongoing research in the corresponding areas of MEBR activity, which are being published simultaneously in four volumes as follows: Vol. 1 Reactor Vessels, Vol. 2 Steam Generators, Vol. 3 Piping, and Vol. 4 Non-Destructive Examination. These plans have been updated and are more detailed expansions of those originally published as part of the Long Research Plan for the Office of Nuclear Regulatory Research in NUREG-1080 Vol. 1.

NUREG-1155 V04: RESEARCH PROGRAM PLAN.Non-Destructive Examination. MUSCARA,J. Division of Engineering Technology. July 1985. 27pp. 8508160074. 32230:073.

This report describes the NRC research program in non-destructive evaluation. Projects are described for the development and evaluation of techniques for periodic inservice inspection of reactor components and for the continuous online monitoring of reactors. The areas of study described are ultrasonic, eddy current testing and acoustic emission.

NUREG-1157: ENVIRONMENTAL ASSESSMENT FOR RENEWAL OF SOURCE MATERIAL LICENSE NO. SUB-1010.Docket No. 40-8027. (Sequoyah Fuels Corporation) * Division of Fuel Cycle & Material Safety. August 1985. 406pp. 8509060254. 32504:010.

In response to an application for renewal of Source Material License SUB-1010 for the Sequoyah Fuels Corporation facility, the NRC staff prepared this Environmental Assessment. The Environmental Assessment includes discussions of the need for the proposed renewal action, alternatives to the action, and the environmental impacts of the proposed action and alternatives.

NUREG-1161: TECHNICAL SPECIFICATIONS FOR MILLSTONE NUCLEAR POWER STATION UNIT 3.Docket No. 50-423.(Northeast Nuclear Energy Company) * Division of Pressurized Water Reactor Licensing - A (post 851125). November 1985. 488pp. 8512180379. 33954:015.

The Millstone Nuclear Power Station, Unit No. 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-1164: INFORMATION ON THE CONFINEMENT CAPABILITY OF THE FACILITY DISPOSAL AREA AT WEST VALLEY,NEW YORK. * Office of Nuclear Regulatory Research, Director. December 1985. 62pp. 8601070531. 34155:001.

This report summarizes the previous NRC research studies, NRC licensee source term data and recent DOE site investigations tht deal with assessment of the radioctive waste inventory and confinement capability of the Facility Disposal Area (FDA) at West Valley, New York. The radioactive waste inventory for the FDA has a total rdioactivity of about 135,000 curies (Ci) and is comprised of H-3 (9,500 Ci), Co-60 (64,000 Ci), SR-90/Y-90 (24,300 Ci), Cs-137/Ba-137m (24,400 Ci), and Pu-241 (13,300 Ci). These wastes are buried in the Lavery Till, a glacial till unit comprised of a clayey silt with very low hydraulic conductivity properties. Recent studies of a tributylphoshate-kerosene plume moving through the shallow ground-water flow system in the FDA indicate a need to better assess the fracture flow components of this system particularly the weathered and fractured Lavery Till unit. The analysis of the deeper ground-water flow system studied by the USGS and NYSGS staffs indicates relatively long pathways and travel times to the accessible environment. Mass wasting, endemic, to the glacial-filled valley, contributes to the active slumping in the ravines surrounding the FDA and also need attention.

NUREG-1165: ESRP 7.1.1 "ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RELEASES OF RADIOACTIVE MATERIALS TO GROUNDWATER." WESCOTT,R.G. Office of Nuclear Reactor Regulation, Director (post 851125). November 1985. 13pp. 8512120111. ESRP 7.1.1. 33875:266.

Environmental Standard Review Plan (ESRP) 7.1.1 provides guidance to the staff for preparation of environmental assessments of "Radiological Impacts - Releases to Groundwater," an input to the staff's environmental statement which addresses the groundwater pathway consequences from postulated reactor core-melt accidents. The ESRP lists the type of information which should be collected, references that may be useful, and provides a procedure for uniform staff review of applicant analyses. The ESRP is applicable to both Construction Permit and Operating License Stage reviews.

NUREG-1167: TPDWR2:THERMAL POWER DETERMINATION FOR WESTINGHOUSE REACTORS,VERSION2.User's Guide. * Division of Emergency Preparedness & Engineering Response (Post 830103). December 1985. 158pp. 8601070496. 34198:236.

TPDWR2 is a computer program which was developed to determine the amount of thermal power generated by any Westinghouse nuclear power plant. From system conditions, TPDWR2 calculates enthalpies of water and steam and the power transferred to or from various components in the reactor coolant system and to or from the chemical and volume control system. From these results and assuming that the reactor core is operaing at constant power and is at thermal equilibrium, TPDWR2 calculates the thermal power generated by the reactor core. TPDWR2 runs on the IBM PC and XT computers when IBM Personal Computer DOS, Version 2.00 or 2.10, and IBM Personal Computer Basic, Version D2.00 or D2.10, are stored on the same diskette with TPDWR2.

NUREG-1172: TECHNICAL SPECIFICATIONS FOR RIVER BEND STATION. Docket No. 50-458.(Gulf States Utilities Company) BENEDICT,R. Office of Nuclear Reactor Regulation, Director (post 851125). November 1985. 539pp. 8512180358. 33955:143.

The River Bend Station 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-1173: TECHNICAL SPECIFICATIONS FOR PALO VERDE NUCLEAR GENERATING STATION,UNIT 2.Docket No.50-529.(Arizona Public Service Company) * Division of Pressurized Water Reactor Licensing - B (post 851125). December 1985. 513pp. 8601070482. 34191:289.

The Palo Verde 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/CP-0058 V01: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. SZAWLEWICZ,S.A. Office of Nuclear Regulatory Research, Director. January 1985. 434pp. 8502040194. 28714:188.

The papers published in this six volume report were presented at the Twelfth Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland during the week of October 22-26, 1984. The papers describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included twenty-six different papers presented by researchers from seven European countries, Japan, and Canada. Volume 1 presents information on Plenary Session - I, Integral System Tests, Separate Effects, International Programs in Thermal Hydraulics, and Calculation of Appendix K Conservatisms.

NUREG/CP-0058 V02: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. SZAWLEWICZ,S.A. Office of Nuclear Regulatory Research, Director. January 1985. 459pp. 8502060596. 28753:039.

The papers published in this six volume report were presented at the Twelfth Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland during the week of October 22-26, 1984. The papers describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included twenty-six different papers presented by researchers from seven European countries, Japan, and Canada. Volume 2 presents information on Pressurized Thermal Shock, Code Assessment and Improvement, 2D/3D Research Program, and the Nuclear Plant Analyzer Program.

NUREG/CP-0058 V03: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. * Office of Nuclear Regulatory Research, Director. January 1985. 731pp. 8502060432. 28750:001.

The papers published in this six volume report were presented at the Twelfth Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland during the week of October 22-26, 1984. The papers describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included twenty-six different papers presented by researchers from seven European countries, Japan, and Canada. Volume 3 presents information on Containment Systems Research, Fuel Systems Research, Accident Source Term Assessment, and Japanese Industry Safety Research.

NUREG/CP-0058 V04: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. SZAWLEWICZ,S.A. Office of Nuclear Regulatory Research, Director. January 1985. 388pp. 8502060428. 28752:012.

The papers published in this six volume report were presented at the Twelfth Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland during the week of October 22-26, 1984. The papers describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included twenty-six different papers presented by researchers from seven European countries, Japan, and Canada. Volume 4 presents information on Materials Engineering Research.

NUREG/CP-0058 V05: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. SZAWLEWICZ,S.A. Office of Nuclear Regulatory Research, Director. January 1985. 470pp. 8502060359. 28755:345.
The papers published in this six volume report were presented at the Twelfth Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland during the week of October 22-26, 1984. The papers describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included twenty-six different papers presented by researchers from seven European countries, Japan, and Canada. Volume 5 presents information on Mechanical Engineering, Structural Engineering, Seismic Research, Process Control, Instrumentation and Control Program, and Equipment Qualification and Nuclear Plant Aging.

NUREG/CP-0058 V06: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. SZAWLEWICZ,S.A. Office of Nuclear Regulatory Research, Director. January 1985. 515pp. 8502060357. 28757:094.
The papers published in this six volume report were presented at the Twelfth Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland during the week of October 22-26, 1984. The papers describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included twenty-six different papers presented by researchers from seven European countries, Japan, and Canada. Volume 6 presents information on Plenary Session - II, Human Factors and Safeguards Research, Health Effects and Radiation Protection, Risk Analysis, and EPRI Safety Research.

NUREG/CP-0059 V01: PROCEEDINGS OF THE MITI-NRC SEISMIC INFORMATION EXCHANGE MEETING,VOLUME I. WEISS,A.J. Brookhaven National Laboratory. April 1985. 423pp. 8506070372. BNL-NUREG-51821. 30796:001.
The first Japan Ministry of International Trade and Industry (MITI) - U.S. Nuclear Regulatory Commission (NRC) Seismic Information Exchange Meeting (SIEM) was held July 18-20, 1984 in Palo Alto, California. The purpose of SIEM was to provide technical information on seismic research being conducted under MITI and NRC sponsorships to the participants. The aim was to improve understanding of the seismic research in progress in Japan and the United States for possible identification of areas of mutual interest which could be the basis for future cooperation. Approximately 40 Japanese and U.S. technical specialists in seismic research participated in the meeting. These proceedings represent the compilation of the papers presented at the meeting.

NUREG/CP-0062: PROCEEDINGS OF THE CONFERENCE ON THE APPLICATION OF GEOCHEMICAL MODELS TO HIGH-LEVEL NUCLEAR WASTE REPOSITORY ASSESSMENT. JACOBS,G.K.; WHATLEY,S.K. Oak Ridge National Laboratory. May 1985. 130pp. 8506130505. ORNL/TM-9585. 30892:227.
A conference on the application of geochemical models in the assessment of high-level nuclear waste repositories was held to discuss the current status of geochemical code development, thermodynamic data bases, reaction kinetics, and coupled-process models as applied to site characterization and performance assessment activities. These proceedings include extended abstracts of the technical presentations given at the conference, a discussion of the role of geochemical modeling in predicting the performance of repositories, and a set of recommendations that identify the key developments needed in order for geochemical models to become more applicable for quantitative evaluations of repositories. Detailed recommendations relevant to the following subjects are discussed: (1) improved simulation of repository performance through inclusion of additional important geochemical processes and parameters into current geochemical models, (2) more careful attention to uncertainties associated with geochemical model calculations, (3) assigning priorities to (through sensitivity studies and critical evaluations) and then improving and/or obtaining important thermodynamic data, and (4) addressing the importance of kinetics in simulating repository behavior.

NUREG/CP-0063: PROCEEDINGS OF THE 1984 STATISTICAL SYMPOSIUM ON NATIONAL ENERGY ISSUES. KINNISON,R.; DOCTOR,P. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1985. 244pp. 8507230193. 31754:241.
The 1984 Statistical Symposium on National Energy Issues was the tenth in a series of annual symposia bringing together statisticians and other interested parties who are actively engaged in the pursuit of solving the nation's energy problems. Initially the symposium was sponsored by U.S. Department of Energy (DOE) and named the DOE Statistical Symposium. The symposium is organized by a steering committee made up of representatives from the national laboratories. The 1984 symposium was hosted by Pacific Northwest Laboratory, and it was organized around four special topical sessions: (1) Assessing and Assuring High Reliability, (2) Spatial Statistical, (3) Quantification of Informed Opinion, and (4) Health Effects of Energy Technologies. These were chosen research and data analysis. Several contributed papers were also presented.

NUREG/CP-0065: TRANSACTIONS OF THE 8TH INTERNATIONAL CONFERENCE ON STRUCTURE MECHANICS IN REACTOR TECHNOLOGY.Panel Session J-K: Status of Research In Structural And Mechanical Engineering For Nuclear Power Plants. BROWZIN,B.S. Division of Engineering Technology. June 1985. 266pp. 8507080187. 31393:277.
These transactions of the J-K/panel session include preprints of papers or abstracts which are listed in Volume A, "Introduction, General Contents, Authors' Index," Proceedings of the 8th International Conference on Structural Mechanics in Reactor Technology. These papers represent the body of the J-K/panel session, "Status of Research in Structural and Mechanical Engineering for Nuclear Power Plants," sponsored by the U.S. Nuclear Regulatory Commission.

NUREG/CP-0066: PROCEEDINGS OF AN INTERNATIONAL WORKSHOP ON HISTORIC DOSE EXPERIENCE AND DOSE REDUCTION (ALARA) AT NUCLEAR POWER PLANTS.MAY 29-JUNE 1,1984. HORAN,J.R.; BAUM,J.W.; DIONNE,B.J. Brookhaven National Laboratory. September 1985. 279pp. 8510040389. BNL-NUREG-51901. 32856:242.
Dose reduction data and experience from 28 foreign and 10 U.S. nuclear power plants was examined to determine causes for the wide variations in occupational dose from country to country. Major topics discussed were: steam generator and refueling maintenance problems; utility and supplier ALARA programs; effectiveness of dose-reduction modifications; attitudes and training; current and future dose-reduction research. While many parameters contribute to differences of occupational doses between plants from different nations, it is clear that most U.S. plants have higher collective dose equivalent per reactor per magawatt-year than most other countries, even for plants of similar size and age. Worldwide, Finnish and Swedish plants, both PWR and BWR, have achieved the lowest values. Major factors which contribute to low doses include: 1) minimization of cobalt in primary system components exposed to water, 2) careful plant design, layout and component segregation and shielding, 3) plant standardization, 4) selection of components and systems for increased reliability, 5) management interest and commitment, 6) minimum number of workers and indepth worker training, 7) careful control of primary system oxygen and pH, 8) good primary system water purity to minimize corrosion product formation, 9) use of special tools and robotics, 10) decontamination and passivation of primary systems and components, and 11) extent of backfitting and mandated inspections.

NUREG/CP-0070: PROCEEDINGS OF THE WORKSHOP ON SEISMIC AND DYNAMIC FRAGILITY OF NUCLEAR POWER PLANT COMPONENTS. HOFMAYER,C.H.; BANDYOPADHYAY Brookhaven National Laboratory. August 1985. 278pp. 8511180634. BNL-NUREG-51924. 33507:085.

The Workshop on Seismic and Dynamic Fragility of Nuclear Power Plant Components was held at Brookhaven National Laboratory (BNL) on June 5-7, 1985. The purpose of the workshop was to provide a forum for exchanging concepts, information and experiences on the fragility of electrical, control and mechanical equipment used in nuclear power plants when subjected to seismic and other dynamic environments. The workshop was divided into six sessions which included discussions on definition, uses and importance of component fragility; parameters affecting component fragility; categorizing equipment and existing test results; methodology and application of fragility data to equipment assemblies; equipment requiring future fragility testing; and, use of fragility data in PRA and Seismic Margin studies. The proceedings represent the compilation of the papers presented at the workshop.

NUREG/CP-0071: TRANSACTIONS OF THE THIRTEENTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. WEISS,A.J. Office of Nuclear Regulatory Research, Director. October 1985. 276pp. 8510170230. 33046:076.

This report contains summaries of papers on reactor safety research to be presented at the 13th Water Safety Research Information Meeting held at the National Bureau of Standards in Gaithersburg, Maryland, October 22-25, 1985. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, USNRC. Summaries of invited papers are also included, which cover the highlights of reactor safety research conducted by the electric utilities through the Electric Power Research Institute, the nuclear industry, and the research of government and industry in Europe and Japan. The summaries have been compiled in one report to provide a basis for meaningful discussion and information exchange during the course of the meeting, and are given in the order of their presentation in each session.

NUREG/CR-1677 V02: PIPING BENCHMARK PROBLEMS.VOLUME II DYNAMIC ANALYSIS INDEPENDENT SUPPORT MOTION RESPONSE SPECTRUM METHOD. BEZLER,P.; SUBUDHI,M.; HARTZMAN,M. Brookhaven National Laboratory. August 1985. 401pp. 8509160031. BNL-NUREG-51267. 32625:321.

Four benchmark problems and solutions were developed for verifying the adequacy of computer programs used for the dynamic analysis and design of elastic piping systems by the independent support motion, response spectrum method. The dynamic loading is represented by distinct sets of support excitation spectra assumed to be induced by non-uniform excitation in three spatial directions. Complete input descriptions for each problem are provided and the solutions include predicted natural frequencies, participation factors, nodal displacements and element forces for independent support excitation and also for uniform envelope spectrum excitation. Solutions to the associated anchor point pseudo-static displacements are not included.

NUREG/CR-1755 ADD01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING NUCLEAR REACTORS AT MULTIPLE-REACTOR STATIONS.Effects On Decommissioning Of Interim Inability To Dispose Of Wastes Offsite. MOORE,E.B. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1985. 41pp. 8505070571. 30209:200.

The purpose of this addendum is to examine the impacts of an interim inability to carryout offsite disposal of radioactive wastes and spent fuel on the decommissioning of multiple-reactor power station. The example selected for study is a four-PWR station in which each PWR is prepared for safe storage at two-year intervals, held in safe storage for 100 year intervals. BWRs are neglected for simplicity and in the expectation that the results would be similar to those for PWRs. Only SAFSTOR is considered because DECON and ENTOMB are unsuitable by definition for interim storage of radioactive wastes and/or spent fuel. It is assumed that all radioactive wastes and spent fuel are shipped offsite by the end of decommissiong.

NUREG/CR-2000 V03N12: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of December 1984. * Oak Ridge National Laboratory. January 1985. 49pp. 8501280397. ORNL/NSIC-200. 28572:297.

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 the 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 for those events (and revisions to those events) occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73-144) on July 26, 1983. NUREG-1022, Licensee Event Report System - Description of Systems and Guidelines for Reporting, provides supporting guidance and information on the revised LER rule. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, keyword, system and general keyword indexes are assigned by the computer using correlation tables from the Sequence Coding and Search System.

NUREG/CR-2000 V04 N1: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of January 1985. * Oak Ridge National Laboratory. February 1985. 59pp. 8503150302. ORNL/NSIC-200. 29390:226.

See NUREG/CR-2000,V03,N12 abstract.

NUREG/CR-2000 V04 N2: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of February 1985. * Oak Ridge National Laboratory. February 1985. 70pp. 8504030412. ORNL/NSIC-200. 29604:359.

See NUREG/CR-2000,V03,N12 abstract.

NUREG/CR-2000 V04 N3: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of March 1985. * Oak Ridge National Laboratory. April 1985. 78pp. 8505070557. ORNL/NSIC-200. 30210:092.

See NUREG/CR-2000,V03,N12 abstract.

NUREG/CR-2000 V04 N4: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of April 1985. * Oak Ridge National Laboratory. May 1985. 87pp. 8506130364. ORNL/NSIC-200. 30867:262.

See NUREG/CR-2000,V03,N12 abstract.

NUREG/CR-2000 V04 N5: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of May 1985. * Oak Ridge National Laboratory. June 1985. 111pp. 8507030669. ORNL/NSIC-200. 31314:155.

See NUREG/CR-2000,V03,N12 abstract.

NUREG/CR-2000 V04 N6: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of June 1985. * Oak Ridge National Laboratory. July 1985. 133pp. 8508150071. ORNL/NSIC-200. 32196:336.

See NUREG/CR-2000,V03,N12 abstract.

NUREG/CR-2000 V04 N7: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of July 1985. * Oak Ridge National Laboratory. August 1985. 129pp. 8509060195. ORNL/NSIC-200. 32505:141.

See NUREG/CR-2000,V03,N12 abstract.

NUREG/CR-2000 V04 N8: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of August 1985. * Oak Ridge National Laboratory. September 1985. 147pp. 8510030309. ORNL/NSIC-200. 32848:162.
See NUREG/CR-2000,V03,N12 abstract.

NUREG/CR-2000 V04 N9: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of September 1985. * Oak Ridge National Laboratory. October 1985. 118pp. 8511110422. ORNL/NSIC-200. 33417:093.
See NUREG/CR-2000,V03,N12 abstract.

NUREG/CR-2000 V04N10: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of October 1985. * Oak Ridge National Laboratory. November 1985. 123pp. 8512100733. ORNL/NSIC-200. 33832:146.
See NUREG/CR-2000,V03,N12 abstract.

NUREG/CR-2000 V04N11: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of November 1985. * Oak Ridge National Laboratory. December 1985. 117pp. 8601070500. ORNL-NSIC-200. 34155:131.
See NUREG/CR-2000,V03,N12 abstract.

NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984. WEISS,A.J. Brookhaven National Laboratory. February 1985. 146pp. 8503080486. BNL-NUREG-51454. 29295:096.

This progress report will describe current activities and technical progress in the programs at Brookhaven National Laboratory sponsored by the Division of Accident Evaluation, Division of Engineering Technology, and Division of Risk Analysis & Operations of the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. The projects reported are the following: High Temperature Reactor Research, SSC Development, Validation and Application, CRBR Balance of Plant Modeling, Thermal-Hydraulic Reactor Safety Experiments, Development of Plant Analyzer, Code Assessment and Application (Transient and LOCA Analyses), Thermal Reactor Code Development (RAMONA-3B), Calculational Quality Assurance in Support of PTS; Stress Corrosion Cracking of PWR Steam Generator Tubing, Probability Based Load Combinations for Design of Category I Structures, Mechanical Piping Benchmark Problems, Identification of Age-Related Failure Modes; Analysis of Human Error Data for Nuclear Power Plant Safety Related Events, Human Factors Aspects of Safety/Safeguards Interactions, Emergency Action Levels, and Protective Action Decision Making.

NUREG/CR-2331 V04 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July * -September 30,1984. WEISS,A.J. Brookhaven National Laboratory. May 1985. 117pp. 8506060147. BNL-NUREG-51454. 30781:002.
See NUREG/CR-2331,V04,N02 abstract.

NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984. WEISS,A.J. Brookhaven National Laboratory. May 1985. 139pp. 8507050378. BNL-NUREG-51454. 31373:001.
See NUREG/CR-2331,V04,N02 abstract.

NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985. WEISS,A.J. Brookhaven National Laboratory. August 1985. 250pp. 8512120133. BNL-NUREG-51454. 33874:133.
See NUREG/CR-2331,V04,N02 abstract.

NUREG/CR-2482 V06: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program.October 1983 - March 1984. SOO,P. Brookhaven National Laboratory. March 1985. 59pp. 8504030414. BNL-NUREG-51494. 29605:065.

This report is part of an ongoing effort to review the national high level waste package program. The contributions of individual waste package components to containment and controlled release of radionuclides after emplacement in salt, basalt, tuff and granite repositories are evaluated. The U.S. crystalline (granite) repository program is reviewed and relevant foreign data are outlined. The use of crushed salt, bentonite and zeolite-containing packing materials is discussed. Temperatures and gamma irradiation are shown to be important environmental parameters in assessing waste package performance.

NUREG/CR-2482 V07: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program April 1984-September 1984. SOO,P. Brookhaven National Laboratory. March 1985. 88pp. 8504030408. BNL-NUREG-51494. 29604:266.

The present effort is part of an ongoing task to review the national high level waste package effort. It includes evaluations of reference waste form, container, and packing material components with respect to determining how they may contribute to the containment and controlled release of radionuclides after waste packages have been emplaced in salt, basalt, tuff, and granite repositories. In the current Biannual Report a review was carried out to determine the ability of spent fuel cladding to provide additional radionuclide containment capability should the container/overpack system fail prematurely.

NUREG/CR-2482 V08: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period October 1984 - March 1985. DAVIS,M.S.; BREWSTER,C.; GAUSE,E.; et al. Brookhaven National Laboratory. December 1985. 300pp. 8601070538. BNL-NUREG-51494. 34181:249.

A large number of technical reports on waste package component performance were reviewed over the last year in support of the NRC's review of the Department of Energy's (DOE's) Environmental Assessment reports. The intent was to assess in some detail the quantity and quality of the DOE data and their relevance to the high-level waste repository site selection process. A representative selection of the reviews is presented for the salt, basalt and tuff repository projects. Areas for future research have been outlined.

NUREG/CR-2482 V09: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period April 1985-September 1985. SULLIVAN,T.; JAIN,H.; ABRAHAM,T.; et al. Brookhaven National Laboratory. December 1985. 98pp. 8601070503. BNL-NUREG-51494. 34183:180.

Detailed evaluations continued on DOE reports and papers concerned with the evaluation of waste package component behavior. The intent was to estimate the quantity and relevance of data being generated for barrier system performance analysis. In addition, several review studies have been completed to evaluate progress in the DOE waste package program. These include work on the selection of a glass composition for West Valley, New York, high level waste, a description of the system at West Valley for vitrifying the waste, and reviews of papers included in the Defense High-Level Waste Leaching Program and the recent Tucson, Arizona, "Waste Management '85" Conference.

NUREG/CR-2531 R03: INTRODUCTORY USER'S MANUAL FOR THE U.S. NUCLEAR REGULATORY COMMISSION REACTOR SAFETY RESEARCH DATA BANK. HARDY,H.A.; LAATS,E.T. EG&G, Inc. May 1985. 95pp. 8505170007. EGG-2164. 30483:079.

The United States Nuclear Regulatory Commission (USNRC) 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 (INEL). 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 correlation needs of the nuclear industry. The administrative portion of the program provides data entry, documentaton, training, and advisory services to users and the USNRC. The NRC/DAE Data Bank and the capabilities of the data access software are described in this document.

NUREG/CR-2663 V01: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Main Report. * Ertec Western, Inc., May 1985. 574pp. 8506270457. 31261:001.

Evaluation of the geologic isolation of radioactive materials from the biosphere requires an intimate knowledge of site geologic conditions, which is gained through precharacterization and site characterization studies. This report presents the results of an intensive literature review, analysis and compilation to delineate the information needs, applicable techniques and evaluation criteria for programs to adequately characterize a site in six geologic media. These media, in order of presentation, are: granite, shale, basalt, tuff, bedded salt, and domed salt. Guidelines are presented to assess the efficacy (application, effectiveness, and resolution) of currently used to exploratory and testing techniques for precharacterization or characterization of a site. These guidelines include the reliability, accuracy, and resolution of techniques deemed acceptable, as well as cost estimate of various field and laboratory techniques used to obtain the necessary information. Guidelines presented do not assess the relative suitability of media. This report consists of two volumes: main report and appendices.

NUREG/CR-2663 V02: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Appendices. * Ertec Western, Inc., May 1985. 700pp. 8506270247. 31259:001.

See NUREG/CR-2663,V01 abstract.

NUREG/CR-2718: STEAM EXPLOSION EXPERIMENTS WITH SINGLE DROPS OF IRON OXIDE MELTED WITH A CO2 LASER.Part II:Parametric Studies. NELSON,L.S.; DUDA,P.M. Sandia National Laboratories. April 1985. 154pp. 8506140047. SAND82-1105. 30908:219.

The steam explosion experiments performed with single drops of molten iron oxide melted with a CO(2) laser, described in Part I of this report, were extended here. The following major parameters were varied: ambient pressure, water temperature and subcooling, melt temperature, and melt composition. Also, a few scoping experiments were performed to explore the effects of changing the nature of the coolant, and the viscosity of the melt. As each of the four major parameters was varied, thresholds could be located beyond which explosions were suppressed. However, in general, the explosions could be reinitiated by increasing the magnitude of the triggering pulse. The effects of increasing the ambient pressure up to 1.12 MPa were faster and finer melt fragmentation, and faster and more complete transfer of heat from melt to water. Moreover, triggering became easier over the range of ambient pressure between about 0.15 MPa and approximately 0.7 MPa.

NUREG/CR-2800 S03: GUIDELINES FOR NUCLEAR POWER PLANT SAFETY ISSUE PRIORITIZATION INFORMATION DEVELOPMENT. ANDREWS,W.B.; BICKFORD,W.E.; COUNTS,C.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1985. 143pp. 8510030435. 32847:289.

This supplemental report is the fourth in a series that document and use methods developed by the Pacific Northwest Laboratory to calculate, for prioritization purposes, the risk, dose and cost impacts of implementing resolutions to reactor safety issues. The initial report in this series was published by Andrews et al. in 1983 as NUREG/CR-2800. This supplement consists of two parts describing separate research efforts: (1) an alternative human factors methodology approach and (2) a prioritization of the NRC's Human Factors Program Plan. The alternative human factors methodology approach may be used in specific future cases in which the methods identified in the initial report (NUREG/CR-2800) may not adequately assess the proper impact for resolution of new safety issues. The alternative methdology included in this supplement is entitled Methodology for Estimating the Public Risk Reduction Affected by Human Factors Improvement. The prioritization section of this report is entitled Prioritization of the U.S. Nuclear Regulatory Commission Human Factors Program Plan.

NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices. BARI,R.A.; BUSLIK,A.J.; CHO,N.Z.; et al. Brookhaven National Laboratory. August 1985. 203pp. 8509110037. BNL-NUREG-51559. 32561:010.

A procedures guide for the performance of probabilistic safety assessment has been prepared for interim use in the Nuclear Regulatory Commission programs. It will be revised as comments are received, and as experience is gained from its use. The probabilistic safety assessment studies performed are intended to produce probabilistic predictive models that can be used and extended by the utilities and by NRC to sharpen the focus of inquiries into a range of issues affecting reactor safety. This first volume of the guide describes the determination of the probability (per year) of core damage resulting from accident initiators internal to the plant (i.e., intrinsic to plant operation) and from loss of off-site electric power. The scope includes human reliability analysis, a determination of the importance of various core damage accident sequences, and an explicit treatment and display of uncertainties for key accident sequences. The second volume deals with the treatment of the so-called external events including seismic disturbances, fires, floods, etc. Ultimately, the guide will be augmented to include the plant-specific analysis of in-plant processes (i.e., containment performance). This guide provides the structure of a probabilistic safety study to be performed, and indicates what products of the study are valuable for regulatory decision making. For internal events, methodology is treated in the guide only to the extent necessary to indicate the range of methods which is acceptable; ample reference is given to alternative methodologies which may be utilized in the performance of the study. For external events, more explicit guidance is given.

NUREG/CR-2815 V02 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 8-12. MCCANN,M.; REED,J.W.; RUGER,C.; et al. Brookhaven National Laboratory. August 1985. 369pp. 8509110050. BNL-NUREG-51559. 32560:001.

See NUREG/CR-2815,V01,R1 abstract.

NUREG/CR-2850 V03: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1981. BAKER,D.A.; PELOQUIN,R.A. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1985. 128pp. 8502060603. PNL-4221. 28745:070.

Population radiation dose commitments have been estimated from reported radionuclide releases from commercial power reactors operating during 1981. 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. 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 from 48 sites ranged from a high of 20 person-rem to a low of 0.008 person-rem with an arithmetic mean of 3 person-rem. The total population dose for all sites was estimated at 160 person-rem for the 98 million people considered at risk. The average individual dose commitment from all pathways on a site basis ranged from a low of 1 x 10(-5) mrem to a high of 0.05 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-2951: THE D9 EXPERIMENT.Heat Removal From Stratified UO2 Debris. OTTINGER,C.A.; MITCHELL,G.W.; LIPINSKI,R.J.; et al. Sandia National Laboratories. June 1985. 74pp. 8507050431. SAND84-1838. 31371:101.

The D9 experiment investigated the coolability of a shallow (77 mm), stratified urania bed in sodium. The bed was fission heated in the Annular Core Research Reactor (ACRR) at Sandia National Laboratories to simulate the effects of radioactive decay heating. It was the first stratified debris bed experiment to use an extended UO2 particle size distribution (0.038 to 4.0 mm). Dryout occurred at powers ranging from 0.10 to 0.58 W/g, which was close to the incipient boiling power and before channels penetrated the subcooled zone in the bed, even with subcoolings as low as 80 degrees centigrade. Channel penetration was observed after dryout began, but the bed became only moderately more coolable. All these observations agree with current models.

NUREG/CR-3005: SUMMARY OF THE NUCLEAR REGULATORY COMMISSION'S LOFT PROGRAM RESEARCH FINDINGS. NALEZNY,C.L. EG&G, Inc. April 1985. 199pp. 8507050424. EGG-2231. 31372:001.

This document is a summary of the main research results of the Loss-of-Fluid Test (LOFT) Program relative to code assessment, code development, licensing, rulemaking, safety technology, and reactor operations. 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 transient response of the LOFT system to accident events is similar to large [1000 MW(e)] commercial PWRs. The main objectives of the LOFT Experimental Program were to qualify the engineered safety systems used in commercial PWRs and to verify the computer codes used in safety analyses. The LOFT Program contributed to the improvement of computer codes used to predict the response of commercial PWRs, demonstrated the adequacy of engineered safety systems, and contributed to improved understanding of PWR accident phenomena, particularly those associated with the evaluation model in Appendix K to 10 CFR 50 (the "ECCS rule").

NUREG/CR-3019: RECOMMENDED WELDED CRITERIA FOR USE IN THE FABRICATION OF SHIPPING CONTAINERS FOR RADIOACTIVE MATERIALS. MONROE,R.E.; WOO,H.H.; SEARS,R.G. Lawrence Livermore National Laboratory. March 1985. 16pp. 8504040007. UCRL-53044. 29619:236.

Welding and related operations are evaluated to assess the controls required to prevent weld-related failure of shipping containers used for transportation of radioactive materials. The report includes (1) recommended criteria for controlling welding as applied to shipping containers, and (2) a discussion of modifications of the recommended industry Codes as applied to shipping containers.

NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA. KATO,W.Y.; HALL,R.E.; TEICHMANN,T.; et al. Brookhaven National Laboratory. February 1985. 267pp. 8503080498. BNL-NUREG-51609. 29285:134.

Brookhaven National Laboratory's Department of Nuclear Energy (DNE) has performed a study of the Licensee Event Report (LER) system. The objective of the study was to assess the feasibility of modifying the LER reporting system as proposed by NRC-AEOD, and/or developing an alternative plan that would in addition collect information about significant events amenable to statistical analysis, such as multi-case, multi-variate analysis. The study indicated that the LERs constitute reports from a large variety of events which have in most cases many different plant parameters, both measured and currently not measured, to characterize the event. In order to determine event-specific plant parameters required for statistical and deterministic analysis, a data matrix approach could be measured and recorded, and those which are required for certain types of events involving thermal-hydraulics and neutronics as illustrative of events requiring in-depth analysis. Also included in the study was a review of INPO's Nuclear Plant Reliability Data System; NASA's Problem Reporting and Corrective Action (PRACA) program; Electricite de France's KIT system, an automatic computer-based reactor parameter monitoring and recording system; and the regulatory relationship between the FAA and the commercial airline industry.

NUREG/CR-3091 V04: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1983 - March 1984. JAIN,H.; VEAKIS,E.; SOO,P. Brookhaven National Laboratory. June 1985. 29pp. 8507050398. BNL-NUREG-51630. 31373:314.

The current study is part of an ongoing task to specify tests that may be used to verify that engineered waste package/repository systems comply with NRC radionuclide containment and controlled release performance objectives. Work covered in this report includes tuff packing material for use in a high level waste tuff repository. Ranges of repository conditions relevant to its testing and other factors important for its performance are discussed.

NUREG/CR-3091 V05: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period April 1984 - September 1984. JAIN,H.; VEAKIS,E.; SOO,P. Brookhaven National Laboratory. June 1985. 34pp. 8507050402. BNL-NUREG-51630. 31372:302.

This ongoing study is part of a task to specify tests that may be used to verify that engineered waste packages/repository systems comply with NRC radionuclide containment and controlled release performance objectives. Work covered in this report includes crushed tuff packing material for use in a high level waste tuff repository. A review of available tests to quantify packing performance is given together with recommendations for future testing work.

NUREG/CR-3091 V06: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1984 - March 1985. SOO,P. Brookhaven National Laboratory. July 1985. 171pp. 8508220318. BNL-NUREG-51630. 32346:154.

The potential of WAPPA, a second-generation waste package system code, to meet the needs of the regulatory community are analyzed. The analysis includes an indepth review of WAPPA's individual process models and a review of WAPPA's operation. It is concluded that the code is of limited use to the NRC in the present form. Recommendations for future improvement, usage, and implementation of the code are given. This report also describes the results of a testing program undertaken to determine the chemical environment that will be present near a high-level waste package emplaced in a basalt repository. For this purpose, low carbon 1020 steel (a current BWIP reference container material), synthetic basaltic groundwater and a mixture of bentonite and basalt were exposed, in an autoclave, to expected conditions some period after repository sealing (150 degrees centigrade, approximately 10.4 MPa). Parameters measured include changes in gas pressure with time and gas composition, variation in dissolved oxygen (DO), pH and certain ionic concentrations of water in the packing material across an imposed thermal gradient, mineralogic alteration of the basalt/bentonite mixture, and carbon steel corrosion behavior. A second testing program was also initiated to check the likelihood of stress corrosion cracking of austenitic stainless steels and Incoloy 825 which are being considered for use as waste container materials in the tuff repository program.

NUREG/CR-3145 V03: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION - ANNUAL REPORT.(October 1982 - September 1983, Volume 3). POLLACK,H.N.; CHRISTENSEN,D.; WELC,J. Michigan, Univ. of, Ann Arbor, MI. May 1985. 51pp. 8507230043. 31753:299.

Earthquake activity in the Western Ohio - Indiana region has been monitored with a precision seismograph network consisting of nine stations located in west-central Ohio and four stations sited in Indiana. Twelve local and near-regional earthquakes have been recorded and located during this report period, ranging in magnitude from 0.3 to 4.0 m(blg). An event which occurred on January 14, 1984, in Toledo, Ohio, and two events on July 28 and August 29, 1984, near Terre Haute, Indiana, were felt. Only minor damage was reported from these events. Of the twelve events, four occurred in the center of the Ohio array, three occurred near the city of Toledo, Ohio, four occurred in Indiana (including one on the Indiana-Illinois border), and one was located near Chicago, Illinois. Teleseismic P-wave residuals have been updated and evaluated by back projection to various depths in the lower crust. The residuals are found to correspond roughly to magnetic anomalies in the lower crust of Ohio. It is thought that these magnetic anomalies may represent the remains of an ancient rift zone or perhaps they are the signature of the Grenville Front complex which may cross through this area.

NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983. HINZE,W.J.; BRAILE,L.W. Purdue Univ., West Lafayette, IN. KELLER,G.R.; et al. Texas, Univ. of, El Paso, TX. April 1985. 60pp. 8504220438. 29946:263.

Recent geophysical investigations have shown that the seismicity of the New Madrid, Missouri, seismogenic region correlates with an ancient rift complex suggesting that the anomalous seismicity is the result of the localization of the regional compressive stress pattern by basement structures. An integrated geophysical/gelogical research program is being conducted to evaluate the rift complex hypothesis, to refine our knowledge of the structure and physical properties of the rift complex, and to investigate the possible northern extensions of the New Madrid Fault zone, especially the possible northeastern connection to the Anna, Ohio, seismic region. Investigation of the northeast extension has focused upon the acquisition and preparation of arrays of gravity and magnetic data sets. During 1983, special emphasis was placed upon integration of these data with basement lithologic and seismicity information which has revealed several major lithologic/structural features in the crust of the Anna area. Current seismicity in this region appears to be related to an ancient rift structure (the Fort Wayne rift) and possibly its contact with a low density pluton. Minor seismicity may be caused by stress concentration associated with local basement inhomogeneities.

NUREG/CR-3178: STRUCTURAL AND TECTONIC STUDIES IN NEW YORK STATE.Final Report,July 1981 - June 1982. ISACHSEN,Y.W. New York, State Univ. of, Albany, NY * Boston College, Chestnut Hill, MA. April 1985. 84pp. 8505100048. 30270:214.

Subjects treated in this report include the distribution, trends, exposure characteristics, aeromagnetic signatures, and detailed geometries of fracture systems, as well as tentative inferences concerning relative ages and causes of reactivation. Stress indicators are discussed, and a beginning is made at working out regional paleostress directions using the attitudes of dated mafic dikes. Attempts at defining Holecene and recent crustal movements using geological, geodetic, and seismological methods are reviewed, as well as attempts to relate projected focal mechanism solutions to ground geology. Finally, the distribution of earthquakes and their relationships to geology is reviewed.

NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT. GOTTULA,R.C.; CONDIE,K.G.; SUNDARUM,R.K.; et al. EG&G, Inc. April 1985. 562pp. 8504160110. EGG-2245. 29833:001.

Forced convective postcritical-heat-flux heat transfer experiments with water flowing upward in a vertical tube have been conducted at the Idaho National Engineering Laboratory. Thermodynamic nonequilibrium in the form of superheated vapor temperatures was measured at a maximum of three different axial levels. Steady-state experiments were conducted at pressures of 0.2 to 0.7 MPa, mass fluxes of 12 to 24 kg/m(2).s, heat fluxes of 7.7 to 27.5 kW/m(2), and test section inlet qualities of 38 to 64%. Quasi-steady-state (slow moving quench front) experiments were conducted at pressures of 0.4 to 7 MPa, mass fluxes of 12 to 70 kg/m(2).s, heat fluxes of 8 to 225 kW/m(2), and test section inlet qualities of -7 to 47%. The multiple probe data and the data taken above 0.4 MPa are new data in parameter ranges not previously obtained. Comparison of the data with current vapor generation models and wall heat transfer models yielded unsatisfactory results. This is attributed to the effects of nonequilibrium, quench front quality, and distance from the quench front, which are factors not included in the current models compared.

NUREG/CR-3197 V01: REACTION BETWEEN SOME CESIUM-IODINE COMPOUNDS AND THE REACTOR MATERIALS 304 STAINLESS STEEL,INCONEL 600 & SILVER. Volume I:Cesium Hydroxide Reactions. ELRICK,R.M.; SALLACH,R.A.; OUELLETTE,A.L.; et al. Sandia National Laboratories. June 1985. 156pp. 8507020369. SAND83-0395. 31307:175.

Laboratory scale scoping studies, using chemical simulants, are examining physical and chemical processes that could occur between fission products and other primary system materials in a steam and hydrogen environment. The chemical systems studied were cesium hydroxide vapor reactions in steam and hydrogen at 970K in a 304 Stainless steel system, at 1120K in a 304SS system, at 1000K with iodine vapor in an alumina system and at 1000K with hydrogen iodine vapor in an alumina system. Major observations and conclusions are that: cesium in the CsOH reacts with the silicon dioxide in the inner oxide formed on stainless steel to produce a cesium silicate; the availability of SiO(2) may therefore control the extent of reaction of CsOH with 304SS in steam; the oxidation rate of 304SS is enhanced by the exposure to CsOH vapor; the reaction of CsOH with Iconel 600 is slow in steam and seems to react with the silica content in the oxide layer.

NUREG/CR-3208: TRAC-PD2 DEVELOPMENTAL ASSESSMENT. KNIGHT,T.D.; METZGER,V. Los Alamos Scientific Laboratory. April 1985. 371pp. 8504160087. LA-9700-MS. 29835:001.

This report describes the final results of the development assessment analyses conducted during the later stages of the TRAC-PD2 development. The calculations discussed in this report used the released version of TRAC-PD2 and cover separate-effects blowdown, heat transfer, and downcomer penetration tests together with integral tests from the Loss-of-Fluid Test and Semiscale facilities. Although these calculations are not an exhaustive test of the code, they demonstrate its capabilities, including automatic steady-state initialization and the complete transient from blowdown through refill and reflood. The results show good agreement between the calculated parameters and the data and indicate that the code is applicable to large-break loss-of-coolant accident analyses.

NUREG/CR-3228 V03: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1984. LOSS,F.J. Materials Engineering Associates, Inc. June 1985. 171pp. 8506260518. MEA-2075. 31245:026.

This program consists of research and engineering relating to fracture, fatigue and radiation sensitivity of nuclear structural steels and weldments and addresses many of the key uncertainties in the margin of safety in operating nuclear plants. All tasks are integrated to focus on structural integrity of LWR pressure boundary components. The approach centers on an experimental characterization of nuclear grade steels and an assessment of fracture and environmental cracking behavior under conditions of a nuclear environment, so investigation of irradiated materials is a key element of each task. Emphasis is placed on identifying metallurgical factors responsible for radiation embrittlement of steels and on developing procedures for embrittlement relief, including guidelines for radiation-resistant steels. Experimental studies are supported by analytical models and investigations of the mechanisms responsible for the observed behavior. Data developed in the program will provide the basis for recommendations for the ASME Boiler and Pressure Vessel Code and ASTM test methods, and revisions to NRC Guides. Current work is organized into three major tasks: (1) fracture mechanics investigations, (2) environmentally-assisted crack growth in high temperature, primary reactor water and (3) radiation sensitivity and postirradiation properties recovery. Research progress in these three tasks for 1984 is summarized here.

NUREG/CR-3237: CONTROL OF EXPLOSIVE MIXTURES IN PWR WASTE GAS SYSTEMS. RANDOLPH,P.D.; ISAACSON,L.; AYERS,A.L.; et al. EG&G, Inc. January 30, 1985. 122pp. 8502130454. EGG-2251. 28914:224.

A study has been performed to evaluate problems associated with the existence of flammable or explosive gas mixtures in Pressurized Water Reactor waste gas systems. Information on existing waste gas systems, waste gas concentrations, and gas monitoring instrumentation obtained from six operating nuclear power plants is summarized. A comparative risk evaluation has been performed for several generic types and configurations of PWR waste gas systems. Waste gas systems in the plants visited are included and categorized as part of the risk evaluation. Existing data on the effect of initial pressure on flammability limits, as well as recently reported data on flammability and detonability of hydrogen/air mixture has been collected and summarized. A survey of commercially available instruments for monitoring hydrogen and oxygen concentrations has been performed and the results tabulated. A series of observations, conclusions and recommendations are given.

NUREG/CR-3293 V01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Main Report. ELDER,H.K. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 327pp. 8506140337. 30932:308.

Technical requirements, costs and safety are conceptually evaluated for the post-accident cleanup and decommissioning of fuel cycle and non-fuel cycle facilities that have experienced a significant accident. Accident cleanup is postulated to include 1) initial decontamination of building surfaces to reduce the subsequent occupational dose to cleanup and decommissioning workers and 2) management of the resulting wastes. Decommissioning is assumed to follow accident cleanup. In order to ensure that worker doses are ALARA, despite higher radiation exposure to workers during post-accident operations, careful planning and rehearsal of cleanup operations and the use of remote and semi-remote cleaning techniques are required to reduce occupancy times in high-radiation areas and to minimize occupational exposures during accident cleanup. The public safety impacts of post-accident cleanup and decommissioning are also evaluated; these are below permissible radiation dose levels in unrestricted areas and well within the range of annual radiation doses from normal background.

NUREG/CR-3293 V02: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Appendices. ELDER,H.K. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 288pp. 8506170550. 30979:001.

This volume contains the appendices concerning the technical requirements, costs and safety aspects conceptually evaluated for post-accident cleanup and decommissioning of fuel cycle and non-fuel cycle facilities that have experienced a significant accident. Accident cleanup is postulated to include 1) initial decontamination of building surfaces to reduce the subsequent occupational dose to cleanup and decommissioning workers and 2) management of the resulting wastes. Decommissioning is assumed to follow accident cleanup. In order to ensure that worker doses are ALARA, despite higher radiation exposure to workers during post-accident operations, careful planning and rehearsal of cleanup operations and the use of remote and semi-remote cleaning techniques are required to reduce occupancy times in high-radiation areas and to minimize occupational exposures during cleanup. The public safety impacts of post-accident cleanup and decommissioning are also evaluated; these are below permissible radiation dose levels in unrestricted areas and well within the range of annual radiation doses from normal background.

NUREG/CR-3301: CATALOG OF PRA DOMINANT ACCIDENT SEQUENCE INFORMATION. CATHEY,N.G.; KRANTZ,E.A.; POLOSKI,J.P.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). August 1985. 29pp. 8509130115. EGG-2259. 32605:254.

Information concerning the dominant accident sequences from twelve published probabilistic risk assessments (PRA) is cataloged in this report, which is published as a part of the Accident Sequence Evaluation Program (ASEP). The purpose of this report is to provide users of PRA information a single reference document. The cataloged results include plant operation information, core-melt and sequence frequencies, and a description of each dominant accident sequence. The report provides a consistent set of insights on the factors that drive the dominant accident sequences. ASEP has reconstructed the PRA fault tree models at the system or train level of detail and requantified the sequence likelihoods to provide the consistent insights. This work provides the information for the other ASEP activities on accident likelihood assessment for the operating and near-term operating plants.

NUREG/CR-3317: TECHNICAL BASES AND USER'S MANUAL FOR THE PROTOTYPE OF SPARC - A SUPPRESSION POOL AEROSOL REMOVAL CODE. OWCZARSKI,P.C.; POSTMA,A.K.; SCHRECK,R.I. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 68pp. 8506240650. PNL-4742. 31152:156.

The Pacific Northwest Laboratory has developed a prototype version of a Suppression Pool Aerosol Removal Code (SPARC). This code was written to calculate the capture of aerosol particles in the pressure suppression pool (wet well) of a boiling water reactor under hypothetical accident conditions. The code incorporates five aerosol scrubbing models and two thermal-hydraulic models. The scrubbing models describe 1) steam condensation, 2) soluble particle growth in a humid atmosphere, 3) gravitational settling, 4) inertial deposition, 5) diffusional deposition. Mechanical entrainment of pool liquid by breaking of bubbles at the surface was also considered. An optional model for equilibrium pool temperature and a model for steam evaporation are the two thermal-hydraulic models used in the code. Steam evaporation was found to significantly retard deposition processes in pools near the boiling point. The code user supplies the values of several controlling variables in the code input. The SPARC output can include the decontamination factors (DF) of twenty different particle size groups, an overall DF for the whole particle distribution, particle log normal distribution parameters, and mass flow rates of particles (wet and dry) leaving the pool.

NUREG/CR-3319: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium. MCELROY,W.N. Hanford Engineering Development Laboratory. August 1985. 533pp. 8509110278. HEDL-TME-85-3. 32563:020.

This NRC physics-dosimetry compendium (Sections 1.0 through 4.0) is a collation of information and data developed from available research and commercial light water reactor vessel surveillance program (RVSP) documents and related surveillance capsule reports. The Section 4.0 data represents the results of the HEDL least-squares FERRET-SAND II Code re-evaluation of exposure units and values for 47 PWR and BWR surveillance capsules. Using a consistent set of auxiliary data and dosimetry-adjusted reactor physics results, the revised fluence values for E > 1 MeV averaged 25% higher than the originally reported values. The range of fluence values (new/old) was from a low of 0.80 to a high of 2.38. These HEDL-derived FERRET-SAND II exposure parameter values are being used for NRC-supported HEDL and other PWR and BWR trend curve data development and testing studies, which support Revision 2 of Regulatory Guide 1.99. These trend curves are used by the utilities and by the NRC to account for neutron radiation damage in settng pressure/temperature limits, in analysing fractures, and in predicting neutron-induced changes in reactor PV steel fracture toughness and embrittlement during the vessel's service life. The status of the development and application of new advancements in LWR reactor surveillance programs is discussed, such as cavity physics-dosimetry for improving the reliability of current and end-of-life (EOL) predictions on the metallurgical conditions of pressure vessels and their support structures.

NUREG/CR-3361: THE EFFECT OF WATER CHEMISTRY ON THE RATES OF HYDROGEN GENERATION FROM GALVANIZED STEEL CORROSION AT POST-LOCA CONDITIONS. LOYOLA,V.M.; WOMELSDUFF,J.E. Sandia National Laboratories. January 1985. 40pp. 8502130366. SAND83-1326. 28920:328.

The rates of hydrogen generation are measured for the corrosion of galvanized steel in three different light water cooled reactor (LWR) water chemistries. The results were obtained over a temperature range of 100 degrees to 175 degrees centigrade and indicate that in a boiling water reactor (BWR) water chemistry, the reaction is faster than in those of two pressurized water reactors (PWR's). A mechanism is proposed which would explain the observed results without requiring that the chemical additives come in direct contact with the corrodible unoxidized metal. Such a mechanism is required because electron microprobe analysis suggests that no chemical additives have diffused into the protective ZnO layer which forms on the unoxidized metal. Arrhenius parameters are calculated for the three chemistries, but some questions are raised about whether those parameters are associated with a diffusion process or with the actual hydrogen producing reaction.

NUREG/CR-3413: OFF-SITE CONSEQUENCES OF RADIOLOGICAL ACCIDENTS:METHODS, COSTS AND SCHEDULES FOR DECONTAMINATION. TAWIL,J.J.; BOLD,F.C.; HARRER,B.J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1985. 379pp. 8509110274. PNL-4790. 32562:001.

This report documents a data base and a computer program for conducting a decontamination analysis of a large, radiologically contaminated area. The data base, which was compiled largely through interviews with knowledgeable persons both in the public and private sectors, consists of the costs, physical inputs, rates and contaminant removal efficiencies of a large number of decontamination procedures. The computer program utilizes this data base along with information specific to the contaminated site to provide detailed information that includes the least costly method for effectively decontaminating each surface at the site, various types of property losses associated with the contamination, the time at which each subarea within the site should be decontaminated to minimize these property losses, the quality of various types of labor and equipment necessary to complete the decontamination, dose to radiation workers, the costs for surveying and monitoring activities, and the disposal costs associated with radiological waste generated during cleanup. The program and data base are demonstrated with a decontamination analysis of a hypothetical site.

NUREG/CR-3426 V01: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY. Facility And Test Design Report. DOLAN,F.X.; VALENZUELA,J.A. Creare, Inc. September 1985. 130pp. 8510020227. EPRI NP-3802. 32839:028.

This report describes the test facility and program designed to measure fluid mixing and heat transfer in a 1/2-scale model of the cold-leg downcomer and lower plenum of a pressurized water reactor under conditions of interest to the issues of pressurized thermal shock. Several cold-leg assemblies are modeled and the downcomer arrangement can be altered to match vendor-specific configurations. The facility can be operated to model flow rates based on Froude number of the injected flow in the cold-leg and with steady or transient inlet boundary conditions. Extensive instrumentation is provided to measure flow rates, temperatures and pressure at the facility boundaries and for detailed measurements of temperatures, velocity and heat transfer data in the cold-leg and downcomer models. The test data are monitored and recorded by a computer data acquisition system that is also used for post-test reduction and plotting. The planned test matrix includes 75 tests with variations in cold-leg and downcomer geometries, loop and HPI flow rates, cold-leg Froude number and loop to HPI density difference. Test results will be reported in a series of Quick-Look Reports.

NUREG/CR-3426 V02: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY.Data Report. VALENZUELA,J.A.; DOLAN,F.X. Creare, Inc. September 1985. 208pp. 8510020217. EPRI NP-3802. 32838:069.

This report presents data from an experimental study of fluid mixing in a 1/2-scale model of the cold-leg, downcomer, lower plenum, pump simulator, and loop seal typical of a Westinghouse Pressurized Water Reactor. The tests were transient cooldown tests in that they simulated an extreme condition of Small-Break Loss-of-Coolant Accident (SBLOCA) during which cold High Pressure Injection (HPI) fluid is injected into stagnant, hot primary fluid with complete loss of natural circulation in the loop. Extensive temperature, velocity, and heat transfer coefficient data are presented at two cold-leg Froude numbers: 0.052 and 0.076. The 1/2-scale data are compared with earlier data from a 1/5-scale,geometrically similar facility to assess scaling principles.

NUREG/CR-3430 V02: NUCLEAR POWER PLANT OPERATING EXPERIENCE - 1982.Annual Report. SILVER,E.G. Oak Ridge National Laboratory. January 1985. 393pp. 8502150078. 29004:010.

This report is the ninth in a series of reports issued annually that summarizes the operating experience of nuclear power 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 1982 data from 72 plants; 24 boiling-water-reactor plants, 47 pressurized-water-reactor plants, and 1 high-temperature gas-cooled reactor plant.

NUREG/CR-3442: RADTWO:A COMPUTER CODE FOR SIMULATING FAST-TRANSIENT, TWO-DIMENSIONAL,TWO-LAYER RADIONUCLIDE CONCENTRATION CONDITIONS IN LAKES,RESERVOIRS,RIVERS,ESTUARIES,AND COASTAL REGIONS. ERASLAN,A.H.; DIAMENT,H. Oak Ridge National Laboratory. July 1985. 444pp. 8509180502. ORNL/TM-8869. 32666:013.

RADTWO is a computer code for predicting the transient, two-dimensional transport of radionuclides in receiving water bodies. The model formulation considers two coupled, depth-averaged transport equations for the water layer and the bottom sediment layer. The coupling conditions incorporate bottom deposition and resuspension effects. The computer code uses a discrete-element method which offers variable size grid-cells, accurate shoreline representation, and numerical accuracy. A sample application is provided for the problem of a hypothetical accidental release of radionuclides to the coastal environment. Results are presented as contours of constant radionuclide concentration in the water layer and the bottom sediment layer at various times during the model simulation period.

NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE. DAVIS,M.S.; PICIULO,P.L.; BOWERMAN,B.S.; et al. Brookhaven National Laboratory. July 1985. 102pp. 8507250157. BNL-NUREG-51699. 31794:123.

This report describes work conducted by BNL on the degradation of simulator chemical decontamination wastes by combustion and acid digestion. Both acid digestion and combustion are capable of effecting 90% destruction of the materials studied, as measured by the conversion of carbon compounds in the waste to carbon dioxide. Work on the direct solidification of simulated decontamination wastes in cement and vinyl ester-styrene is reported also. Laboratory scale waste forms were prepared using these binders. However, process control programs and full scale solidification studies are necessary to confirm the acceptability of the wastes.

NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA. HUCHTON,R.L.; TKACHYK,J.W.; TAYLOR,J.T.; et al. Westinghouse Electric Corp. April 1985. 80pp. 8505230585. WINCO-1024. 30546:254.

A comparison was made among various silver impregnated adsorption media to determine their iodine, krypton, and xenon retention efficiencies. The program consisted of three components. First, laboratory measurements of the noble gas retention efficiencies of commercially available adsorption media were determined as a function of relative humidity, sample duration, test cartridge geometry, and ambient air purge. Second, a literature survey was performed to evaluate the iodine species retention efficiencies of the selected media. Third, data associated with a media previously proposed for an emergency response air sampler were incorporated to enlarge the data base.

NUREG/CR-3469 V02: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS. Annotated Bibliography Of Selected Readings In Radiation Protection And ALARA. BAUM,J.W.; WEILANDICS,C. Brookhaven National Laboratory. June 1985. 150PP. 8507020380. BNL-NUREG-51708. 31306:251.

This is the second volume of abstracts dealing with occupational dose, dose control, dose reduction and application of the ALARA (as low as reasonably achievable) principle at nuclear power plants. This volume contains abstracts selected from APPLIED HEALTH PHYSICS ABSTRACTS AND NOTES, Volumes 1, No. 1, 1975 through Volume 5, No. 4, October 1979, and from recent publications known to the authors. Author and subject indexes are included. The subject index in this volume covers abstracts in both Volumes 1 and 2. This volume contains abstract Numbers 252 through 549.

NUREG/CR-3481 V02: NUCLEAR POWER PLANT PERSONNEL QUALIFICATIONS AND TRAINING: TAPS -- The Task Analysis Profiling System. JORGENSEN,C.C. Oak Ridge National Laboratory. July 1985. 246pp. 8508090705. ORNL/TM-9308/V2. 32104:128.

This report discusses an automated task analysis profiling system (TAPS) designed to provide a linking tool between the behaviors of nuclear power plant operators in performing their tasks and the measurement tools necessary to evaluate their in-plant performance. TAPS assists in the identification of the entry-level skill, knowledge, ability and attitude (SKAA) requirements for the various tasks and rapidly associates them with measurement tests and human factors principles. This report describes the development of TAPS and presents its first demonstration. It begins with characteristics of skilled human performance and proceeds to postulate a cognitive model to formally describe these characteristics. A method is derived for linking SKAA characteristics to measurement tests. The entire process is then automated in the form of a task analysis computer program. The development of the program is detailed and a user guide with annotated code listings and supporting test information is provided.

NUREG/CR-3485: PRA REVIEW MANUAL. EL-BASSIONI,A.; CHO,N.Z.; HANAN,N.; et al. Brookhaven National Laboratory. September 1985. 2pp. 8509230653. BNL-NUREG-51710. 32702:019.

This PRA Review Manual describes the approach for reviewing a Level 1 PRA, i.e., one which carries the accident analysis to the point of determination of core damage frequency, but excludes questions of containment integrity (but does include containment failure induced core damage) and of offsite consequences. The manual will be revised as comments are received, and as experience is gained from its use. The procedure involves three parts: The first (Phase 1) is concerned with the formal aspects of the PRA. Phase 1 surveys its apparent completeness, scrutability, and determines to what extent the PRA can usefully be further examined. It also identifies salient and distinctive features, of the study, methods, and reported results. The second part (Phase 2) reviews the analyses in a comprehensive and thorough but qualitative way, which is designed to focus on unusual or unsupported features, and to lay the groundwork for further, more detailed studies. The final stage (Phase 3) addresses details of issues and concerns raised in the earlier phases, and involves detailed quantitative examination of selected areas to ensure the overall validity. The first part of this manual, dealing with "internal" event PRAs, handles these phases sequentially as a whole. In the second part, which treats "external" events, the phases are identified within each event section, while Chapter 9 gives a sequential summary of the end results for each event.

NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations. START,G.E.; CATE,J.H.; SAGENDORF,J.F.; et al. Commerce, Dept. of, Natl. Oceanographic & Atmospheric Administration. February 1985. 75pp. 8503120084. 29340:116.

The 1981 Idaho Field Experiment was conducted in southeastern Idaho over the Upper Snake River Plain. Nine test-day case studies were measured between July 15 and 30, 1981. Eight-hour releases of SF(6) gaseous tracer were made from 46 m above the ground. Tracer was sampled hourly, for 12 sequential hours at about 100 locations within an area 24 km square. Also, a single total integrated sample of about 30 hours duration was collected at approximately 100 sites within an area 48 by 72 km (using 6 km spacings). Extensive tower profiles of meteorology at the release point were collected. RAWINSONDES, RABALS and PIBALS were collected at 3 to 5 sites. Horizontal, low-altitude winds were monitored using the INEL MESONET. SF(6) tracer plumes were marked with co-located oil fog releases and bi-hourly sequential launches of tetroon pairs. Aerial LIDAR observations of the oil fog plume and airborne samples of SF(6) were collected. High altitude aerial photographs of daytime plumes were also collected. The Idaho Field Experiment is reported in three volumes. Volume 3 contains descriptions of the nine intensive measurement days. General meteorological conditions are described, trajectories and their relationships to analyses of gaseous tracer data are discussed, and overviews of test day cases are presented. Calculations using the ARLFRD MESODIF model are included and related to the gaseous tracer data. Finally, a summary and list of recommendations are presented.

NUREG/CR-3498: TWO-DIMENSIONAL MODELING OF INTRA-SUBASSEMBLY HEAT TRANSFER AND BUOYANCY-INDUCED FLOW REDISTRIBUTION IN LMFBRS. KHATIB-RAHBAR; CAZZOLI,E.G. Brookhaven National Laboratory. January 1985. 179pp. 8501210102. BNL-NUREG-51713. 28496:154.

Phenomenological models and numerical techniques for prediction of coolant flow and temperature fields during forced, mixed, and free convection regimes of operation in LMFBR subassemblies are addressed. It is shown that, simplified integral solutions provide an excellent approach to assessing the importance of the intra-subassembly buoyancy induced flow redistribution, and the transverse thermal conduction and mixing effects on the assembly wide peak coolant temperatures. Furthermore, a more detailed steady-state and transient parabolic two-dimensional porous-body model, resulting in the TWIST computer code is developed. Comparison of calculated results and out-of-pile sodium and water test data indicate generally good agreement in cross-assembly temperature profiles. However, the impact of fuel pin distortion and bowing, caused by large transverse power gradients on transverse distributions are found to be significant.

NUREG/CR-3514 V02: THE CHEMICAL BEHAVIOR OF IODINE IN AQUEOUS SOLUTIONS UP TO 150 C.II.Radiation-Redox Conditions. TOTH,L.M.; DODSON,K.E. Oak Ridge National Laboratory. April 1985. 22pp. 8506100496. ORNL/TM-8664/V2. 30830:083.

Redox reactions that might alter the volatility of aqueous iodine solutions have been examined experimentally using absorption spectrophotometry. Oxygen and hydrogen atmospheres had no effect on the iodine chemistry at temperatures up to 150 degrees centigrade. However, irradiation of aqueous solutions with a (60)Co source, 0.8 x 10(6) R/h, produced radiolysis products that either oxdized iodine ion or reduced 10(3)- in the pH range 6-9 and generated significant amounts of volatile iodine. The amount of iodine volatilized varied from a few percent for solute concentrations of 10(-4) M to as much as 10 to 19% for 10(-6) M CsI or KI0(3) solutes. Silver metal has been shown to provide an effective gettering route for I- in solution if these ions are first oxidized by OH radicals generated during the radiolysis of the solutions.

NUREG/CR-3516: A SURVEY OF THE USES OF RADIOACTIVE MATERIALS IN LOUISIANA'S OFFSHORE WATERS. BENNETT,J.J.; HOOK,S.E.; PALAZZO,R.J.; et al. Louisiana, State of. February 1985. 37pp. 8503130140. 29360:122.

As a result of a contract agreement with the U.S. Nuclear Regulatory Commission, the State of Louisiana, and in particular, the Louisiana Nuclear Energy Division (LNED), conducted a survey of the use of radioactive materials in Louisiana's "offshore waters." Offshore waters are here defined as "that area of land and water on and above the United States' Outer Continental Shelf." The objectives of the survey were fourfold: 1) identification of those licensees using radioactive materials offshore Louisiana; 2) identification of work locations where radioactive materials are being used; 3) a description of the types of work performed; and 4) performance of at least three site visits to offshore locations where radioactive materials are being used. By telephone survey, LNED attempted to contact those licensees thought to be using radioactive materials offshore. Of the 69 licensees reached by telephone, 43, or 61%, indicated they have current offshore activities. The results of the telephone survey, conducted in May-June 1983 are presented in detail in this report. To meet objective four of the survey, three visits were made to offshore rigs and platforms, two involving industrial radiography and one involving well-logging. Also included in this report are summaries of these visits and a description of previous work done by LNED concerning radiation safety on "lay barges."

NUREG/CR-3519: HUMAN ERROR PROBABILITY ESTIMATION USING LICENSEE EVENT REPORTS. VOSKA,K.J.; O'BRIEN,J.N. Brookhaven National Laboratory. February 1985. 116pp. 8502220414. BNL-NUREG-51717. 29062:163.

The objective of this report is to present a method for using field data from nuclear power plants to estimate human error probabilities (HEPs). These HEPs are then used in probabilistic risk activities. This method of estimating HEPs is one of four being pursued in NRC-sponsored research. The other three are (1) structured expert judgment, (2) analysis of training simulator data, and (3) performance modeling. The type of field data analyzed in this report is from Licensee Event Reports (LERs) which are analyzed using a method specifically developed for that purpose. However, any type of field data or human errors could be analyzed using this method with minor adjustments. This report assesses the practicality, acceptability, and usefulness of estimating HEPs from LERs and comprehensively presents the method of use.

NUREG/CR-3537: EXPEDIENT METHODS OF RESPIRATORY PROTECTION:III. SUBMICRON PARTICLE TESTS AND SUMMARY OF QUALITY FACTORS. PRICE,J.M.; COOPER,D.W.; YEE,C.S.; et al. Sandia National Laboratories. September 1985. 95pp. 8509260257. SAND83-7450. 32757:118.

The efficacy of readily available materials, such as cotton fabrics, toweling a surgical mask, and a single-use respirator, for providing emergency respiratory protection was evaluated by determing the filtration efficiency as a function of aerosol particle size over the size range of 0.001 to 5.0 mm and as a function of filtration face velocity. Filtration face velocity was set at 1.5, 5.0, and 15.0 cm/s. This report describes the equipment and procedures used to obtain efficiency measurements for particles 0.5 mm in diameter and smaller, and summarizes the results of all three phases of this research. Particles with diameters from 0.10 to 0.50 mm proved to be the most difficult sizes of particles to remove. Particles smaller than 0.10 mm were removed due to diffusion while particles larger than 0.50 mm were removed due to inertia and gravitational settling. Deposition of the smallest particles was favored by the use of low face velocities. A fractional efficiency curve was determined for each material at each velocity for comparison. Values of the quality factor, [-1n (penetration)]/(pressure drop), were calculated. Quality factors were less for wet materials than for dry; less at high velocities rather than low; and best for the single-use respirator mask, next best for the surgical mask and often third best for the toweling.

NUREG/CR-3551: SAFETY IMPLICATIONS ASSOCIATED WITH IN-PLANT PRESSURIZED GAS STORAGE AND DISTRIBUTION SYSTEMS IN NUCLEAR POWER PLANTS. GUYMON,R.H.; CASTO,W.R.; COMPERE,E.L. Oak Ridge National Laboratory. May 1985. 82pp. 8506140622. ORNL/NOAC-214. 30934:031.

Storage and handling of compressed gases at nuclear power plants were studied to identify any potential safety hazards. Gases investigated were air, acetylene, carbon dioxide, chlorine, Halon, hydrogen, nitrogen, oxygen, propane, and sulfur hexafluoride. Physical properties of the gases were reviewed as were applicable industrial codes and standards. Incidents involving pressurized gases in general industry and in the nuclear industry were studied. In this report general hazards such as missiles from ruptures, rocketing of cylinders, pipe whipping, asphyxiation, and toxicity are discussed. Even though some serious injuries and deaths over the years have occurred in industries handling and using pressurized gases, the industrial codes, standards, practices, and procedures are very comprehensive. The most important safety consideration in handling gases is the serious enforcement of these well-known and established methods. Recommendations are made concerning compressed gas cylinder missiles, hydrogen line ruptures or leaks, and identification of lines and equipment.

NUREG/CR-3558: HANDBOOK OF NUCLEAR POWER PLANT SEISMIC FRAGILITIES. Seismic Safety Margins Research Program. COVER,L.E.; BOHN,M.P.; CAMPBELL,R.D.; et al. Lawrence Livermore National Laboratory. June 1985. 321pp. 8507080210. UCRL-53455. 31402:238.

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 and fully-coupled analysis procedure, including methods and computer codes, for estimating the risk of earthquake-induced radioactive release from a commercial nuclear power plant. As part of this program, calculations of the seismic risk from a typical commercial nuclear reactor were made. These calculations required a knowledge of the probability of failure (fragility) of safety-related components in the reactor system that actively participate in the hypothesized accident scenarios. This report describes the development of the required fragility relations and the data sources and data reduction techniques upon which they are based. Both building and component fragilities are covered. The building fragilities are for the Zion Unit 1 reactor, the specific plant used for development of methodology in the program. Some of the component fragilities are site-specific, but most would be usable for other sites as well.

NUREG/CR-3609: EVALUATION OF NEUTRON DOSIMETRY TECHNIQUES FOR WELL-LOGGING OPERATIONS. CUMMINGS,F.M.; HAGGARD,D.L.; ENDRES,G.W. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1985. 51pp. 8508010304. PNL-4942. 31928:170.

Neutron dose and energy spectral measurements from (241) AmBe and a 14 MeV neutron generator were performed at a well-logging laboratory. The measurement technique included the tissue equivalent proportional counter, multisphere, two types of remmeters and five types of personnel neutron dosimeters. Several source configurations were used to attempt to relate data to field situations. The results of the measurements indicated that the thermoluminescent albedo dosimeter was the most appropriate personnel neutron dosimeter, and that the most appropriate calibration source would be the source normally employed in the field with the calibration source being used in the unmoderated configuration.

NUREG/CR-3611: RADIOACTIVE MATERIAL (RAM) ACCIDENT/INCIDENT DATA ANALYSIS PROGRAM. EMERSON,E.L.; MCCLURE,J.D. Sandia National Laboratories. April 1985. 40pp. 8504220385. SAND82-2156. 29946:323.

This report describes the development of the Radioactive Materials Transportation Accident/Incident Data Base (RAM-AIDB), which contains information on the occurrences of transportation accidents and incidents, for radioactive materials (RAM) that are involved in the process of transportation, loading and unloading operations, or temporary storage. These transportation operations are in support of the nuclear fuel cycle for electrical energy generations of RAM. This study analyzes in some detail basic accident/incident statistical data, RAM packaging accident response data, and the health effects associated with RAM transport accidents/incidents. This report presents a summary of U.S. RAM transport accident/incident experience for the period 1971 through December 1981. In addition, a sample annual summary of accident/incident experience is presented for the calendar year 1981.

NUREG/CR-3613 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Annual Report For 1984. ATTERIDGE,D.G.; BRUEMMER,S.M.; PAGE,R.E. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 63pp. 8506270333. PNL-4971. 31262:215.

Pacific Northwest Laboratory (PNL), under a program sponsored by the Division of Engineering Technology of the U.S. Nuclear Regulatory Commission (NRC), is conducting a program to determine a method for evaluating the acceptance of welded and repair-welded stainless steel (SS) piping for light-water reactor (LWR) service. Validated models, based on experimental data, will be developed to predict the degree of sensitization (DOS) and the intergranular stress corrosion cracking (IGSCC) susceptibility in the heat affected zone (HAZ) of the SS weldments. IGSCC is caused by a combination of a sensitized microstructure, an aggressive environment, and tensile stress. Control of any of these three factors can eliminate IGSCC in most practical situations. This program will measure and model the development of a sensitized microstructure as it pertains to welded and repair-welded SS pipe. An empirical correlation between a material's DOS and its susceptibility to IGSCC will be determined using constant extension rate tests (CERTs). The successful completion of these tasks will result in a method for assessing the effects of welding/repairing parameters on the IGSCC susceptibility of component-specific nuclear reactor welds/repairs.

NUREG/CR-3613 V03 N1: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Semiannual Report For October 1984 Through March 1985. ATTERIDGE,D.G.; CHARLOT,L.A.; BRUEMMER,S.M.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1985. 59pp. 8510040360. PNL-4941. 32856:184.

Pacific Northwest Laboratory, under the sponsorship of the Division of Engineering Technology of the U.S. Nuclear Regulatory Commission, is conducting a program to determine a method for evaluating welded and repair-welded stainless steel (SS) piping for light-water reactor service. Validated models, based on experimental data, are being developed to predict microstructural development (e.g., the degree of sensitization) and the stress-corrosion cracking (SCC) resistance in the heat-affected zone of the SS weldments. Stress-corrosion cracking is caused by a combination of a susceptible microstructure, an aggressive environment, and tensile stress. Control of any of these three factors can eliminate SCC in most practical situations. This program will measure and model the development of a susceptible microstructure as it pertains to welded and repair-welded SS pipe. Empirical correlations between material microstructure and SCC will be determined using constant extension rate tests. The successful completion of these tasks will result in a method for assessing the effects of welding/repairing parameters on the SCC resistance of component-specific nuclear reactor welds/repairs. The present report describes the progress of these studies during the first half of the 1985 fiscal year.

NUREG/CR-3626 V02: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: DESCRIPTION OF MODEL CONTENT,STRUCTURE,AND SENSITIVITY TESTING. SIEGEL,A.I.; BARTTER,W.D.; WOLF,J.J.; et al. Oak Ridge National Laboratory. April 1985. 322pp. 8504170234. ORNL/TM-9041/V2. 29902:002.

This volume of NUREG/CR-3626 presents details of the content, structure, and sensitivity testing of the Maintenance Personnel Performance Simulation (MAPPS) model that was described in summary in volume one of this report. The MAPPS model is a generalized stochastic computer simulation model developed to simulate the performance of maintenance personnel in nuclear power plants. The MAPPS model considers workplace, maintenance technician, motivation, human factors, and task oriented variables to yield predictive information about the effects of these variables on successful maintenace task performance. All major model variables are discussed in detail and their implementation and interactive effects are outlined. The model was examined for disqualifying defects from a number of viewpoints, including sensitivity testing. This examination led to the identification of some minor recalibration efforts which were carried out. These positive results indicate that MAPPS is ready for initial and controlled applications which are in conformity with its purposes.

NUREG/CR-3633 V01 S1: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS. TAYLOR,D.D.; SHUMWAY,R.W.; SINGER,G.L.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). September 1985. 122pp. 8510040411. EGG-2294. 32855:116.

The TRAC-BD1/MOD1 computer program provides a best-estimate analysis capability for the analysis of the full range of postulated accidents in Boiling Water Reactor (BWR) systems and related experimental facilities. The program is described in four volumes: Volume 1, Code Description; Volume 2, User's Guide; Volume 3, Code Structure and Programming Information; and Volume 4, Developmental Assessment. Volume 1 describes the thermal-hydraulic models, numerical methods, and component models available. Volume 2 describes the input and output of the TRAC-BD1/MOD1 code and provides guidelines for use of the code modeling of BWR systems. Volume 3 is designed for the programmer or model developer who needs to implement or modify the TRAC-BD1/MOD1 program. Volume 4 discusses the results of the development assessment calculations performed with TRAC-BD1/MOD1.

NUREG/CR-3633 V04: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.Volume 4: Developmental Assessment. SHUMWAY,R.W. EG&G Idaho, Inc. (subs. of EG&G, Inc.). September 1985. 101pp. 8510040407. EGG-2294. 32858:314.

This volume of the TRAC-BD1/MOD1 manual discusses the results of developmental assessment calculations performed mainly with a preliminary version of TRAC-BD1/MOD1 (V21), and some selected cases performed with the official version of TRAC-BD1/MOD1 (V22), which differed from the preliminary version due to few small corrections. Twenty one test cases have been performed, ranging from simple single-effect flow tests up to a full BWR/6 system calculation. The four groups of tests are: separate effects hydraulic tests, steady state heat transfer tests, transient heat transfer tests, and integral system effects tests. The separate effects test cases were each chosen to exercise a specific hydraulic or heat transfer model in the code, while the integral system effects tests were chosen to exercise the code as a whole. The TRAC code version initially used was TA021 for all cases. Code errors were evident in some of the heat transfer runs.

NUREG/CR-3634: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: USER'S MANUAL. KOPSTEIN,F.F.; WOLF,J.J. Oak Ridge National Laboratory. September 1985. 115pp. 8512270273. ORNL/TM-9545. 34079:172.

This document serves as the user's manual for the MAintenance Personnel Performance Simulation (MAPPS) model. MAPPS is a generalized, stochastic computer simulation model which simulates the performance of maintenance personnel in nuclear power plants. The model considers workplace, maintenance technician, motivational, human factors, and task-oriented variables to yield predictive information about the effects of these variables on successful maintenance task performance. As such, MAPPS is believed to represent a fundamental probabilistic risk assessment (PRA) analytic tool. Moreover, it serves the needs of nuclear power plant maintenance managment for maintenance operations analysis, and the needs of architectural and engineering firms for maintenance system design evaluation. This manual deals with the procedural aspects of operating the MAPPS computer program. The first section of the present report describes the use of the MAPPS model within the nuclear power plant context. This section also presents the principal potential uses of the model and the major types of information that can be helpful to its various types of users in making decisions. Section 2 provides an overview of MAPPS utilization and explains the process of planning and executing simualtions runs. Data input requirements are outlined, and the various data types are explained. Also, the model's data outputs are illustrated. Section 3 provides the detailed guidance to users for interacting with MAPPS via a terminal.

NUREG/CR-3638: HYDROGEN-STEAM JET-FLAME FACILITY AND EXPERIMENTS. SHEPHERD,J.E. Sandia National Laboratories. July 1985. 138pp. 8508010764. SAND84-0060. 31925:104.

As part of NRC-sponsored research on light-water reactor safety, the high-temperature combustion of steam-hydrogen jets in an air atmosphere is being investigated at Sandia. This research is oriented at understanding the generic issues involved in accident-generated jets and the specific problems of using deliberate flaring from high-point vents to eliminate hydrogen from the primary system. In this report we give some background on diffusion-flame combustion, describe the experimental facility constructed at Sandia to study high-temperature, steam-hydrogen jets and discuss our results.

NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT. KNIGHT,T.D.; BOOKER,C.P.; BOYACK,B.E.; et al. Los Alamos Scientific Laboratory. October 1985. 229pp. 8512050451. LA-10548-MS. 33770:246.

The Transient Reactor Analysis Code (TRAC) provides an advanced, best-estimate analysis capability for pressurized water reactors and for many thermal-hydraulic test facilities. The most recent publicly released version of TRAC is TRAC-PF1. This code version includes a full two-fluid modeling capability in both the three-dimensional vessel component and the one-dimensional components. We have improved the numerical methods in the one-dimensional components to provide a more stable solution and to permit the code to run faster. The Los Alamos report, "TRAC-PF1: An Advanced Best-Estimate Computer Program for Pressurized Water Reactor Analysis," LA-9944-MS (NUREG/CR-3567), provides a detailed description of the code. This report documents the Los Alamos results of the second assessment phase, independent assessment for TRAC-PF1. We analyzed separate-effects tests in the Semiscale facility to investigate natural-circulation and reflux cooling. We analyzed integral tests from the Semiscale and the Loss-of-Fluid Test facilities to explore the small- and intermediate-break loss-of-coolant accident (LOCA) capability and the non-LOCA capability. We also analyzed the loss-of-feedwater transient in the Crystal River plant. The results show reasonably good agreement with the data, but indicate that improvements are required for the critical-flow model and the interphasic-condensation model.

NUREG/CR-3647: DESIGN AND FABRICATION OF A 1/8-SCALE STEEL CONTAINMENT MODEL. REESE,R.T.; HORSCHEL,D.S. Sandia National Laboratories. April 1985. 131pp. 8504170693. SAND84-0048. 29907:022.

A 1/8-scale steel model containment building was designed and fabricated in support of the Containment Safety Margins Program. This program is directed to determine the margin of safety of containments in severe accident conditions. It is planned to internally pressurize the model to failure. In this testing program, failure modes of the pressure vessel and sealed penetrations will be examined in detail. The model was designed according to Section III of ASME Code for Class MC containment vessels with the exception that no code stamp was required since no nuclear materials would be housed within the model. All the general requirements (subsection NCA) and specific requirements (subsection NE) of Section III of the ASME Code were met. The majority of the model was fabricated from 3/16-in. SA516 Grade 70 steel plate in the form of a right circular cylinder capped with a hemispherical dome. Eleven penetrations and two lifting trunnions were included in the model. The cylinder/dome section was joinded to a 2:1 ellipsoidal base (test fixture) composed of thicker (1 1/8-in. and 1 1/2-in.) plate material. The model was supported on six legs to permit access for personnel, instrumentation, data acquisition, power, and pressure piping. The model was fabricated in April through October 1983 by Chicago Bridge and Iron and erected at the test site in Albuquerque, New Mexico, in November 1983.

NUREG/CR-3651: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES. HARDY,J.E.; HERSKOVITZ,M.B. Oak Ridge National Laboratory. April 1985. 56pp. 8506070366. ORNL/TM-9067. 30798:265.

Instrumentation for making two-phase measurements in experimental refill-reflood test facilities was developed by Oak Ridge National Laboratory (ORNL) through the Advanced Instrumentation for Reflood Studies (AIRS) program. These unique instrumentation systems were designed to survive the severe in-vessel environmental conditions that exist during a simulated pressurized water reactor loss-of-coolant accident (LOCA). The measurements include two-phase flow velocity, void fraction, and film thickness and velocity, and are required for better understanding of reactor behavior during LOCAs. The adequacy (survivability and data quality) of the instrumentation systems installed in four experimental reflood test facilities is assessed. Signal conditioning electronics and sensor thermocouples functioned extremely well. For the first time, two-phase flow measurements were made in-core during a simulated LOCA. Because of the harsh environment and geometrical constraints, some sensor failures were considered likely; the number actually failing in service was within expectations. An exception to this record occurred in the Slab Core Test Facility -- Core 1. A chloride-ion stress corrosion problem destroyed signal cables at the vessel seal for most sensors. This problem was corrected by changing the sealant material at the vessel penetration in the subsequent facilities. Overall, the performance of the instrumentation was very satisfactory yielding valuable data during simulated LOCAs in refill-reflood test facilities.

NUREG/CR-3657: PRELIMINARY SCREENING OF FUEL CYCLE AND BY-PRODUCT MATERIAL LICENSES FOR EMERGENCY PLANNING. BENNETT,D.E.; RUNKLE,G.E.; ALPERT,D.J.; et al. Sandia National Laboratories. April 1985. 137pp. 8506060385. SAND84-0186. 30775:062.

This report summarizes work done for the U.S. Nuclear Regulatory Commission as part of a program considering the need for and appropriate level of emergency response planning at fuel cycle and by-product material facilities. The purpose is to (1) provide a base of technical information for identifying and ranking those facilities for which the need for emergency response planning and preparedness should be further considered, and (2) perform an initial screening of licenses issued by NRC. A data base containing the radionuclide possession limits for each license was developed. Dose estimates for a unit (1 curie) release of each of the radionuclides in the data base were calculated. To account for the variability in weather, distributions of doses were estimated for a full range of meteorological conditions. As requested by NRC, doses at the 99th percentile of the distribution were used. An initial screening analysis was performed for the approximately 9400 licenses by comparing the estimated 99th percentile dose for a postulated release of a fraction of the licensed possession limit to the dose levels suggested in the Environmental Protection Agency's Protective Action Guides. Using relatively conservative assumptions in the screening analysis, all but at most a few hundred licenses were found to have estimated doses below the Protective Action Guide levels. The few hundred identified in this initial screening should be further evaluated using realistic assumptions and site-specific information to establish the need for, appropriate level and extent of, and potential effectiveness of emergency response planning and preparedness beyond that currently required.

NUREG/CR-3659: A MATHEMATICAL MODEL FOR ASSESSING THE UNCERTAINTIES OF INSTRUMENTATION MEASUREMENTS FOR POWER AND FLOW OF PWR REACTORS. HESSON,G.M.; CLIFF,W.C.; STEVENS,D.L. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1985. 48pp. 8503220007. PNL-4973. 29487:217.

A method of assessing the quantitative uncertainties in the determination of PWR powers and coolant flows caused by measurement uncertainties is provided. The method defines the parameters entering into the calculation, the types and sources of measurement errors which must be considered, together with sources of quantitative data for the uncertainties. A mathematical model is developed which combines the measurement uncertainties in a rigorous statistical manner to give the overall uncertainty in the desired parameter together with a sample calculation.

NUREG/CR-3660 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 1:Summary Report. HOLMAN,G.S.; CHOU,C.K. Lawrence Livermore National Laboratory. July 1985. 103pp. 8508010773. UCID-19988. 31923:247.

As part of its reevaluation of the double-ended guillotine break (DEGB) of reactor coolant loop piping as a design basis event for nuclear power plants, the U.S. Nuclear Regulatory Commission (NRC) contracted with the Lawrence Livermore National Laboratory (LLNL) to estimate the probability of occurrence of a DEGB, and to assess the effect that earthquakes have on DEGB probability. This report describes a probabilistic evaluation of reactor coolant loop piping in PWR plants having nuclear steam supply systems designed by Westinghouse. Two causes of pipe break were considered: pipe fracture due to the growth of cracks at welded joints ("direct" DEGB), and pipe rupture indirectly caused by failure of component supports due to an earthquake ("indirect" DEGB). The probability of direct DEGB was estimated using a probabilistic fracture mechanics model. The probability of indirect DEGB was estimated by estimating support fragility and then convolving fragility and seismic hazard. The results of this study indicate that the probability of a DEGB from either cause is very low for reactor coolant loop piping in these plants, and that NRC should therefore consider eliminating DEGB as a design basis event in favor of more realistic criteria.

NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes. RAVINDRA,M.K.; CAMPBELL,R.D.; KENNEDY,R.P.; et al. Lawrence Livermore National Laboratory. February 1985. 199pp. 8503130008. UCID-19988. 29360:183.

The requirements to design the nuclear power plants for the effects of an instantaneous double-ended guillotine break (DEGB) of the reactor coolant loop (RCL) piping have led to excessive design costs, interference of normal plant operation and maintenance, and unnecessary radiation exposure of plant maintenance personnel. This report describes an aspect of the NRC/Lawrence Livermore National Laboratory sponsored research program aimed at demonstrating that the probability of DEGB in RCL piping of nuclear power plants is acceptably small and the requirements to design for the DEGB effects (e.g., provision of pipe whip restraints) may be removed. This study estimated the probability of indirect DEGB in RCL piping as consequence of seismic-induced structural failures within the containment of Westinghouse supplied pressurized water reactor nuclear power plants in the United States. The median probability of indirect DEGB was estimated to be about 3x10(-6) per year with a 10% to 90% subjective probability range approximately for 1x10(-7) per year to 4x10(-5) per year.

NUREG/CR-3660 V04: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 4:Pipe Failure Induced By Crack Growth In West Coast Plants. CHINN,D.J.; HOLMAN,G.S.; LO,T.Y.; et al. Lawrence Livermore National Laboratory. July 1985. 58pp. 8508020207. UCID-19988 V04. 31962:143.

The U.S. Nuclear Regulatory Commission contracted with the Lawrence Livermore National Laboratory to conduct a study to determine if the probability of occurrence of a double-ended guillotine break in primary coolant piping warrants the current design requirements that safeguard against the effects of such a break. This report assesses the reactor-coolant-loop piping system of west coast Westinghouse plants. The results indicate that directly induced DEGB is an unlikely event in the west coast Westinghouse plants. For the Trojan plant, leak is far more likely than a direct DEGB. Further, earthquakes have very little effect on the probabilities of leak and direct DEGB. At the Diablo Canyon plant, the increase in postulated seismic levels due to reevaluation of the site to account for the Hosgri Fault has caused directly induced DEGB failure probability to be dependent on earthquake occurrences. The resulting direct DEGB failure probability is still much lower than the indirect DEGB failure probability for Diablo Canyon.

NUREG/CR-3663 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 1:Summary Report. HOLMAN,G.S.; LO,T.Y.; CHOU,C.K. Lawrence Livermore National Laboratory. January 1985. 81pp. 8502120055. UCRL-53500 V01. 28972:057.

As part of its reevaluation of the double-ended guillotine break (DEGB) as a design requirement for reactor coolant piping, the U.S. Nuclear Regulatory Commission (NRC) contracted with the Lawrence Livermore National Laboratory (LLNL) to estimate the probability of occurrence of a DEGB, and to assess the effect that earthquakes have on DEGB probability. This report describes a probabilistic evaluation of reactor coolant loop piping in PWR plants having nuclear steam supply systems designed by Combustion Engineering. Two causes of pipe break were considered: pipe fracture due to the growth of cracks at welded joints ("direct" DEGB), and pipe rupture indirectly caused by failure of component supports due to an earthquake ("indirect" DEGB). The probability of direct DEGB was estimated using a probabilistic fracture mechanics model. The probability of indirect DEGB was estimated by estimating support fragility and then convolving fragility with seismic hazard. The results of this study indicate that the probability of a DEGB from either cause is very low for reactor coolant loop piping in these plants, and that NRC should therefore consider eliminating DEGB as a design basis in favor of more realistic criteria.

NUREG/CR-3663 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 3:Double Ended Guillotine Break Indirectly Induced By Earthquakes. RAVINDRA,M.K.; CAMPBELL,R.D.; KENNEDY,R.P.; et al. Lawrence Livermore National Laboratory. January 1985. 118pp. 8501280412. 28573:001.

The requirements to design the nuclear power plants for the effects of an instantaneous double-ended guillotine break (DEGB) of the reactor coolant loop (RCL) piping have led to excessive design costs, interference of normal plant operation and maintenance personnel. This report describes an aspect of the NRC/Lawrence Livermore National Laboratory sponsored research program aimed at demonstrating that the probability of DEGB in RCL piping of nuclear power plants is acceptably small and the requirements to design for the DEGB effects (e.g., provision of pipe whip restraints) may be removed. This study estimated the probability of indirect DEGB in RCL piping as a consequence of seismic-induced structural failures within the containment of Combustion Engineering supplied pressurized water reactor nuclear power plants in the United States. The median probability of indirect DEGB was estimated to be in the range of 10(-6) per year of older plants, and less than 10(-8) per year for modern plants; using very conservative assumptions, the 90% subjective probability value (confidence) of P DEGB was found to be less than 5x10(-5) per year for older plants and less than 3x10(-7) per year for modern plants.

NUREG/CR-3688 V01: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 1:Main Report. COMER,M.K.; SEAVER,D.A.; STILLWELL,W.G.; et al. General Physics Corp. January 1985. 61pp. 8502210260. SAND84-7115. 29028:272.

The U.S. Nuclear Regulatory Commission is conducting a research program to determine the practicality, acceptability, and usefulness of several different methods for obtaining human reliability data and estimates that can be used in nuclear power plant probabilistic risk assessments (PRA). One method, investigated as part of this overall research program, uses expert judgment to generate human error probability (HEP) estimates and associated uncertainty bounds. The project described in this document evaluated two techniques for using expert judgment: paired comparisons and direct numerical estimation. Volume 1 of this report provides a brief overview of the background of the project, the procedure for using psychological scaling techniques to generate HEP estimates and conclusions from evaluation of the techniques. Volume 2 provides detailed procedures for using the techniques, detailed descriptions of the analyses performed to evaluate the techniques, and HEP estimates generated as part of this project. The results of the evaluation indicate that techniques using expert judgment should be given strong consideration for use in developing HEP estimates. In addition, HEP estimates for 35 tasks related to boiling water reactors (BWRs) were obtained as part of the evaluation. These HEP estimates are also included in the report.

NUREG/CR-3688 V02: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 2:Appendices. COMER,M.K.; SEAVER,D.A.; STILLWELL,W.G.; et al. General Physics Corp. January 1985. 176pp. 8502210262. SAND84-7115. 29028:095.

See NUREG/CR-3688,V01 abstract.

NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY. ROHATGI,U.S.; PU,J.; SAHA,P.; et al. Brookhaven National Laboratory. April 1985. 98pp. 8505070497. BNL-NUREG-51750. 30211:005.

Several Oconee-1 overcooling transients that were computed by LANL and INEL using the latest versions of TRAC-PF1 and RELAPS/MOD1.5 codes have been reviewed by BNL. Three of these transients were selectd for detailed review as they either had the potential of challenging the integrity of the pressure vessel or highlighted the effect of code differences. These are (1) Main Steam Line Break (MSLB), (2) All Turbine Bypass Valves Stuck Open, and (3) 2-Inch Small Break LOCA.

NUREG/CR-3706: TRAC ANALYSES OF SEVERE OVERCOOLING TRANSIENTS FOR THE OCONEE-1 PWR. IRELAND,J.R. Los Alamos Scientific Laboratory. August 1985. 261pp. 8509160026. LA-10055-MS. 32625:059.

This report describes the results of several Transient Reactor Analysis Code (TRAC)-PF1 calculations of overcooling transients in a Babcock & Wilcox lowered-loop, pressurized water reactor (Oconee-1). The purpose of this study is to provide detailed input on thermal-hydraulic data to Oak Ridge National Laboratory for pressurized thermal-shock analyses. The transient calculations performed were plant specific in that details of the primary system, the secondary system, and the plant-integrated control system of Oconee-1 were included in the TRAC input model. The results of the calculations indicate that the turbine-bypass valve failure transient was the most severe in terms of resulting in relatively cold liquid temperatures in the downcomer region of the vessel. The power-operated relief valve loss-of-coolant accident transient was the least severe in terms of downcomer liquid temperatures because of vent-valve fluid mixing and near-saturated conditions in the primary system. It is recommended that future calculations consider a wider range of operator actions to cover the spectra of overcooling transient sequences more completely.

NUREG/CR-3709: METHODS OF MINIMIZING GROUND-WATER CONTAMINATION FROM IN SITU LEACH URANIUM MINING.Final Report. DEUTSCH,W.J.; MARTIN,W.J.; EARY,L.E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1985. 158pp. 8504040003. PNL-5319. 29629:075.

This is the final report of a research project dealing with methods of minimizing ground-water contamination from in situ leach uranium mining. Field work and laboratory experiments were conducted to identify excursion indicators for monitoring purposes during mining, and to evaluate effective aquifer restoration techniques following mining. Many of the solution constituents were found to be too reactive with the aquifer sediments to reliably indicate excursion of leaching solution from the ore zone; however, in many cases, the concentrations of chloride and sulfate and the total dissolved solids level of the solution were found to be good excursion indicators. Aquifer restoration by ground-water sweeping consumed ground water and was not effective for the redox-sensitive contaminants often present in the ore zone. Surface treatment methods were effective in lowering the amount of water used, but also had the potential for creating conditions in the aquifer under which the redox-sensitive contaminants would be mobile. In situ restoration by chemical reduction, in which a reducing agent is added to the solution recirculated through the ore zone during restoration, has the capability of restoring the ore zone sediment as well as the ground water. This method could lead to a stable chemical condition in the aquifer similar to conditions before mining.

NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT. SEITZ,M.G.; BOWERS,D.L.; GERDING,T.J.; et al. Argonne National Laboratory. August 1985. 150pp. 8508260304. ANL-84-16. 32368:071.

Experiments are described that combine backfill, radioactive waste, and basalt rock in a single flowing groundwater stream in a manner analogous to a hydraulic breach of a waste repository. The experiments were used to study chemical interactions that would occur if repository components were breached by flowing water. The result of most significance to issues of repository performance was that uranium, neptunium, and plutonium were found to move more rapidly through repository components that were altered to represent aging than through fresh materials. In contrast, cesium moved slower through altered repository materials, as had been deduced from previous work using batch adsorption tests. Two other parameters studied experimentally, the metal alloy used in the apparatus and an ionizing radiation field imposed on the experimental apparatus, had little or no measurable effect on radioactive element transport by flowing water. Inasmuch as the alteration of the repository materials aging in an actual repository, we conclude that changes with age will detrimentally affect the ability of a repository to isolate uranium, neptunium, and plutonium. Because these elements have long-lived radioactive isotopes in nuclear waste, the degradation with time is a major issue regarding the performance of a nuclear waste repository in basalt.

NUREG/CR-3721 V01: PRESSURE MEASUREMENTS IN A HYDROGEN COMBUSTION ENVIRONMENT. Hydrogen-Air Combustion Test Series 1 And 2 In The FITS Tank. ROLLER,S.F. Sandia National Laboratories. April 1985. 59pp. 8504170005. SAND83-2621/1. 29904:307.

Hydrogen combustion tests were performed in the Fully Instrumented Test Site (FITS) tank under the Hydrogen Behavior Program performed by Sandia National Laboratories under contract with the US Nuclear Regulatory Commission. Test series 1 and 2 examined the effects of a number of parameters on hydrogen-air combustion: the initial temperature and pressure of the gases, the effect of added steam or carbon dioxide as diluents, and the percent hydrogen in air. For tests in the range of 20% to 40% hydrogen in air, recorded peak pressures were equal to adiabatic, isochroic, complete combustion (AICC) values within an experimental error of 15%. This was contrary to the results of tests at a number of other facilities. The pre-ignition temperature had a strong effect on the peak pressure, while pre-ignition pressure in the range examined had no effect on combustion pressure ratios. Calculations showed that, although the effect of dynamic head on the peak pressure was a few percent or less, interactions of the wave preceding the flame front with the flame and with the vessel walls may be apparent in the experimental records.

NUREG/CR-3723: STRESS-INTENSITY-FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS. BALL,D.G.; BASS,B.R.; BRYSON,J.W.; et al. Oak Ridge National Laboratory. February 1985. 53pp. 8503290280. ORNL/CSD/TM-216. 29563:295.

In the fracture-mechanics analysis of reactor pressure vessels, stress-intensity-factor influence coefficients are used in conjunction with superposition techniques to reduce the cost of calculating stress-intensity factors. The present study uses a finite-element code, together with a virtual crack extension technique, to obtain influence coefficients for semielliptical surface flaws in a cylinder, and particular emphasis was placed on mesh convergence (less than 1% error was sought in the results from any one mesh construction parameter). Comparison of the coefficients with those obtained by other investigators shows good agreement. Furthermore, stress-intensity factors obtained by superposition for a severe thermal-transient loading condition agree within 1% of the values calculated by a direct finite-element method. Influence coefficients were calculated for three specific axially oriented semielliptical surface flaws. The first was a 2-m-long inner-surface flaw in a nuclear reactor pressure vessel with depth-to-wall-thickness ratios between 0.2 and 0.9. The second was an inner-surface flaw in the reactor vessel with a surface-length-to-depth ratio of 6 and with depth-to-wall-thickness ratios between 0.05 and 0.2. The third was a 1-m-long flaw on the outer surface of a test vessel with depth-to-wall-thickness ratios between 0.1 and 0.9. For the reactor vessel, separate coefficients were calculated for the cladding on the inner surface and for the base-material region. This allows for an accurate accounting of the effect of thermal stresses in the cladding on the stress-intensity factor for surface flaws that extend through the cladding into the base material.

NUREG/CR-3736: FIELD AND THEORETICAL INVESTIGATIONS OF FRACTURED CRYSTALLINE ROCK NEAR ORACLE,ARIZONA. JONES,J.W.; SIMPSON,E.S.; NEUMAN,S.P.; et al. Arizona, Univ. of, Tucson, AZ. August 1985. 115pp. 8508290526. 32410:254.

A combination of geophysical and hydraulic testing has been conducted in granite near Oracle, Arizona. The purpose of the work is to determine relationships, if any, among (1) fracture distribution, (2) geophysical properties, and (3) hydraulic properties of fractured rock of low hydraulic conductivity. To date, eight vertical borings spaced 20 to 50 feet apart, ranging from 250 to 300 feet in depth, have been drilled. The data obtained from neutron, gamma, acoustic-velocity, electrical-resistivity, and acoustic-televiewer logs, with the results of over 100 single-hole, straddle-packer injection tests make possible a detailed description of the fracture system. Geophysical logs readily detect fractures and are sensitive to subtle lithologic variations of the granite. Orientation and distribution of individual fractures were determined from the interpretation of the acoustic-televiewer data, and from the analysis of core obtained from one borehole. Fracture densities over the 13-foot long straddle-packer test intervals did not correlate with measured hydraulic conductivity measurements. A strong correlation between the neutron-log response and measured hydraulic conductivity does exist; it was used to supplant conductivity measurements. The geostatistical technique of kriging provided a three-dimensional map of hydraulic conductivity that can be compared with subsurface interpretations of the geophysical logs.

NUREG/CR-3738: ENVIRONMENTAL EFFECTS OF THE URANIUM FUEL CYCLE.A Review Of Data For Technetium. TILL,J.E. Radiological Assessments Corp. SHOR,R.W.; HOFFMAN,F.O. Oak Ridge National Laboratory. February 1985. 135pp. 8503120452. ORNL/TM-9150. 29340:293.

Sources of (99)Tc releases to the environment are reviewed for the uranium fuel cycle considering the recycle of spent uranium fuel and no fuel recycling. Without recycling the only source of (99)Tc release is an extremely small amount associated with airborne emissions from the processing of high-level wastes. With recycling, (99)Tc releases are associated with the operation of reprocessing facilities, UF(6) conversion plants, uranium enrichment plants, fuel fabrication facilities, and low- and high-level waste processing and storage facilities. Among these, the most prominent (99)Tc releases are from the liquid effluents of uranium enrichment facilities. An extensive review of data estimate parameters for predicting the environmental behavior and fate of (99)Tc indicates a reduced radiological significance for the ingestion of milk and meat. More important pathways of exposure to (99)Tc will probably be associated with drinking water and the consumption of aquatic organisms, garden vegetables, and eggs. For each parameter reviewed in this study, a range of values is recommended for radiological assessment calculations. Where obvious discrepancies exist between these range and the default values listed in USNRC Regulatory Guide 1.109, consideration for revision of the USNRC default values is recommended.

NUREG/CR-3741 V02: EVALUATION OF POWER REACTOR FUEL ROD ANALYSIS CAPABILITIES. Phase 2 Topical Report.Volume 2:Code Evaluation. COLEMAN,D.R. Control Data Corp. November 1985. 120pp. 8512190246. 34010:267.

FRAPCON-2 (V1M5) was applied to generate fuel performance predictions for selected rods of a recently evaluated power reactor data sample. Rod design, operational, and performance data was obtained from the EPRI Fuel Performance Data Base. The data was systematically processed to generate code input parameters. After initial debugging, FRAPCON-2 was applied for bnechmark studies to qualify revised cladding creepdown, growth, and fission gas release models. Benchmark results indicated improved gas release and clad deformation modeling, but increased computer requirements and mechanical conservatism during hard PCI. Comparisons between measured and calculated fuel and cladding corrosion are presented and interpreted relative to code error magnitudes, distributions, and trends versus rod design and operating parameters. The results indicate that FRAPCON-2 (V1M5) is an improved code having adequate capability for best estimate analysis of moderate duty fuel rod performance, up to average burnups of at least 40 GWD/MTU. The main model limitations involve lack of certain physical effects which can generally be compensated for when setting up input or interpreting results; namely, densification effect on decreasing pellet relocation, swelling accommodation by fuel porosity, cladding creepout effect on PCI stress relaxation, and coolant chemistry and pellet impurity effects on clad ID and OD corrosion.

NUREG/CR-3744 V02: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR APRIL-SEPTEMBER 1984. PUGH,C.E. Oak Ridge National Laboratory. January 1985. 244pp. 8502210332. ORNL/TM-9154/V2. 29052:001.

The Heavy-Section Steel Technology (HSST) 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 cracklike flaws. Current work is organized into ten tasks: (1) program management, (2) fracture-methodology and analysis, (3) material characterization and properties, (4) environmentally assisted crack growth studies, (5) crack arrest technology, (6) irradiation effects studies, (7) cladding evaluations, (8) intermediate vessel tests and analysis, (9) thermal-shock technology, and (10) pressurized thermal-shock technology.

NUREG/CR-3746 V02: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.Semiannual Progress Report,April 1984 - September 1984. LIPPINCOTT,E.P.; MCELROY,W.N. Hanford Engineering Development Laboratory. April 1985. 220pp. 8505070562. HEDL-TME 84-21. 30209:020.

This report describes progress made in the Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) during FY84. The primary concern of this program is to improve, test, verify, and standardize the physics-dosimetry-metallurgy and associated reactor and damage analysis procedures and data used for predicting the integrated effects of neutron exposure to LWR-PVs and their support structures. These procedures and data are being recommended in a new and updated set of ASTM standards being prepared, tested, and verified by program participants. These standards, together with parts of the US Code of Federal Regulations and ASME codes, are needed and used for the assessment and control of the condition of LWR-PVs and their support structures during the 30- to 60-year lifetime of a nuclear power plant.

NUREG/CR-3746 V03: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.1984 Annual Report,October 1,1983 - September 30, 1984. MCELROY,W.N. Hanford Engineering Development Laboratory. April 1985. 110pp. 8505070543. HEDL-TME 84-31. 30208:270.

See NUREG/CR-3746,V02 abstract.

NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS. FOLEY,M.G.; KIMBALL,C.S.; MYERS,D.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 119pp. 8506140396. PNL-5064. 30933:275.

Under contract to the U.S. Nuclear Regulatory Commission, Pacific Northwest Laboratory has developed an approach for selecting and testing rock for its suitability and durability as armor for protecting decommissioned uranium mill tailings piles. A preliminary survey of the literature determined that existing techniques for testing rock durability were inadequate for evaluating long-term (>100 years) applications. Suites of rock samples with common lithologies and documented durations of exposure to weathering were then collected and submitted to three-axis ultrasonic testing in an attempt to develop a more reliable testing technique. We found little correlation between the duration of weathering and ultrasound velocity or attenuation in the rock. Through further study, we determined that the best screening approach incorporates common geomorphologic field collection techniques and laboratory tests. Suites of samples with known durations of exposure to weathering can be subjected to wet abrasion and wetting-drying tests to screen local rock types and select those with the greatest potential durability. Furthermore, the expected decrease of rock mass with environmental stresses (e.g., flood impingement and diurnal wetting-drying cycles) can be estimated using this approach.

NUREG/CR-3752: EFFECTS OF HYDROLOGIC VARIABLES ON ROCK RIPRAP DESIGN FOR URANIUM TAILINGS IMPOUNDMENTS. WALTERS,W.H.; SKAGGS,R.L. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1985. 55pp. 8501280379. PNL-5069. 28572:072.

Pacific Northwest Laboratory is studying the mitigation of erosion of earthen radon suppression covers for uranium tailings impoundments. Because the covers will require erosion protection for upwards of 1000 years, rock riprap (armoring) has been proposed as the primary protection method. This study investigates the sensitivity of riprap design procedures to extreme flood events that can generate high flow velocities and shear stresses. The study uses two decommissioned tailings sites

(Grand Junction and Slick Rock, Colorado) as case studies to evaluate the sensitivity of design rock size with respect to variables such as flood discharge, side slope, specific gravity, safety factor, and channel roughness. The results indicate that design rock size can vary significantly for different design procedures. Other significant results indicate that embankment side slopes of about 4H:1V are optimum for rock riprap and that the use of rock material with specific gravities less than 2.50 may prove too costly.

NUREG/CR-3757: TRAN B-2:THE EFFECT OF LOW STEEL CONTENT ON FUEL PENETRATION IN A NON-MELTING CYLINDRICAL FLOW CHANNEL. MCARTHUR,D.A.; MAST,P.K. Sandia National Laboratories. April 1985. 72pp. 8505160176. SAND84-0814. 30457:029.

The TRAN B-Series of experiments is being conducted at Sandia National Laboratories to investigate the characteristics of fuel removal and freezing through the upper axial blankets of an LMFBR during the transition phase of a hypothetical core disruptive accident. The second experiment in this series, TRAN B-2, was performed in July 1983. This experiment involved the injection of a mixture of 95% UO(2) and 5% stainless steel into a simple thick-walled steel cylindrical flow channel. The initial temperature of the steel channel was low, such that melting of the walls upon contact with the hot melt was not expected. Previous experiments under similar conditions but using pure UO(2) melts had shown stable crust growth and fairly long penetration distances. This experiment was intended to investigate whether those results were also applicable for the case of UO(2)/steel mixtures. The results of the TRAN B-2 experiment, consisting of data from online instrumentation and post-irradiation examination, suggest that low steel content fuel/steel mixtures behave very similarly to pure UO(2) melts.

NUREG/CR-3764: BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE. HARRINGTON,R.M.; FULLER,L.C. Oak Ridge National Laboratory. February 1985. 167pp. 8503120456. ORNL/TM-9163. 29338:179.

The BWR-LTAS code was developed by the SASA program at Oak Ridge National Laboratory for the detailed study of specific accident sequences at Browns Ferry Unit One: station blackout, small break LOCA outside primary containment, loss of decay heat removal, loss of vessel water injection, and anticipated transient without scram. The primary use of the code has been to estimate the effects of operator actions on the timing and course of events during the part of the sequence leading up to but not including severe fuel damage. This report documents the basis of the methods used to simulate the response of reactor vessel, primary coolant system, primary containment, and other reactor systems; the output from a sample use of the code is presented.

NUREG/CR-3767: INTERACTIVE SIMULATOR EVALUATION FOR CRT-GENERATED DISPLAYS. BLACKMAN,H.S.; GILMORE,W.E. EG&G, Inc. January 1985. 43pp. 8502210188. EGG-2308. 29035:284.

The United States Nuclear Regulatory Commission (USNRC) is sponsoring an on-going research program to develop methods of assessing various types of computer-generated displays currently being implemented in nuclear power plant control rooms. The purpose of this report is to present an interactive simulation technique for the evaluation of computer-generated displays. The independent variables for this experiment were transient type (six levels), and display type including the levels of star + control panel, bar + control panel, meter + control panel, pressure-temperature map + control panel, and control panel only. The dependent measures were deviations of parameter values comprising the safety functions at risk, percent of time these parameters were out of tolerence from onset of the transient, and accuracy of the operator path in transient mitigation. The results indicate that an interactive simulation method can be used to evaluate various display types, and that the workstation and computer/simulator is an effective configuration. The implications of these results for display evaluation and design are discussed.

NUREG/CR-3772: RELAP5 ASSESSMENT:SEMISCALE SMALL BREAK TESTS S-UT-1,S-UT-2, S-UT-6,S-UT-7 AND S-UT-8. PETERSON,A.C. Sandia National Laboratories. February 1985. 230pp. 8503040547. SAND84-0884. 29198:001.

The RELAP5 independent assessment project is part of an overall effort to evaluate the capability of various system codes to calculate the detailed thermal/hydraulic response of LWRs during accident and off-normal conditions. The RELAP5 computer code is being assessed against test data from various integral and separate effects test facilities. As part of the assessment effort, several small break tests with and without upper head injection (UHI) of emergency core coolant (ECC), performed in the Semiscale Mod-2A facility, have been analyzed. The results show that RELAP5/MOD1 is capable of calculating some aspects of the important phenomena during small breaks both with and without UHI. The times for the system to depressurize to the UHI and/or loop accumulator flow initiation were calculated satisfactorily. The correct trends of the effects of break size and of UHI on the system pressure response were also calculated. The injection rate from the UHI and loop accumulators was not always calculated correctly; the flows cycled on and off because large flow surges caused the accumulator pressures to temporarily decrease below the system pressure. This cycling of the flow had a significant effect on the system response during UHI accumulator flow. When the upper head was liquid-filled from UHI flow, a core liquid level depression was calculated, but not measured, that resulted in a dryout of the core. During UHI flow the calculated densities in the upper plenum and near the top of the core were too high, which also affected the vessel mass distribution. The calculated break flow rates were too large, when the break uncovered later in the transients, contributing to a low liquid level in the vessel and late-time core heatup. Higher late-time core temperatures were calculated than measured both with and without UHI.

NUREG/CR-3774 V02: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2A:Technical Requirements For Belowground Vault Disposal Of Low Level Radioactive Waste. WARRINER,J.B.; BENNETT,R.D. Army, Dept. of, Army Engineer Waterways Experiment Station. October 1985. 93pp. 8511110452. 33412:224.

The study reported herein contains the results of Task 2a (Technical Requirements for Belowground Vault Disposal of Low-Level Radioactive Waste) of a four-task study entitled "Criteria for Evaluating Engineered Facilities." The overall objectives of this study is to ensure that the criteria needed to evaluate five alternative low-level radioactive waste (LLW) disposal methods are available to potential license applicants. The belowground vault disposal alternative is one of several methods that may be proposed for disposal of low-level radioactive waste. In this report, the term belowground vault disposal refers to a near-surface disposal alternative in which the wastes would be disposed of in vaults constructed below ground in excavations and covered with soil. The experience and knowledge gained with this method are described and updated in this report. A generic description of the features and components and operation of a belowground vault disposal facility is provided. Features and components that could enhance the long-term performance are described, including site conditions for which they would be applicable. The applicability of existing criteria developed for near-surface disposal (10 CFR Part 61 Subpart D) to the belowground vault disposal method, as assessed in Task 1, are reassessed herein. With few exceptions, these criteria were found to be applicable in the reassessment. These conclusions differ slightly from the Task 1 findings.

NUREG/CR-3774 V03: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2B:Technical Requirements For Aboveground Vault Disposal Of Low Level Radioactive Waste. BENNETT,R.D.; WARRINER,J.B. Army, Dept. of, Army Engineer Waterways Experiment Station. October 1985. 89pp. 8511110459. 33412:135.

The study reported herein contains the results of Task 2b (Technical Requirements for Aboveground Vault Disposal of Low-Level Radioactive Waste) of a four-task study entitled "Criteria for Evaluating Engineered Facilities." The overall objective of this study is to ensure that the criteria needed to evaluate five alternative low-level radioactive waste (LLW) disposal methods are availabe to poential license applicants. The aboveground vault disposal alternative is one of several methods that may be proposed for disposal of low-level radioactive waste. In this report, the term aboveground vault refers to an engineered structure with roof, walls and floor enclosing the disposal space. The limited experience and knowledge gained with this method are described and updated in this report. The short term experience does not conclusively demonstrate the capability of this method to satisfy the Part 61 Performance Objectives. A generic description of the features and components and operation of an aboveground vault disposal facility is provided. Features and components that could enhance the long-term performance are described. The applicability of existing criteria developed for near-surface disposal (10 CFR Part 61 Subpart D) to the aboveground vault disposal method, as assessed in Task 1, are reassessed herein. With few exceptions, these criteria were found to be applicable in the reassessment. These conclusions differ slightly from the Task 1 findings.

NUREG/CR-3774 V04: ALTERNATIVE METHOD FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2C:Technical Requirements For Earth Mounded Concrete Bunker Disposal Of Low Level Radioactive Waste. MILLER,W.O.; BENNETT,R.D. Army, Dept. of, Army Engineer Waterways Experiment Station. October 1985. 96pp. 8511110415. 33417:218.

The study reported herein contains the results of Task 2c (Technical Requirements for Earth Mounded Concrete Bunker Disposal of Low-Level Radioactive Waste) of a four-task study entitled "Criteria for Evaluating Engineered Facilities." The overall objective of this study is to ensure that the criteria needed to evaluate five alternatve low-level radioactive waste (LLW) disposal methods are availabie potential license applicants. The earth mounded concrete bunker disposal alternaitve is one of several methods that may be proposed for disposal of low-level radioactive waste. The name of this alternative is descriptive of the disposal method used in France at the Centre de la Manche. Experience gained with this method at the Centre is described, including unit operations and features and components. Some improvements to the French system are recommended herein, including the use o previous backfill around momoliths and extending the limits of a low permeability surface layer. The applicability of existing criteria developed for near-surface disposal (10 CFR Part 61 Subpart D) to the earth mounded concrete bunker disposal method, as assessed in Task 1, are reassessed herein. With minor qualifications, these criteria were found to be applicable in the reassessment. These conclusions differ slightly from the Task 1 findings.

NUREG/CR-3774 V05: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2E:Technical Requirements For Shaft Disposal Of Low Level Radioactive Waste. BENNETT,R.D. Army, Dept. of, Army Engineer Waterways Experiment Station. October 1985. 105pp. 8511110408. 33412:031.

The study reported herein contains the results of Task 2e (Technical Requirements for Shaft Disposal of Low-Level Radioactive Waste) of a four-task study entitled "Criteria for Evaluating Engineered Facilities." The overall objective of this study is to ensure that the criteria needed to evaluate five alternaive low-level radioactive waste (LLW) disposal methods are available potential license applicants. The shaft disposal alternative is one of several methods that may be proposed for disposal of low-level radioactive waste. In this report, the term shaft disposal refers to a near-surface disposal alternative in which the wastes would be disposed of in shafts or boreholes augered, bored, or sunk by any other conventional method. The experience and knowledge gained with this method are described and updated in this report. This includes experience in the use of shafts for storage in the U.S. and Canada and research with the method both here and abroad. A generic description of the features and components that could enhance the long-term performance are described, including site conditions for which they would be applicable. The applicability of existing criteria developed for near-surface disposal (10 CFR Part 61 Subpart D) to the shaft disposal method, as assessed in Task 1, are reassessed herein. Without exception, these criteria were found to be applicable in the reassessment. These conclusions differ slightly from the Task 1 findings.

NUREG/CR-3791: CLOSEOUT OF IE BULLETIN 79-09:FAILURE OF GE TYPE AK-2 CIRCUIT BREAKERS IN SAFETY-RELATED SYSTEMS. DEAN,R.S.; FOLEY,W.J.; MILLS,W.R.; et al. Parameter, Inc. January 1985. 47pp. 8501280735. IEB-79-09. 28629:130.

Twelve failures of General Electric Type AK-2 safety-related circuit breakers reported in 1975, 1978 and 1979 are described in IE Bulletin 79-09. Because of these failures, the bulletin was issued April 17, 1979 to require responses and specific actions by all licensees and holders of construction permits. The failures were attributed to either binding within the linkage mechanism of the undervoltage trip device and trip shaft assembly or faulty adjustment of that linkage mechanism. It was concluded that the twelve failures resulted from inadequate preventive maintenance. Because many occurrences of the same kind happened after 1979, a significant number of later NRC documents which are included in Appendix A were issued. The bulletin has been closed out for 101 of the 129 current facilities which reported either that they had no Type AK-2 breakers in safety-related systems or none with undervoltage trip devices. Proposed followup items for the remaining 28 current facilities are presented in Appendix C. Because followup is based on the requirements of later Bulletins 83-04 and 83-08, Bulletin 79-09 is considered closed.

NUREG/CR-3794: CLOSEOUT OF IE BULLETIN 80-25:OPERATING PROBLEMS WITH TARGET ROCK SAFETY-RELIEF VALVES AT BWRS. FOLEY,W.J.; HENNICK,A. Parameter, Inc. January 1985. 44pp. 8501280090. PARAMETER IE-13. 28574:157.

During the three-month period beginning July 25, 1980, five events occurred involving two types of malfunctions of Target Rock safety-relief valves at Boston Edison Company's Pilgrim Nuclear Power Station Unit 1. The first three events were caused by direct failures of the valves; the last two events were caused by nitrogen supply system overpressure which led to valve failure. IE Information Notice 80-40 was issued November 7, 1980 to call attention to the two nitrogen overpressure events. As a result of all five events, IE Bulletin 80-25 was issued December 19, 1980 for action to all 30 BWR facilities with operating licenses or near-term operating licenses, and for information only to 24 facilities then under construction. Actions were to be taken with respect to (1) all Target Rock two-stage, pilot-operated safety-relief valves (SRVs), (2) any make or model of SRV which fails to function as designed, excepting for pressure setpoint requirements and (3) SRV nitrogen/air supply systems. Upon evaluation of utility responses and NRC inspection reports, the bulletin has been closed out for eight of the 30 facilities to which the bulletin was issued for action. For use by NRC/IE, followup items for 22 current facilities with open bulletin status are proposed in Appendix C. Remaining areas of concern and continuing actions dealing with them are described. The bulletin has served its purpose by resulting in identification of the need for corrective actions at all of the 27 current operating facilities to which the bulletin was issued for action.

NUREG/CR-3802: RELAP5 ASSESSMENT:QUANTITATIVE KEY PARAMETERS AND RUN TIME STATISTICS. KMETYK,L.N.; BUXTON,L.D.; THOMPSON,S.L. Sandia National Laboratories. February 1985. 30pp. 8503210472. SAND84-1013. 29479:072.

The advanced best-estimate systems codes currently being developed for the NRC are designed to provide realistic, rather than conservative, predictions of LWR plant behavior during a variety of accidents and transients. The RELAP5 independent code assessment project at Sandia National Laboratories is part of an overall code assessment effort funded by the NRC to arrive at a qualified judgment of the accuracy with which these codes can predict the detailed thermal/hydraulic response of LWRs during accident and off-normal conditions. The heart of the assessment project involves extensive comparison of code results with test data. RELAP5/MOD1 has been assessed at Sandia against a variety of test data from both integral and separate effects test facilities. All these analyses have been documented in detail in individual topical reports and in an overall summary and conclusions report. In this paper, we tabulate the quantitative key parameters for those scenarios which involve leakage of coolant from the primary system to containment, as well as the run time statistics for all the analyses we have performed.

NUREG/CR-3803: THE EFFECTS OF POST-LOCA CONDITIONS ON A PROTECTIVE COATING (PAINT) FOR THE NUCLEAR POWER INDUSTRY. LOYOLA,V.M.; WOMELSDUFF,J.E. Sandia National Laboratories. May 1985. 50pp. 8505230538. SAND84-0806. 30549:014.

We have studied the oxidation of zinc in a zinc-rich coating used in the nuclear power industry and have measured the rates of hydrogen generation from these coatings due to zinc oxidation at temperatures of up to 175 degrees centigrade. The results suggest that the real-time rates of hydrogen generation are considerably higher than previously believed. The higher rates measured in this study are probably due to differences in experimental methodologies between this and previous studies. In this study, the measurements were real-time measurements, as opposed to time-averaged values which are typically obtained. The results suggest, as have the results of other investigators, that the measured rates and reaction parameters may not be those of any specific reaction, but are instead the "effective" values of a series of complex systems operating together. However, the total quantity of hydrogen generated by this mechanism is significantly less than can be produced from other sources, e.g., steam: zirconium.

NUREG/CR-3804 V03: PHYSICS OF REACTOR SAFETY.Quarterly Report,July-September 1984. * Argonne National Laboratory. January 1985. 23pp. 8502210261. ANL-84-35 V03. 29057:325.

This quarterly progress report summarizes work done during the months of July-September 1984 in Argonne National Laboratory's Applied Physics and Components Technology Divisions for the Division of Reactor Safety Research in 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 at 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-3804 V04: PHYSICS OF REACTOR SAFETY.Quarterly Report,October-December 1984. * Argonne National Laboratory. April 1985. 20pp. 8504250268. ANL-84-35 V04. 30032:091.

This quarterly progress report summarizes work done during the months of October-December 1984 in Argonne National Laboratory's Applied Physics and Components Technology Divisions for the Division of Reactor Safety Research in 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 at 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-3805 V02: ENGINEERING CHARACTERIZATION OF GROUND MOTION.Task II: Effects Of Ground Motion Characteristics On Structural Response Considering Localized Structural Nonlinearities And Soil-Structure Interaction Effects. KENNEDY,R.P.; KINCAID,R.H.; SHORT,S.A. Structural Mechanics Associates. March 1985. 158pp. 8504030436. 29604:105.

This report presents the results of part of a two-task study on the engineering characterization of earthquake ground motion for nuclear power plant design. Task I of the study, which is presented in NUREG/CR-3805, Vol. 1, developed a basis for selecting design response spectra taking into account the characteristics of free-field ground motion found to be significant in causing structural damage. Task II incorporates additional considerations of effects of spatial variations of ground motions and soil-structure interaction on foundation motions and structural response. The results of Task II are presented in four parts: (1) effects of ground motion characteristics on structural response of a typical PWR reactor building with localized nonlinearities and soil-structure interaction effects; (2) empirical data on spatial variations of earthquake ground motion; (3) soil-structure interaction effects on structural response; and (4) summary of conclusions and recommendations based on Tasks I and II studies. This report presents the results of the first part of Task II. The results of the other parts will be presented in NUREG/CR-3805, Vols. 3-5.

NUREG/CR-3810 V03: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,July-September 1984. EDLER,S.K. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1985. 35pp. 8503130105. PNL-5106-3. 29360:151.

This document summarizes work performed by Pacific Northwest Laboratory from July 1 through September 30, 1984, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. Results from an instrumented fuel assembly irradiation program being performed at Halden, Norway, are reported. Accelerated pellet-cladding interaction modeling is being conducted to predict the probability of fuel rod failure under normal operating conditions. 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. Fuel assemblies and analytical support are being provided for experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Thermal-hydraulic models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Severe fuel damage accident tests are being conducted in the NRU Reactor, Chalk River, Canada.

NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,October-December 1984. EDLER,S.K. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 31pp. 8506140415. PNL-5106-4. 30908:188.

This document summarizes work performed by Pacific Northwest Laboratory from October 1 through December 31, 1984, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. Results from an instrumented fuel assembly irradiation program being performed at Halden, Norway, are reported. Accelerated pellet-cladding interaction modeling is being conducted to predict the probability of fuel rod failure under normal operating conditions. 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. Fuel assemblies and analytical support are being provided for experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Thermal-hydraulic computer programs are providing best-estimate analyses for a variety of safety issues in light-water reactors. Severe fuel damage tests are being conducted in the NRU Reactor, Chalk River, Canada.

NUREG/CR-3816 V01: REACTOR SAFETY RESEARCH.Quarterly Report,January-March 1984. * Sandia National Laboratories. January 1985. 160pp. 8501300074. SAND84-1072. 28680:095.

The overall objective of this report is to provide NRC with a comprehensive data base essential to (1) defining key safety issues, (2) understanding risk-significant accident sequences, (3) developing and verifying models used in safety assessments, and (4) assuring the public that 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. This report describes progress in a number of activities dealing with current safety issues relevant to both light water reactors (LWRs) and breeder reactors. The work includes a broad range of experiments to simulate accidental conditions to provide the required data base to understand important accident sequences and to serve as a basis for development and verification of the complex computer simulation models and codes used in accident analysis and licensing reviews. Such a program must include the development of analytical models, verified by experiment, which can be used to predict reactor and safety system performance under a broad variety of abnormal conditions.

NUREG/CR-3816 V02: REACTOR SAFETY RESEARCH.Quarterly Report,April-June 1984. * Sandia National Laboratories. April 1985. 211pp. 8504160554. SAND84-1072. 29825:153.

This report describes progress in a number of activities dealing with current safety issues relevant to both light water reactors (LWRs) and breeder reactors. The work includes a broad range of experiments to simulate accidental conditions to provide the required data base to understand important accident sequences and to serve as a basis for development and verification of the complex computer simulation models and codes used in accident analysis and licensing reviews. Such a program must include the development of analytical models, verified by experiment, which can be used to predict reactor and safety system performance under a broad variety of abnormal conditions.

NUREG/CR-3816 V03: REACTOR SAFETY RESEARCH.Quarterly Report,July-September 1984. * Sandia National Laboratories. July 1985. 190pp. 8507250114. SAND84-1072. 31787:147.

Sandia National Laboratories is conducting, under USNRC's sponsorship, phenomenological research related to the safety of commercial nuclear power reactors. The overall objectives of this work is to provide NRC a comprehensive data base essential to (1) defining key safety issues, (2) understanding risk-significant accident sequences, (3) developing and verifying models used in safety assessments, and (4) assuring the public that 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. Together with other programs, the Sandia effort is directed at assuring the soundness of the technology base upon which licensing decisions are made. This report describes progress in a number of activities dealing with current safety issues relevant to both light water and breeder reactors. The work includes a broad range of experiments to simulate accidental conditions to provide the required data base to understand important accident sequences and to serve as a basis for development and verification of the complex computer simulation models and codes used in accident analysis and licensing reviews. Such a program must include the development of analytical models, verified by experiment, which can be used to predict reactor and safety system performance under a broad variety of abnormal conditions.

NUREG/CR-3816 V04: REACTOR SAFETY RESEARCH.Quarterly Report,October-December 1984. * Sandia National Laboratories. September 1985. 242pp. 8510040365. SAND84-1072. 32855:238.

Sandia National Laboratories is conducting, under the USNRC's sponsorship, phenomenological research related to the safety of commercial nuclear power reactors. The research includes experiments to simulate the phenomenology of the accident conditions and the development of analytical models, verified by experiment, which can be used to predict reactor and safety systems performance and behavior under abnormal conditions. The objective of this work is to provide NRC requisite data bases and analytical methods to (1) identify and define safety issues, (2) understand the progression of risk-significant accident sequences, and (3) conduct safety assessments. The collective NRC-sponsored effort at Sandia National Laboratories is directed at enhancing the technology base supporting licensing decisions.

NUREG/CR-3817: DEVELOPMENT,USE AND CONTROL OF MAINTENANCE PROCEDURES IN NUCLEAR POWER PLANTS.Problems And Recommendations. MORGENSTERN,M.; BARNES,V.E.; RADFORD,L.R.; et al. Battelle Human Affairs Research Centers. January 1985. 120pp. 8501280403. PNL-5121. 28572:136.

This report describes the results of activities conducted to assess and document the need for guidance or regulatory involvement by the Nuclear Regulatory Commission (NRC) in the development, upgrading, use and control of maintenance procedures in U.S. nuclear power plants. Reported are the findings of the following four activities: (1) a survey of current maintenance procedure practices in seven U.S. nuclear power plants, (2) a review and analysis of plant administrative and maintenance procedures, (3) a survey of maintenance procedure practices in industries that share some characteristics with the nuclear industry, and (4) a review of the research pertaining to job performance aids and a brief analysis of their applicability to maintenance in nuclear power plants. Based on these findings, several recommendations for NRC action to upgrade maintenance procedure programs are offered.

NUREG/CR-3819: SURVEY OF AGED POWER PLANT FACILITIES. ROSE,J.A.; DEWALL,K.G.; STEELE,R.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). July 1985. 51pp. 8507250162. EGG-2317. 31786:061.

This report presents the results of the survey of Aged Nuclear Power Plant Facilities conducted for the USNRC Office of Nuclear Regulatory Research. The results of this report recommend methods to help formulate comprehensive research program that will systematically identify aging and service wear effects which are likely to affect plant safety. The survey centered on safety related plant systems with regard to component failures from operating histories. The age related failure information gathered from the plant histories was analyzed for reoccurring failure patterns. Emphasis was on identification of specific equipment with high failure rates and of failure mechanism relationships. The data would not support specific equipment identification. It did imply a direct relationship between failure and failure mechanism. 70% of the failures reported were due to four failure mechanisms. In addition there appeared to be a strong correlation between cause of failure and the system in which the component operates. This is verified by detailed study of several plant systems and corroborated by personnel interviews. This survey indicates identification and elimination of system level cause of component failure is a viable approach to prevent and mitigate major reported aging effects.

NUREG/CR-3820 V03: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM.Quarterly Report,July-September 1984. THOMPSON,S.L. Sandia National Laboratories. April 1985. 85pp. 8505160172. SAND84-1025. 30451:130.

The TRAC-PF1/MOD1 independent assessment program at Sandia National Laboratories is part of a multi-faceted effort sponsored by the Nuclear Regulatory Commission to determine the ability of various systems codes to predict the detailed thermal/hydraulic response of LWRs during accident and off-normal conditions. This program is a successor to the RELAP5/MOD1 independent assessment project underway at Sandia for the last two years. The TRAC-PF1/MOD1 code will be assessed against data from various integral and separate effects experimental test facilities, and the calculated results will also be compared with results from our previous RELAP5/MOD1 independent assessment analyses whenever possible.

NUREG/CR-3825 V03-4: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Quarterly Report, April 1984 - September 1984.Volumes 3 and 4. HUTTON,P.H.; KURTZ,R.J. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1985. 22pp. 8504090010. PNL-5125. 29754:016.

Technical progress toward continuous acoustic emission monitoring of nuclear reactor pressure boundaries for flaw detection is described for the period April-September 1984. A draft report of ZB-1 vessel test results was completed. Growth of machined flaws was detected by AE during both 65 degree C and 285 degree C testing. AE data was generally proportional to crack growth. A key result was clear detection of a natural crack in a fabrication weld by AE. Crack growth rates estimated from AE data compared well with measured crack growth rates. In service hydro test monitoring gave mixed results. Impending failure conditions are readily detectable. However, with low overpressure (1.15 x operating pressure), flaws as deep as 70% through-wall did not produce significant AE. With higher overpressure (1.4 x operating pressure), flaws produced identifying AE. An engineering prototype AE monitor system has been completed for use in operational monitoring at Watts Bar Unit 1 reactor. A modified approach to crack growth AE signal identification is producing about 95% correct determinations on recorded waveforms from the ZB-1 vessel test. A report on results from AE monitoring hot functional testing at Watts Bar Unit 1 has been published.

NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS. BOWERMAN,B.S.; SWYLER,K.J.; DOUGHERTY,D.R.; et al. Brookhaven National Laboratory. March 1985. 167pp. 8503280017. BNL-NUREG-51784. 29548:142.

The Technical Position on Waste Form and Container Materials (TP) provides guidance to generators of low-level radioactive waste for meeting the regulations under 10 CFR Part 61 governing the disposal of these wastes. Testing methods are recommended in the TP for assessing material properties relevant to long-term performance in shallow land burial. These tests were reviewed with respect to their application to specific materials: cement, bitumen, vinyl ester-styrene, and polyethylene. In some cases, the applicability of the tests was found to be inadequate, and modifications to the existing tests or alternative methods were recommended. An experimental evaluation of one of the recommended biodegradation tests (the Bartha-Pramer method) was also carried out. Conditions under which this test should be conducted are recommended.

NUREG/CR-3830 V02: AEROSOL RELEASE AND TRANSPORT PROGRAM.Semiannual Progress Report For April 1984-September 1984. ADAMS,R.E.; TOBIAS,M.L. Oak Ridge National Laboratory. January 1985. 50pp. 8502210193. 29035:233.

This report summarizes progress for the Aerosol Release and Transport Program sponsored by the Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Division of Accident Evaluation, for the period April 1984-September 1984. Topics discussed include (1) the experimental program in the Fuel Aerosol Simulant Test Facility, (2) NSPP experiments involving an aerosol of limestone-aggregate concrete in a steam-air atmosphere, (3) revisions in the NSPP experimental program, (4) experiments relating to NSPP thermohydraulic conditions, (5) aerosol-moisture interaction test plans, (6) aerosol code implementation activities, (7) improvements in data processing procedures for NSPP experiments, and (8) a study comparing in-vessel and ex-vessel cascade impactor aerosol size measurements in the NSPP.

NUREG/CR-3831: THE IN-PLANT RELIABILITY DATA BASE FOR NUCLEAR PLANT COMPONENTS.Interim Report - Diesel Generators,Batteries, Chargers And Inverters. KAHL,W.K.; BORKOWSKI,R.J. Oak Ridge National Laboratory. February 1985. 109pp. 8502250363. ORNL/TM-9216. 29094:248.

The objective of the In-Plant Reliability Data (IPRD) program is to develop a comprehensive, component-specific reliability data base for probabilistic risk assessment and for other statistical analyses relevant to component reliability evaluations. This objective is being attained through a cooperative effort with several utilities which have provided access to maintenance files and pertinent population information. This pilot data base includes (1) a component population list (for each plant) of selected electromechanical and mechanical equipment (e.g., pump, valves, etc.), and (2) comprehensive component failure and repair histories based on corrective maintenance actions on these components. This document is the product of a pilot study that was undertaken to demonstrate the methodology and feasibility of applying IPRDS techniques to develop and analyze the reliability characteristics of key electrical components in five nuclear power plants. These electrical components include diesel generators, batteries, battery charges and inverters. The sources used to develop the data base and produce the component failure rates and mean repair times were the plant equipment lists, plant drawings, maintenance work requests, Final Safety Analysis Reports (FSARs), and interviews with plant personnel. The data spanned approximately 33 reactor-years of commercial operation.

NUREG/CR-3837: MULTIPLE-SEQUENTIAL FAILURE MODEL.Evaluation Of And Procedures For Human Error Dependency. SAMANTA,P.K.; O'BRIEN,J.N.; MORRISON,H.W. Brookhaven National Laboratory. May 1985. 114pp. 8512270358. BNL-NUREG-51786. 34090:203.

This report provides an evaluation of the practicality, acceptability, and usefulness of using the Multiple Sequential Failure (MSF) model originally described in NUREG/CR-2111, 1981. The MSF model is described, discussed, and procedures for its use provided. The model was found to be practical, acceptable, and useful as a PRA tool for assessing the dependence due to human interactions with components in systems employing redundant components.

NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984. KELMERS,A.D.; ARNOLD,W.D.; MEYER,R.E.; et al. Oak Ridge National Laboratory. January 1985. 43pp. 8501280387. ORNL/TM-9191/V3. 28572:255.

Geochemical information relevant to the retention of radionuclides by candidate high-level waste repositories being developed by Department of Energy (DOE) projects is being evaluated by Oak Ridge National Laboratory (ORNL) for the Nuclear Regulatory Commission (NRC). During this report period, the project has evaluated radionuclide sorption and solubility values applicable to the candidate repository site in the Columbia River basalts at the Hanford Reservation. The removal of technetium from pertechnetate-traced groundwater by McCoy Canyon basalt under anoxic redox conditions (air excluded) at 27 degrees centigrade was found to be sensitive to the groundwater composition. Sorption of uranium from groundwater by McCoy Canyon basalt under oxic redox conditions at 60 degrees centigrade showed low sorption ratios (1.8 to 2.4 L/kg) similar to those previously obtained at 27 degrees centigrade. The average sorption ratio for strontium in groundwater onto McCoy Canyon basalt under oxic redox conditions at 27 degrees centigrade was 225 L/kg. Column chromatographic experiments with neptunium in groundwater to measure retardation factors at temperatures from 25 to 80 degrees centigrade gave calculated sorption ratio values that were in good agreement with the values previously obtained in batch contact tests.

NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984. KELMERS,A.D.; KESSLER,J.H.; SEELEY,F.G.; et al. Oak Ridge National Laboratory. September 1985. 8[illegible]p. 8509260079. ORNL/TM-9191. 32760:166.

Geochemical information relevant to the retention of radionuclides by candidate high-level nuclear waste geologic repositories being characterized by Department of Energy (DOE) projects is being evaluated by Oak Ridge National Laboratory (ORNL) for the Nuclear Regulatory Commission (NRC). Emphasis has been given to the experimental evaluation of key radionuclides relevant to the Hanford Site being characterized by the Basalt Waste Isolation Project (BWIP). In work by the BWIP, hydrazine was added to groundwater to simulate the reducing redox condition expected in the repository. Such laboratory methodology may not adequately model in situ repository geochemical conditions. We have beem employing anoxic redox conditions to allow the basalt to establish the effective redox condition in batch contact sorption experiments. Sorption of Np(V) or Tc(VII) by basalt from synthetic groundwater under anoxic redox conditions may involve chemisorption reduction reactions on the basalt surface. Our sorption ratio for neptunium under oxic redox conditions does not compare favorably with the value published by the BWIP. The published solubility of technetium under the reducing redox conditions expected by BWIP at the repository probably is based on calculations involving inadequate thermodynamic data. Under oxic redox conditions, our uranium sorption ratio was much lower than values reported by the BWIP. A mineralogical and chemical characterization was completed for the three basalt samples used in our work. Significant differences were seen in both the quantity and composition of the mesostasis. A potential deficiency in the information published by the BWIP is the absence of lithological information as well as mineralogical and chemical characterization for the basalt samples. Our geochemical modeling work suggested that code-to-code evaluation for geochemical calculations may be less important than a detailed evaluation of the data bases.

NUREG/CR-3854: FABRICATION CRITERIA FOR SHIPPING CONTAINERS. FISCHER,L.E.; LAI,W. Lawrence Livermore National Laboratory. March 1985. 17pp. 8504040417. UCRL-53544. 29619:007.

Criteria are identified for controlling the fabrication of metal* components of shipping containers used for transporting radioactive materials. The criteria have been selected from the ASME Code and are based on the level of radioactive materials being transported and the nuclear safety function of the container's components. Criteria are identified for fabrication processes which are related to materials controls, forming, heat treatment, examination and acceptance testing. Implementation of the criteria will ensure the structural integrity of shipping containers at levels consistent with the radioactive materials being transported. (*NOTE: Applies to all metals used in shipping containers construction except cast irons.)

NUREG/CR-3855: CHARACTERIZATION OF NUCLEAR REACTOR CONTAINMENT PENETRATION - FINAL REPORT. SHACKELFORD,M. Argonne National Laboratory. * Sandia National Laboratories. April 1985. 361pp. 8505060534. SAND84-7139. 30192:018.

This report summarizes the survey work conducted by Argonne National Laboratory on the design and details of major penetrations in 48 nuclear power plants. The survey includes all containment types and materials in current use. It also includes details of all types of penetrations (except for electrical penetration assemblies and valves) and the seals and gaskets used in them. The report provides a test matrix for testing major penetrations and for testing seals and gaskets in order to evaluate their leakage potential under severe accident conditions.

NUREG/CR-3862: DEVELOPMENT OF TRANSIENT INITIATING EVENT FREQUENCIES FOR USE IN PROBABILISTIC RISK ASSESSMENTS. MACKOWIAK,D.P.; GENTILLON,C.D.; SMITH,K.L. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1985. 278pp. 8506240069. EGG-2323. 31150:002.

Transient initiating event frequencies are an essential input to the analysis process of a nuclear power plant probabilistic risk assessment. These frequencies describe events causing or requiring scrams. This report documents an effort to validate and update from other sources a computer-based data file developed by the Electric Power Research Institute (EPRI) describing such events at 52 United States commercial nuclear power plants. Operating information from the United States Nuclear Regulatory Commission on 24 additional plants from their date of commercial operation has been combined with the EPRI data, and the entire data base has been updated to add 1980 through 1983 events for all 76 plants. The validity of the EPRI data and data analysis methodology and the adequacy of the EPRI transient categories are examined. New transient initiating event frequencies are derived from the expanded data base using the EPRI transient categories and data display methods. Upper bounds for these frequencies are also provided. Additional analyses explore changes in the dominant transients, changes in transient outage times and their impact on plant operation, and the effects of power level and scheduled scrams on transient event frequencies. A more rigorous data analysis methodology is developed to encourage further refinement of the transient initiating event frequencies derived herein.

NUREG/CR-3863: ASSESSMENT OF CLASS 1E PRESSURE TRANSMITTER RESPONSE WHEN SUBJECTED TO HARSH ENVIRONMENT SCREENING TESTS. FURGAL,D.T.; CRAFT,C.M.; SALAZAR,E.A. Sandia National Laboratories. April 1985. 194pp. 8506140052. SAND84-1264. 30907:289.

An experimental investigation into the performance of Class 1E electronic pressure transmitters exposed to environments within and beyond the design basis was conducted. Emphasis was placed on determining the instruments' failure and degradation modes in separate and simultaneous environmental exposures. Five unaged ITT Barton Model 763 pressure transmitters were tested and exposed to a unique environment. The response of the transmitters showed that temperature was the primary environmental stress affecting the tested transmitters' performance. Initial large errors that decrease with time-at-temperature were observed. The source of these errors is believed to be a common mode design weakness in the transmitters' calibration potentiometers. This weakness results from a dependency of material dielectric properties on temperature. The modification recommended by the manufacturer, although palliative in nature, did reduce this temperature-induced effect after the first few minutes of accident exposure. A potential second common failure mode which activates slowly with time-at-temperature was also identified. The operation of this failure mechanism is believed to be catalyzed by the presence of a lubricant used in the production of some potentiometers. The design of this transmitter proved to be exceptionally hard to radiation effects and there appeared to be no significant synergistic effects between radiation and temperature. The observed responses of the transmitters offer support for the position of IEEE 381-1977 which recommends that electronic modules aged to varying degrees of advanced life should be tested.

NUREG/CR-3865: EVALUATION OF THE RADIOACTIVE INVENTORY IN,AND ESTIMATION OF ISOTOPIC RELEASE FROM,THE WASTE IN EIGHT TRENCHES AT THE SHEFFIELD LOW-LEVEL WASTE BURIAL SITE. MACKENZIE,D.R.; SMALLEY,J.F.; KEMPF,C.R.; et al. Brookhaven National Laboratory. January 1985. 196pp. 8503040014. BNL-NUREG-51792. 29197:113.

An inventory has been compiled of the isotopes of half-life >5 years buried in eight of the trenches at the Sheffield radioactive shipment records (RSRs). Pertinent information from some 3200 fuel cycle RSRs and 1700 non-fuel cycle RSRs has been stored in a computerized data base and used to develop the inventory. Results of the compilation are in disagreement with the two previous estimates for H-3 and C-14. In particular, non-fuel cycle H-3 inventory for the eight trenches of the present study is approximately a factor of 2 higher than either previous estimates of total site inventory. Modeling of release processes has been carried out in order to obtain estimates of isotopic release rates from waste packages to the trenches. This modeling is highly speculative, but believed to be state-of-the-art. It required information not only on amounts of the different isotopes, but also on the waste forms and containers holding them. Such information was generally not given on the RSRs and had to be obtained by contact with the generators. Estimated numerical release rate data are given for each trench for H-3, C-14, Cs-137, Sr-90, and Co-60. I-129 is expected to have been totally released within a year of container breaching by corrosion. Most of the Pu, in the form of oxide, will probably not be released at a significant rate.

NUREG/CR-3866: TRAC-PD2 INDEPENDENT ASSESSMENT. KNIGHT,T.D. Los Alamos Scientific Laboratory. March 1985. 440pp. 8503220009. LA-10166-MS. 29488:307.

This report documents the Los Alamos results of the second assessment phase, independent assessment, for TRAC-PD2. We documented the results of the developmental assessment for TRAC-PD2 in an earlier report. This report describes calculations run with the released versions of TRAC-PD2. We analyzed separate-effects tests to investigate the critical-flow calculation, the emergency-core-cooling (ECC) bypass behavior, and the reflood-tracking capability. We analyzed integral tests to explore the gravity-driven reflood behavior, and the small-break LOCA behavior. The results show good agreement between the calculated parameters and the data for those tests related to large-break LOCA. In general, the comparisons to small-break LOCA tests indicated that the code can be useful but that some model improvements are required.

NUREG/CR-3872: DATA ACQUISITION AND CONTROL OF THE HSST SERIES V IRRADIATION EXPERIMENT AT THE ORR. MILLER,L.F.; HOBBS,R.W. Oak Ridge National Laboratory. April 1985. 97pp. 8505230574. ORNL/TM-9253. 30547:232.

Documentation relative to data acquisition and control for support of the HSST Series V Irradiation Experiment at the Oak Ridge Research (ORR) is included in this report. Part A describes the computer system hardware and real-time application support software, and Part B describes the temperature control methodology. Software that acquires data from analog input provides this information to the control algorithm software. Results from the control algorithm are, in turn, utilized by software which controls digital output hardware. Time intervals of execution, as well as sequencing of software modules, are controlled through commands to the operating system. Temperature data are recorded at one-hour intervals on computer printouts for documentation and immediate analysis and one magnetic media for permanent storage and subsequent analyses. Results from processing of data files show that the average temperatures at the 1/4T and 3/4T positions are maintained within 2.6 degrees centigrade of 288 degrees centigrade with associated standard deviations of less than 3 degrees centigrade. Average temperatures of the other thermocouples are maintained within 288 degrees centigrade plus or minus 12 degrees centigrade with standard deviations less than 3 degrees centigrade.

NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES. HWANG,H.; KAGAMI,S.; REICH,M.; et al. Brookhaven National Laboratory. August 1985. 99pp. 8509110003. BNL-NUREG-51795. 32561:213.

This report describes a research effort for the development of the probability-based load combination criteria for design of concrete containment structures. The proposed criteria are in a load and resistance factor design (LRFD) format. In order to test the performance objectives of the proposed criteria, four representative structures are selected using a Latin hypercube sampling technique. Next, the reliability analysis method developed by Brookhaven National Laboratory is employed to assess the reliability of these representative containments. Furthermore, an objective function is defined and a minimization technique is developed to find the optimum load factors. The load factors for accident pressure due to the design basis accident and safe shutdown earthquake are derived for three target limit state probabilities. Other load factors are also discussed on the basis of prior experience with probability-based design criteria for ordinary building construction. The proposed load combinations are based on the best available data to date pertaining to loads and resistances. If in the future the data base changes, the developed methodology can readily be utilized to update the load factors resulting from these changes.

NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE. BOEGEL,A.J.; CHOCKIE,A.D.; HUENEFELD,J.C.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 120pp. 8507080183. PNL-5160. 2*392:345.

This report presents the results of a project designed to compare and contrast Japanese and United States nuclear power plant operating experience, preventive maintenance/surveillance requirements, and organization and management practices relating to maintenance. Findings are based on information obtained on the November-December 1983 and November 1984 visits to Japan by the NRC and representatives of Battelle's Pacific Northwest Division, and on various documents obtained from the Japanese (primarily the Ministry of International Trade and Industry--MITI) during and subsequent to the visits. U.S. data sources included NUREG-0020 (Greybook) and plant technical specifications. The study shows that Japanese plants experienced far fewer manual shutdowns, manual scrams, automatic scrams, and reduced loads than U.S. plants and that their mean-time-between-event (MTBE), even when adjusted for differences in average plant availability, was approximately 10 times greater than the U.S. MTBE. The report also points out significant differences in the Japanese approach to preventive maintenance, and in the Japanese regulatory approach to maintenance, their management and organizational context for maintenance, and other socioeconomic factors that may affect the performance of maintenance.

NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1-September 30,1984. BALL,S.J.; CLEVELAND,J.C.; HARRINGTON,R.M.; et al. Oak Ridge National Laboratory. April 1985. 28pp. 8505230519. ORNL/TM-9267/V3. 30549:063.

Modeling and code development work on the modular High-Temperature Gas-Cooled Reactor (HTGR) were continued. The longer-term heatup accident scenario in which cavity wall cooling is lost was also modeled. Sensitivity studies were run for a variety of parameter variations. Fission-product (FP) release and transport experiments were completed for several additional elements.

NUREG/CR-3885 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, October 1-December 31,1984. BALL,S.J.; CLEVELAND,J.C.; HARRINGTON,R.M.; et al. Oak Ridge National Laboratory. August 1985. 24pp. 8508210424. ORNL/TM-9267/V4. 32334:209.

Modeling and code development work on the modular High-Temperature Gas-Cooled Reactor (HTGR) continued with the development and testing of a thermal model of the upper reflector. The longer-term heatup accident scenario in which cavity wall cooling is lost was also modeled. Sensitivity studies were run for variations in soil conductivity and decay heat generation rate. Fission-product (FP) release and transport experiments were completed and initiated for several additional elements. Progress was made in establishing an FP redistribution capability in the ORECA code.

NUREG/CR-3887: HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS. KROIS,P.A.; HAAS,P.M.; MANNING,J.J.; et al. Oak Ridge National Laboratory. October 1985. 237pp. 8512120160. ORNL/TM-9266. 33871:139.

This report describes a human factors research project performed to: (1) support the Severe Accident Sequence Analysis (SASA) program and (2) develop a descriptive model of operator response in accident managment. The first goal was accomplished by working with SASA analysts on the Browns Ferry Unit One anticipated transient without scram (ATWS) accident sequence to systematically assess critical operator actions and thereby demonstrate contributions to SASA analyses from human factors data and methods. The second goal was accomplished by developing a model called the Function Oriented Accident Management (FOAM) model, which provides both a conceptual structure linking off-normal safety functions with potential unconventional emergency responses and a method for developing technical guidance for those responses based on operations, engineering, and human factors data and expertise. The four components comprising the model are described and their use is shown through a table-top demonstration.

NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE. KOH,B.R.; KIM,S.H.; TALEYARKHAN,R.; et al. Oak Ridge National Laboratory. May 1985. 279pp. 8505160635. 30444:230.

This report summarizes improvements and modifications made in the MELRPI computer code. A major difference between this new, updated version of the code, called MELRPI.MOD2, and the one reported previously, concerns the inclusion of a model for the BWR emergency core cooling systems (ECCS). This model and its computer implementation, the ECCRPI subroutine, account for various emergency injection modes, for both intact and rubblized geometries. Other changes to MELRPI deal with an improved model for canister wall oxidation, rubble bed modeling, and numerical integration of system equations. A complete documentation of the entire MELRPI.MOD2 code is also given, including an input guide, list of subroutines, sample input/output and program listing.

NUREG/CR-3900 V02: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,July-September 1984. STAHL,D.; MILLER,N.E. Battelle Memorial Institute, Columbus Laboratories. January 1985. 117pp. 8502130361. 28914:098.

During this reporting period, it was found that glass-water contact during the nonisothermal periods of leach testing may influence the test results. Modeling of wasteform degradation focused on dissolution/reprecipitation kinetics. An experiment is planned to verify this model. A procedure was developed to disperse RuO(2) in MCC 76-68 glass. Potentiodynamic polarization tests were performed to determine the effects of single chemical species in groundwater on the cracking and pitting susceptibility of carbon steel. Slow strain rate tests show that carbon steel is especially susceptible to stress-corrosion cracking in aqueous FeCl(3) at low strain rates. The strength of commercial high-purity iron was found not to be affected by hydrogen; however, ductility was somewhat reduced. The description of groundwater radiolysis was further refined during this quarter. Integral experiments are being prepared to provide information on combined-effects processes that may influence the long-term performance of the waste package.

NUREG/CR-3900 V03: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,October-December 1984. STAHL,D.; MILLER,N.E. Battelle Memorial Institute, Columbus Laboratories. May 1985. 88pp. 8506060746. 30776:173.

Experiments for evaluating the glass-dissolution model are underway, and the procedure developed last quarter for dispersing RuO(2) in MCC 76-68 glass has been tested and proved to produce appropriate particle concentrations. Acetic and humic acids have been chosen to test the effect of natural organic acids on waste glass performance. In the overpack-corrosion effort, potentiodynamic polarization tests indicate that of the 15 chemical species tested, all but perchloate and hydrogen may effect stress-corrosion cracking behavior of carbon steel; several synergistic effects were also indicated. In slow strain rate studies, specimens tested in 0.0005 M FeCl(3) (a much lower chloride concentration than expected in groundwater) exhibited significant cracking over the temperature range 250-315 degrees centigrade. Pits were found to propagate readily, but slowly, in 1018 carbon steel exposed to aerated basalt groundwater at 90 degrees centigrade. The general-corrosion correlation was changed to incorporate a finite rate of film growth. Integral experiments are being prepared to provide information on combined-effects processes that may influence the long-term performance of the waste package.

NUREG/CR-3900 V04: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Annual Report,April 1984 - April 1985. STAHL,D.; MILLER,N.E. Battelle Memorial Institute, Columbus Laboratories. July 1985. 235pp. 8508150047. BMI-2127. 32197:107.

Waste form experimentation has focused on borosilicate glass, using the reference composition MCC 76-68. An experiment investigated the influence of continuous contact between the glass specimen and the leachate on the results of corrosion studies. It was found that precipitates formed during cooling can affect the results. Other experiments evaluated the influence of crystallization on glass waste-form performance and the influence of organic acid on the waste-form and radionuclide mobility in groundwater. Models were used to analyze glass dissolution, including the reprecipitation of dissolved glass species. The effect of groundwater species on the electrochemistry of steels is being analyzed to evaluate susceptability to pitting and stress-corrosion cracking. Species identified as potential cracking agents are being investigated by slow strain rate experiments. Hydrogen embrittlement studies of steel showed annealed cast steel to be more sensitive to embrittlement. Realistic general and pitting corrosion models are being developed, based on known principles of mass transport and radiolytic production. Mechanical and water-chemistry-related stresses which influence mechanical degradation were evaluated. Groundwater-radiolysis and water-chemistry studies are continuing as part of the integrated system performance task.

NUREG/CR-3901: DOCUMENTATION AND USER'S GUIDE:GS2 & GS3 - VARIABLY SATURATED FLOW AND MASS TRANSPORT MODELS. DAVIS,L.A.; SEGOL,G. Water, Waste & Land, Inc. June 1985. 305pp. 8507250143. WWL/TM-1791-2. 31789:002.

This report presents documentation and user's manual for programs GS2 (two-dimensional version) and GS3 (three-dimensional version). Mathematical equations and physical principles utilized to develop the code are presented in Section 2. The numerical approach used (Galerkin Finite Elements) is presented in Section 3. Section 4 presents an overview of how problems should be set up to properly use the code while detailed input instructions are presented in Section 5. Output produced by the code is discussed in Section 6. Three example problems, including sample input data sets and output data, are presented in Section 7. Program information is provided in Section 8. A listing of important program variables along with complete program listings are presented in the Appendices. This report was prepared as part of a project in which NRC staff was presented a training course on how to properly use this computer program. Programs GS2 and GS3 can be utilized to analyze flow and mass transport in unsaturated, partially saturated, or fully saturated flow regions. It is anticipated that the NRC will use the model for checking information provided by a licensee, for evaluating alternative sites and designs for waste disposal, and for comparing their results with results from other methods of solution.

NUREG/CR-3904: A COMPARISON OF UNCERTAINTY AND SENSITIVITY ANALYSIS TECHNIQUES FOR COMPUTER MODELS. IMAN,R.L.; HELTON,J.C. Sandia National Laboratories. May 1985. 118pp. 8506190020. SAND84-1461. 31019:001.

Uncertainty analysis and sensitivity analysis are important elements in the development and implementation of computer models for complex processes. Typically, there are many uncertainties associated with both the development and the application of such models. Understanding of these uncertainties and their causes is required to effectively interpret model behavior. Many different techniques have been proposed for performing uncertainty and sensitivity analyses. The objective of the present study is to compare several widely used techniques on three models having large uncertainties and varying degrees of complexity in order to highlight some of the problem areas that must be addressed in actual applications. The following approaches to uncertainty and sensitivity analysis are considered: (1) response surface methodology based on input determined from a fractional factorial design, (2) Latin hypercube sampling with and without regression analysis, and (3) differential analysis. These techniques are compared on the basis of (1) ease of implementation, (2) flexibility, (3) estimation of the cumulative distribution function of the output, and (4) adaptability to different methods of sensitivity analysis. With respect to these criteria, the technique using Latin hypercube sampling and regression analysis gives the best results overall. The models used in the comparisons are well documented, thus making it possible for researchers to make comparisons of other techniques with the results in this study.

NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide. GREEN,N.M.; MAYS,G.T.; JOHNSON,M.P.; et al. Oak Ridge National Laboratory. April 1985. 164pp. 8505070182. ORNL/NSIC-223. 30216:139.

Operating experience data from nuclear power plants are essential for safety and reliability analyses, especially analyses of trends and patterns. The licensee event reports (LERs) that are submitted to the Nuclear Regulatory Commission (NRC) by the nuclear power plant utilities contain much of this data. The NRC's Office of Analysis and Evaluation of Operational Data (AEOD) has developed, under contract with NOAC, a system for codifying the events reported in the LERs. The primary objective of the Sequence Coding and Search System (SCSS) is to reduce the descriptive text of the LERs to coded sequences that are both computer-readable and computer-searchable. This system provides a structured format for detailed coding of component, system, and unit effects as well as personnel errors. This four volume report documents and describes SCSS in detail. Volume 1 is a User's Guide for searching the SCSS database. Chapter 2 of this guide is a tutorial on retrieving, displaying, and analyzing LERs and provides hands-on experience in executing basic commands. Volume 2 contains all valid and acceptable codes used for searching and encoding the LER data. Volumes 3 and 4 provide a technical processor, new to SCSS, the information and methodology necessary to capture descriptive data from the LER and to codify that data into a structured format and serve as reference material for the more experienced technical processor, and contains information that is essential for the more advanced user who needs to be familiar with the intricate coding techniques in order to retrieve specific details in a sequence.

NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings. GALLAHER,R.B.; GUYMON,R.H.; MAYS,G.T.; et al. Oak Ridge National Laboratory. April 1985. 271pp. 8505070170. ORNL/NSIC-223. 30214:022.

See NUREG/CR-3905,V01 abstract.

NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual. GALLAHER,R.B.; GUYMON,R.H.; MAYS,G.T.; et al. Oak Ridge National Laboratory. April 1985. 381pp. 8505070006. ORNL/NSIC-223. 30213:001.

See NUREG/CR-3905,V01 abstract.

NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual. GALLAHER,R.B.; GUYMON,R.H.; MAYS,G.T.; et al. Oak Ridge National Laboratory. April 1985. 347pp. 8505070184. ORNL/NSIC-223. 30212:001.

See NUREG/CR-3905,V01 abstract.

NUREG/CR-3906: URANIUM MILL TAILINGS NEUTRALIZATION:CONTAMINANT COMPLEXATION AND TAILINGS LEACHING STUDY. OPITZ,B.E.; DODSON,M.E.; SERNE,R.J. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 77pp. 8506190026. PNL-5179. 31019:119.

Laboratory experiments were performed to compare the effectiveness of limestone and hydrated lime for improving waste water quality through the neutralization of acidic uranium mill tailings liquor. The experiments were designed to assess the effects of three proposed mechanisms — carbonate complexation, elevated pH and colloidal particle adsorption — on the solubility of toxic contaminants found in a typical uranium mill waste solution. Of special interest were the effects of each of these possible mechanisms on the solution concentrations of trace metals such as Cd, Co, Mo, Zn and U after neutralization. Acidic untreated solid tailings from two mill sites and tailings neutralized with lime were leached with a laboratory-prepared ground water for several pore displacement volumes. Analyses performed on the column effluents indicate that prior neutralization results in a significant reduction in the concentration of all pH dependent constituents in the column effluents. In contrast, relatively high concentrations of several trace metals and macro ions were found in effluent solution from the untreated tailings columns.

NUREG/CR-3911 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Quarterly Report,April-June 1984. ATTERIDGE,D.G.; BRUEMMER,S.M.; PAGE,R.E. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1985. 42pp. 8503220003. PNL-5181. 29487:176.

The Division of Engineering Technology, U.S. Nuclear Regulatory Commission, is sponsoring a program at Pacific Northwest Laboratory to evaluate welded and repair-welded stainless steel piping for light-water reactor service. Stainless steels often become sensitized, or less resistant to stress corrosion cracking (SCC), after undergoing heating and cooling cycles such as those encountered in welding. The weld heat-affected zone is often the site of crack initiation. This program will therefore measure and model the development of a sensitized microstructure and its resultant resistance to SCC in welded and repair-welded stainless steel pipe. The result will be a method to assess the effects of welding variables on the SCC susceptibility of component-specific nuclear reactor/repairs.

NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT. CAMP,A.L.; BEHR,V.L.; HASKIN,F.E. Sandia National Laboratories. January 1985. 214pp. 8503050511. SAND83-0501. 29264:001.

The MARCH and HECTR computer codes are used in this study to examine hydrogen production, transport, and combustion in an ice-condenser containment for a number of hypothesized severe accidents. Both degraded-core and core-melt-down accidents are treated. The sensitivity of the containment pressure-temperature response is assessed for a number of factors, including the hydrogen and steam source-term assumptions, ignition and propagation limits, combustion completeness, flame speed, spray operation, and recirculation fan operation. The highest containment pressures occur for those cases where the igniters are assumed to fail, the recirculation fans or containment sprays are assumed to fail, or very large steam and hydrogen releases accompanying vessel breach are predicted.

NUREG/CR-3913: HECTR VERSION 1.0 USER'S MANUAL. CAMP,A.L.; WESTER,M.J.; DINGMAN,S.E. Sandia National Laboratories. April 1985. 325pp. 8504160098. SAND84-1522. 29832:026.

This report describes the features and use of HECTR Version 1.0. HECTR is a relatively fast-running, lumped-volume containment analysis computer program that is most useful for performing parametric studies. The main purpose of HECTR is to analyze nuclear reactor accidents involving the transport and combustion of hydrogen, but HECTR can also function as an experiment analysis tool and can solve a limited set of other types of containment problems. HECTR Version 1.0 has been particularly tailored to analyze accidents in ice-condenser PWR and Mark III BWR containments. HECTR is designed for flexibility and provides for user control of many important parameters, particularly those related to hydrogen combustion. Built-in correlations and default values of key parameters are also provided.

NUREG/CR-3914: PUMP AND VALVE QUALIFICATION REVIEW GUIDE. MILLER,B.E. Brookhaven National Laboratory. October 1985. 58pp. 8511050305. BNL-NUREG-51807. 33339:122.

This report provides NRC reviewers with guidance, assistance, and examples relating to the information and procedures to be included in an applicant's pump and valve operability assurance program, and to the scope and depth of the review. Discussed are the applicable components, concerns, methodologies, documentation, evaluation procedures, and examples of design and operability issues for pump and valve assemblies. These items should be of concern and included in an applicant's qualification program.

NUREG/CR-3915: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL. HUTTON,P.H.; KURTZ,R.J.; PAPPAS,R.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1985. 240pp. 8509300233. PNL-5184. 32811:317.

Acoustic emission (AE) monitoring of flaw growth in an intermediate scale vessel during cyclic loading at 65 degrees centigrade and 288 degrees centigrade is described in this report. The report deals with background, methodology, and results. The work discussed is of major significance in a program supported by NRC to develop and demonstrate application of AE monitoring for continuous surveillance of reactor pressure boundaries to detect and evaluate growing flaws. Several areas of technical concern are addressed. Results support the feasibility of effective continuous monitoring.

NUREG/CR-3919: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:NEPTUNUS PRESSURIZER TEST Y05. PETERSON,A.C. Sandia National Laboratories. February 1985. 60pp. 8503040543. SAND84-1534. 29218:281.

The TRAC independent assessment project is part of an overall effort to determine the capability of various system codes to predict the detailed thermal/hydraulic response of light water reactors during accident and off-normal conditions. The TRAC computer code is being assessed against test data from various integral and separate effects test facilities. As part of this assessment effort, a separate effects component test performed in the NEPTUNUS pressurizer test facility for Thermal Power Engineering at Delft University of Technology was analyzed with TRAC-PF1/MOD1. The test simulated insurges, combined with spray flow, and outsurges from a pressurizer, and was selected for code assessment because the capability of the computer codes used in safety analyses to calculate the correct pressurizer response is an important concern of the NRC. The report summarizes results showing somewhat higher pressures and fluid temperatures were calculated during insurges with spray flow than were measured in the test. A contributing factor to the calculation of high pressures and fluid temperatures appears to be that the interfacial heat transfer from superheated vapor to subcooled liquid was too low. The calculational results for the base analysis and some modeling studies are discussed. A TRAC-PF1/MOD1 input listing of the base case model is also provided.

NUREG/CR-3922 V01: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Main Report And Appendices A And B. MURPHY,G.A. Oak Ridge National Laboratory. CASADA,M.L.; JOHNSON,M.P.; et al. JBF Associates. January 1985. 114pp. 8502120062. ORNL/NOAC-224. 28872:137.

This report describes the first phase of an NRC-sponsored project that identified and evaluated system interaction (SI) events that have occurred at commercial nuclear power plants in the United States. The project included: an assessment of nuclear power plant operating experience data sources; the development of search methods and event selection criteria for identifying SI events; review of possible SI events; and final evaluation and categorization of events. The report outlines each of these steps and presents the results of the project. The results include 235 events identified as adverse system interactions and 2? categories into which those events were assigned. The categories represent groups of similar events and include areas such as: adverse interactions between normal or offsite power and emergency power systems; degradation of safety systems by vapor or gas intrusion; degradation of safety-related equipment by fire protection systems; and flooding of safety-related equipment through plant drain systems. After evaluating each category (and the events contained in them), the emphasis on the potential for continued problems in these areas should be examined; and current system interaction analyses methods should be studied to determine their effectiveness for identifying system interaction events. (Phase II of this project, "Evaluation of System Interaction Methods," will assess the effectiveness of current methods using the events identified in this report).

NUREG/CR-3922 V02: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Appendices C And D. MURPHY,G.A. Oak Ridge National Laboratory. CASADA,M.L.; JOHNSON,M.P.; et al. JBF Associates. January 1985. 265pp. 8502120069. ORNL/NOAC-224. 28871:002.

See NUREG/CR-3922,V01 abstract.

NUREG/CR-3930: OBSERVED BEHAVIOR OF CESIUM,IODINE,AND TELLURIUM IN THE ORNL FISSION PRODUCT RELEASE PROGRAM. COLLINS,J.L.; OSBORNE,M.F.; LORENZ,R.A.; et al. Oak Ridge National Laboratory. April 1985. 73pp. 8504170679. ORNL/TM-9316. 29907:154.

Two control tests were conducted to study the behavior of CsI, CsOH, and Te in the experimental apparatus used to conduct fission product release tests with highly irradiated LWR fuel at ORNL. In this report the control tests are described, and the results are compared with those obtained for cesium, iodine, and tellurium in 26 tests of irradiated fuel and other tests using tracers. In good agreement with the LWR fuel tests, the CsI behavior in the control tests was similar to that observed for iodine in the fuel tests; iodine was released primarily as CsI rather than highly volatile molecular iodine. Cesium (not associated with CsI) behaved like CsOH in the LWR fuel tests. In both LWR fuel tests and the control tests, cesium hydroxide was observed to react with and be retained by zirconia ceramic surfaces in the temperature range 800 to 1200 degrees centigrade, probably forming cesium metazirconate (Cs(2)ZrO(3). In one of the control tests, cesium hydroxide reacted with tellurium in the gas phase and was collected as CsTe. Although the results are limited at this time, the indicated collected behavior of teeiurium in the LWR fuel tests has been that of a telluride.

NUREG/CR-3935: THERMAL-HYDRAULIC ANALYSES OF OVERCOOLING SEQUENCES FOR THE H.B. ROBINSON UNIT 2 PRESSURIZED THERMAL SHOCK STUDY. FLETCHER,C.D.; DAVIS,C.B.; OGDEN,D.M. EG&G Idaho, Inc. (subs. of EG&G, Inc.). July 1985. 291pp. 8507250171. EGG-2335. 31785:130.

Oak Ridge National Laboratory (ORNL), as a part of the Nuclear Regulatory Commission's (NRC's) pressurized thermal shock (PTS) integration study for the resolution of Unresolved Safety Issue A49, identified overcooling sequences of interest to the H.B. Robinson PTS study. For each sequence, reactor vessel down-comer fluid pressure and temperature histories were required for the two-hour period following the initiating event. Analyses previously performed at the Idaho National Engineering Laboratory (INEL) fully investigated a limited number of the sequences using a detailed RELAP5 model of the H.B. Robinson, Unit 2 (HBR-2) plant. However, a full investigation of all sequences using the detailed model was not economically practical. New methods were required to generate results for the remaining sequences. Pressure and temperature histories for these remaining sequences were generated at the INEL through a process combining: (a) partial-length calculations using the detailed RELAP5 model, (b) full-length calculations using a simplified RELAP5 model, and (c) hand calculations. This report documents both the methods used in this process and the results. The sequences investigated contain significant conservatisms concerning equipment failures, operator actions, or both. Consequently, care should be taken in applying the results presented herein without an understanding of the conservatisms and assumptions. The results of the thermal-hydraulic analyses presented here, along with additional analyses of multidimensional and fracture mechanics effects, will be utilized by ORNL to assist the NRC in resolving the PTS unresolved safety issue.

NUREG/CR-3936: RELAP5 ASSESSMENT:CONCLUSIONS AND USER GUIDELINES. KMETYK,L.N. Sandia National Laboratories. January 1985. 182pp. 8502010096. SAND84-1122. 28700:001.

The RELAP5 independent assessment project at Sandia National Laboratories is part of an overall effort funded by the NRC to determine the ability of various systems codes to predict the detailed thermal/hydraulic response of LWRs during accident and off-normal conditions. The RELAP5/MOD1 code has been assessed at Sandia against a variety of test data from both integral and separate effects test facilities. All these analyses have been documented in detail in individual topical reports; in this paper we attempt to evaluate the overall code performance by comparing results from many different calculations, and to offer other users some guidelines based on our experience to date. All results show that good primary side steady state initial and/or operating conditions are readily obtained, given adequate facility descriptions and some user experience or guidelines, although problems are usually encountered in the steam generator secondary sides.

NUREG/CR-3937: STEAM GENERATOR TUBE RUPTURE IODINE TRANSPORT MECHANISMS. Task 1:Experimental Studies. GIESEKE,J.A.; FLANIGAN,L.J.; COLLIER,R.P.; et al. Battelle Memorial Institute, Columbus Laboratories. October 1985. 36pp. 8511110327. BMI-2114. 33412:317.

Iodine release from a nuclear power plant during steam generator tube rupture accidents is expected to be strongly dependent on the drop sizes formed as high pressure primary system water is flashed and atomized as it passes through the rupture opening. This study was based on the need for information on drop sizes formed under such conditions. Experiments to measure the fraction of water flashed and the drop sizes formed were performed at typical operating pressures and temperatures with the actual tube diameters and lengths nearly to scale. The mass median drop sizes measured were in the range from about 20 to 60 micrometers for both open-ended and slit rupture geometries. No significant effect on drop size of primary system pressure level was noted over the range from 1100 to 2100 psig.

NUREG/CR-3943: THE BWR PLAN ANALYZER. WULFF,W.; CHENG,H.S.; LEKACH,S.V.; et al. Brookhaven National Laboratory. February 1985. 345pp. 8503120461. BNL-NUREG-51812. 29339:131.

This final report describes the modeling, the software and the hardware of the plant analyzer. The report also presents the first developmental assessment and contains a user guide for the plant analyzer. A large number of transients have been simulated. The simulation encompasses the neutron kinetics, the thermal conduction in fuel structures and the hydraulics of non-equilibrium, nonhomogeneous two-phase flow in the nuclear steam supply system, steam line dynamics, turbines, condensers, feedwater trains and the suppression pool, as well as the control and plant protection systems. All simulations can be carried out at speeds up to 10 times faster than real-time process speeds. The technology presented here has been developed primarily for cost-effective safety analyses, but is also invaluable for plant monitoring, failure diagnosis and computer-aided mitigation of accidents.

NUREG/CR-3944: TRAN B-3:EXPERIMENTAL INVESTIGATION OF FUEL CRUST STABILITY ON MELTING SURFACES OF AN ANNULAR FLOW CHANNEL. MCARTHUR,D.A.; MAST,P.K. Sandia National Laboratories. April 1985. 61pp. 8505030229. SAND84-1646. 30160:302.

The TRAN B series of experiments is being conducted at Sandia National Laboratories to investigate the characteristics of fuel removal/freezing through the upper axial blankets of an LMFBR during the transition phase of a hypothetical core disruptive accident. The third experiment in this series, TRAN B-3, was performed in February 1984, and the results are reported herein. This experiment involved the injection of molten UO(2) into an annular flow channel. Unlike the similar TRAN B-1 experiment, the initial steel wall temperature in B-3 was sufficiently high that instantaneous steel melting would occur upon contact with molten fuel. The earlier TRAN B-1 results had shown that fuel crusts are initially stable, both on the inside of a steel tube as well as on the outside of a steel rod, when no steel melting occurred. TRAN B-3 was designed to investigate this question of crust stability on surfaces of opposite curvature when surface melting did occur.

NUREG/CR-3945: FATIGUE CRACK GROWTH RATES OF LOW-CARBON AND STAINLESS PIPING STEELS IN PWR ENVIRONMENT. CULLEN,W.H. Materials Engineering Associates, Inc. February 1985. 65pp. 8502150048. MEA-2055. 28958:202.

Fatigue crack growth rates of A 106 Gr. C and A 516 Gr. 70 carbon steels, and A 351-CF8A stainless steel in PWR environments have been determined over a load ratio range (R) of 0.2 to 0.85, a temperature range of 93 degrees centigrade to 338 degrees centigrade, and a test frequency range of 17 mHz to 1 Hz using sinusoidal waveforms. In addition, growth rates have been determined for various orientations of the crack plane with respect to the product form. Crack growth rates in 288 degrees centigrade air environments have been measured in order to provide a reference baseline. These results define the magnitude of and major influences on the environmentally-assisted fatigue crack growth rates for these piping steels, and are supported by fractographic observations of the fatigue fracture surface.

NUREG/CR-3948: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY. MCCARDELL,R.K.; PLOGER,S.A.; MCCORMICK,R.D.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). September 1985. 78pp. 8510030433. EGG-2297. 32911:221.

Operational transients occur occasionally in light water reactors when minor malfunctions of certain system components affect the reactor core. This report presents the results of the operational transient Test OPTRAN 1-1 and OPTRAN 1-2, including a comparison of the data with posttest calculations and the postirradiation examination results. The OPTRAN1-1 tests simulated operational transients with reactor scram. Four progressively higher and broader power transients at a constant coolant flow rate were performed. The first transient simulated a BWR-5 turbine trip without steam bypass, with fuel rods operating near BWR-6 core average rod powers. The second transient simulated a generator load rejection without steam bypass, with fuel rods operating near core average powers. The last two transients were performed at higher core average peak rod powers than safety analyses predict to be possible in commercial reactors to define failure threshold margins. Test OPTRAN 1-2 was performed to evaluate the probability and extent of fuel rod damage for the most severe BWR anticipated transient without scram (ATWS) that results in boiling transition, a main steam line isolation valve closure transient without scram. Two sets of two fuel rods were tested. In each set, an unirradiated fuel rod was used to heat the coolant to typical BWR conditions for each previously irradiated fuel rod. Following an extensive fuel conditioning period of operation, a single power transient was performed that simulated the power history and coolant conditions calculated for a main steam line isolation valve closure ATWS.

NUREG/CR-3949 V01: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Semiannual Progress Report For Period Ending June 30,1984. DODD,C.V.; DEEDS,W.E.; SMITH,J.H.; et al. Oak Ridge National Laboratory. January 1985. 14pp. 8502210202. ORNL/TM-9339/V1. 29028:333.

Eddy-current inspection is the most suitable method for rapid boreside evaluation of steam generator tubing. However, small flaws can be masked by the effects of harmless variables, such as tube supports. To identify the critical properties accurately and reliably in the presence of extraneous signals caused by variations of unimportant properties, sufficient information is needed to identify harmful variations and to reject harmless ones. For this reason we are developing instrumentation capable of measuring both the amplitude and phase of the eddy-current signal at several different frequencies and computer equipment capable of processing that data quickly and reliably. Our probes and test conditions are also computer-optimized. The most recent probe design embodies an array of small flat "pancake" coils and improves the detection of small flaws and the rejection of tube support signals. We adapted our new IBM System 9000 computer to take and process the larger amounts of data required by additional variables, such as copper coating and intergranular attack. We also completed construction of the hand-wired versions of the 8- and 16-coil arrays and the multiplexing circuitry and computer codes to handle the data.

NUREG/CR-3949 V02: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Annual Progress Report For Period Ending December 31,1984. DODD,C.V.; DEEDS,W.E.; SMITH,J.H.; et al. Oak Ridge National Laboratory. August 1985. 16pp. 8509110016. ORNL/TM-9339/V2. 32564:288.

Eddy-current inspection is the most suitable method for rapid boreside evaluation of steam generator tubing. However, small flaws can be masked by the effects of harmless variables, such as tube supports. To identify the critical properties accurately and reliably in the presence of extraneous signals caused by variations of unimportant properties, sufficient information is needed to identify harmful variations and reject harmless ones. For this reason we have been developing instrumentation capable of measuring both the amplitude and phase of the eddy-current signal at several different frequencies, as well as computer equipment capable of processing the data quickly and reliably. Our probes and test conditions are also computer-optimized. The most recent probe design embodies an array of small flat "pancake" coils and improves the detection of small flaws and the rejection of tube support signals. We have also experimentally verified the accuracy of our computer programs for calculating the signals produced by defects in tubing and are adapting our new IBM System 9000 computer to take and process the larger amounts of data required by additional variables, such as copper coating and intergranular attack.

NUREG/CR-3950 V01: FUEL PERFORMANCE ANNUAL REPORT FOR 1983. BAILEY,W.J.; DUNENFELD,M.S. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1985. 123pp. 8503280020. PNL-5210. 29547:139.

This annual report, the sixth in a series, provides a brief description of fuel performance during 1983 in commercial nuclear power plants. Brief summaries of fuel design changes, fuel surveillance programs, fuel operating experience, fuel problems, high-burning fuel experience, and items of general significance are provided. References to additional, more detailed information and related NRC evauations are included.

NUREG/CR-3952: SEQUOYAH EQUIPMENT HATCH SEAL LEAKAGE. GREIMANN,L.; FANOUS,F.; BLUHM,D. Ames Laboratory, Energy & Mineral Resources Research Institute. July 1985. 49pp. 8508090579. IS-4862. 32105:296.

Nuclear containments which will not leak significantly, that is, beyond technical specifications, during a design accident may leak severely during a severe accident when the pressure increases beyond the design level. Small leaks which are visualized as occurring at local details may occur before complete vessel failure. Buckling of the hatch door, large deformations and ovaling of the hatch sleeve are potential causes of mismatch at the sealing surface which can result in a leakage path. As a typical example of steel containments the Sequoyah equipment hatch was selected. If penetrations effects are neglected, gross yielding of the 1/2-inch shell plate near the springline of the Sequoyah containment will occur at an internal pressure of between 50 and 60 psi. The results of a finite element analysis showed that a maximum of 0.9 inch of 85 to 90 psi, far above gross yielding of shell. Although buckling increased the relative seal motions, they remained sufficiently small to prevent leakage. The Sequoyah equipment hatch should not leak before strains of several percent develop in the 1/2-inch containment shell plate near the springline, which occurs between 50 and 60 psi. In the unlikely event of hatch buckling, postbuckling deformations would not introduce leakage.

NUREG/CR-3953: THE USE OF MAG-1 SPECTACLES WITH POSITIVE- AND NEGATIVE-PRESSURE RESPIRATORS. REED,K.A.; MOORE,T.O. Los Alamos Scientific Laboratory. May 1985. 41pp. 8506060798. LA-10229-MS. 30775:198.

Results of testing conducted at Los Alamos National Laboratory, Personnel Protection Studies Section, using MAG-1 spectacles in conjunction with positive- and negative-pressure full-facepiece respirators, are reported. The purpose of the three-phase study was to determine if the specially constructed strap of the MAG-1s affected the protection factors (PFs) of the respirators or the cylinder life of selected self-contained breathing apparatus (SCBA). The following respirators were tested with the MAG-1s: a) Phases I and II, positive-pressure full facepiece: Presur-Pak II SCBA (pressure-demand) Scottoramic facepiece, MSA 401 Air Mask Ultravue facepiece (medium), Survivair pressure-demand SCBA/silicone full facepiece, MSA powered air-purifying respirator/Ultravue facepiece (medium); b) Phase III, negative-pressure full facepiece: MSA Ultravue (small, medium, large), MSA Ultra-twin (small, medium, large), Norton Series 7600 (one size only). Statistical analysis and review of the test data from Phases I and II indicated little, if any, variation with and without the MAG-1s with most protection factors greater than 10,000. Test data also indicated little, if any, difference in the cylinder life with and without the MAG-1s, except the Scott Presur-Pak II SCBA used with the Scottoramic facepiece. Statistical analysis of the quantitative fit test data indicated no difference in PFs for the negative-pressure devices for the Ultravue negative-pressure respirator, but a significance at the 0.05 and 0.01 levels for the Ultra-twin and Norton full facepieces, respectively.

NUREG/CR-3954: HECTR ANALYSIS OF EQUIPMENT TEMPERATURE RESPONSES TO SELECTED HYDROGEN BURNS IN AN ICE CONDENSER CONTAINMENT. DANDINI,V.J.; MCCULLOCH,W.H. Sandia National Laboratories. February 1985. 135pp. 8503280013. SAND84-1704. 29547:001.

The temperature response of three generic surface models in each of three locations in an ice condenser containment building were calculated assuming a hydrogen deflagration event and using the HECTR code. The intent of using the three generic surfaces was to conservatively represent surfaces of various types of safety equipment. Analyses were performed for four accident sequence types with variations. The general observations drawn from these analyses are that (1) higher equipment surface temperatures than calculated for S(2)D type arrested sequences were calculated for other sequence types of comparable core melt frequency, and (2) surface temperatures greater than qualification temperatures were calculated to occur for some sequence types.

NUREG/CR-3972: SETTLEMENT OF URANIUM MILL TAILINGS PILES: A COMPARISON OF ANALYSIS TECHNIQUES. FAYER,M.J.; MCKEON,T.J. Battelle Memorial Institute, Pacific Northwest Laboratories. December 1984. 91pp. 8503040052. PNL-5222. 29198:274.

Two empirical methods of settlement analysis (Terzaghi's theory and a simplified version of the Fredlund-Morgenstern two-stress-state approach) were compared to the computer code TRUNC, a modified version of the TRUST code for variably saturated flow in deformable porous media. The three methods were used to predict settlement of a 12.2-m-deep pile of tailings slimes with a drain at the bottom. The simpler, empirical methods of settlement analysis were just as effective as TRUNC in predicting total settlement. For saturated tailings, predictions of total settlement by Terzaghi's theory and TRUNC were in close agreement (1.69 and 1.73 m, respectively). For partially saturated tailings, the simplified stress-state approach and TRUNC predicted similar total settlements (0.52 and 0.51 m, respectively). Terzaghi's theory, as applied, overestimated the time of settlement under saturated conditions (170 days versus 140 days predicted by TRUNC) because it did not account for gravitational gradients. No empirical or analytical means were available to predict the time of settlement under partially saturated conditions. If the magnitude of partially saturated settlement is considered significant, then the time over which it occurs will most likely be the deciding factor in determining when to place the cover on the tailings pile.

NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR. FLETCHER,C.D.; BOLANDER,M.A.; WATERMAN,M.E.; et al. EG&G, Inc. April 1985. 233pp. 8506140624. EGG-2341. 30934:352.

Thermal-hydraulic analyses of fourteen hypothetical pressurized thermal shock (PTS) scenarios for the H. B. Robinson, Unit 2 pressurized water reactor were performed at the Idaho National Engineering Laboratory (INEL) using the RELAP5 computer code. The scenarios, which were developed at Oak Ridge National Laboratory (ORNL), contain significant conservatisms concerning equipment failures, operator actions, or both. The results of the thermal-hydraulic analyses presented here, along with additional analyses of multidimensional and fracture mechanics effects, will be utilized by ORNL, integrator of the PTS study, to assist the U.S. Nuclear Regulatory Commission in resolving the pressurized thermal shock unresolved safety issue.

NUREG/CR-3978: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL PLATE AND WELDS FOR THE FOURTH HSST IRRADIATION SERIES. MCGOWAN,J.J. Oak Ridge National Laboratory. January 1985. 25pp. 8503110098. ORNL/TM-9516. 29328:206.

The Heavy Section Steel Technology (HSST) program office conducted a series of experiments to determine the effect of neutron irradiation on the fracture toughness of nuclear pressure vessel materials. One plate (HSST plate 02) and four welds of A533 grade B class 1 steel were examined. The welds were made by current (about 1979) practice. As part of this study, tensile properties were measured after irradiation to 2 x 10(23) neutrons/m(2) (>1 MeV) at 288 degrees C. The strength of all four welds increased with irradiation. Yield strength was about 10% more sensitive to irradiation than was ultimate strength. Tensile ductility was not affected significantly by irradiation.

NUREG/CR-3980 V02: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,April-June 1984. REST,J. Argonne National Laboratory. February 1985. 36pp. 8503290274. ANL-84-61 V02. 29563:354.

This progress report summarizes the Argonne National Laboratory work performed during April, May, and June 1984 on water reactor safety problems related to fuel and fuel cladding materials. The research and development areas covered are Transient Fuel Response and Fission Product Release and Clad Properties for Code Verification.

NUREG/CR-3980 V03: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,July-September 1984. REST,J. Argonne National Laboratory. May 1985. 51pp. 8507050429. ANL-84-61. 31371:306.

This progress report summarizes the Argonne National Laboratory work performed during July, August, and September 1984 on water reactor safety problems related to fuel and fuel cladding materials. The research and development areas covered are Transient Fuel Response and Fission Product Release and Clad Properties for Code Verification.

NUREG/CR 3980 V04: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,October-December 1984. CHUNG,H.M.; REST,J. Argonne National Laboratory. September 1985. 63pp. 8510040332. ANL-84-61. 32856:118.

This progress report summarizes the Argonne National Laboratory work performed during October, November, and December 1984 on water reactor safety problems related to fuel and fuel cladding materials. The research and development areas covered are Transient Fuel Response and Fission Product Release and Clad Properties for Code Verification.

NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH. KAHN,B.; TURGEON,K.S.; MARTINI,D.K.; et al. Georgia Institute of Technology, Atlanta, GA. February 1985. 170pp. 8502210154. 29051:001.

Bluegill and catfish maintained in flow-through tanks were fed P-32 at two feeding levels. Fish were analyzed in triplicate for P-32 and phosphorus at intervals of 1 - 8 days. Additional aquaria experiments were performed to determine the effects of other factors and to observe P-32 uptake from water by unfed fish (including fish with blocked esophagus). The bluegill showed a weight gain of 0.2 %/d, a phosphorus turnover constant in muscle of 0.43 %/d, and a BF(r)/BF ratio of 0.081 at the higher feeding rate, and 0.05 %/d, 0.34 %/d, and 0.064 at the lower feeding rate. Hence, respective P-32 BF(r) values are 6,000 and 4,000 at a phosphorus BF of 70,000. The BF(r) values for catfish were approximately twice as high. The aquarium experiments suggest that the higher factors are due to a much higher phosphorus intake, higher water temperature, higher retention from pellets than from worms, and possible higher retention by catfish than bluegill under the same conditions.

NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984. EIDSON,A.F. Inhalation Toxicology Research Institute. January 1985. 21pp. 8501230614. LMF-11. 28536:228.

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 uranium distribution patterns among organs are compared. These studies will facilitate calculations of organ doses for potential exposures and will identify important bioassay procedures. Results of studies in rats to investigate retention of yellowcake in a wound showed that retention of less soluble yellowcake from the body was significantly more prolonged than of more soluble yellowcake. However, retention could not be quantitatively related to the chemical composition or in vitro dissolution behavior of the implanted powder. Studies of Beagle dogs following nose-only inhalation of aerosols of commercial yellowcake were continued. Histological observations showed kidney damage that appeared 4 to 8 days after exposure to the more soluble yellowcake with repair occurring by 64 days after exposure. The concentration of uranium in kidney was 8-17 mg U/g kidney at 4-8 days after exposure. No evidence of kidney damage was observed in dogs exposed to the less soluble yellowcake form.

NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS. RANKIN,W.L.; RIDEOUT,T.B.; TRIGGS,T.J.; et al. Battelle Human Affairs Research Centers. June 1985. 102pp. 8507050435. PNL-5158. 31371:001.

This report presents the design philosophy and associated functional criteria and design principles for developing advanced computerized annunciator systems for use in the control rooms of nuclear power plants. The scope of the work includes advanced system recommendations that could be incorporated into operating nuclear power plants. The information contained in this report was obtained from a review of the revelant computer and visual display terminal literature, from site visits to advanced control rooms in the nuclear power and related industries, and from a study of technical reports on computerized control rooms. This report should assist the staff in development of a regulatory position regarding the design of computerized control room annunciator systems. The guidance in this report is consistent with that provided in NUREG-0700.

NUREG/CR-3989: TIME- AND VOLUME-AVERAGED CONSERVATION EQUATIONS FOR MULTIPHASE FLOW.Part One:System Without Internal Solid Structures. SHA,W.T.; CHAO,B.T.; SOO,S.L. Argonne National Laboratory. February 1985. 127pp. 8503110328. ANL-84-66. 29328:001.

A set of rigorously derived conservation equations of mass, momentum, and energy for multiphase systems without internal solid structures via time-volume averaging of point, instantaneous conservation equations is presented. These equations are differential-integral equations in which the area integrals account for the interfacial transport of mass, momentum, and energy. The equations from volume averaging contain averages of the product of the dependent variables which must be expressed in terms of the products of their averages. In nonturbulent flows, this is achieved by expressing the "point" variables as the sum of its intrinsic volume average and a spatial deviation. In turbulent flows for which further time-averaging is required, the "point" variable is then decomposed into a low-frequency component and a high-frequency component. Time averaging following volume averaging preserves the identity of the dynamic phases. Under certain simplifying conditions, the proposed set of rigorously derived conservation equations reduces closely to various forms that are currently "accepted" for two-phase flow analysis. This set of conservation equations serves as a reference point for modeling multiphase flow and provides theoretical guidance and physical insight that may be useful to develop correlations for quantifying interfacial transport of mass, momemtum, and energy.

NUREG/CR-3990: CHARCOAL PERFORMANCE UNDER ACCIDENT CONDITIONS IN LIGHT-WATER REACTORS. DEITZ,V.R. Navy, Dept. of, Naval Research Lab. March 1985. 190pp. 8504040424. 5528. 29628:058.

Nuclear-grade carbons were systematically degraded by exposure to unfiltered outdoor air with decrease in radioactive methyl iodide trapping. Local meteorological conditions of high humidity combined with atmospheric pollutants in the test vicinity contributed jointly to the degradation. When service carbons were exposed to radiation levels of 10(7) to 10(9) rads, the iodine isotope exchange capacity was regenerated. The adsorptive properties were only slightly improved. It was possible to regenerate the iodine isotope-exchange efficiencies by reaction with airborn chemical reducing agents such as hydrazine for carbons removed from nuclear power operations. The depth profile in methyl iodide-131 penetration changed from simple exponentional through new carbons to a non-linear profile for weathered and service aged carbons. The behavior is attributed to the chromatographic distribution of the contaminants that accumulate in the bed. The removal of radioactive iodine depends on a minimum of 3 distinguishable processes: adsorption on the activated carbon, iodine isotope exchange with impregnated iodine-127, and chemical combination with impregnated tertiary amines when present. When a carbon is new, all 3 mechanisms are at peak performance. After the carbon is placed in service, the 3 mechanisms degrade at different rates; the adsorption process degrades faster than the others.

NUREG/CR-3991: FAILURE MODES AND EFFECTS ANALYSIS (FMEA) OF THE ICS/NNI ELECTRIC POWER DISTRIBUTION CIRCUITRY AT THE OCONEE-1 NUCLEAR PLANT. * Oak Ridge National Laboratory. MCBRIDE,A.F.; MAYO,C.W.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). October 1985. 96pp. 8512270363. ORNL/TM-9383. 34078:185.

The effects of nonnuclear instrumentation (NNI) and integrated control system (ICS) electric power supply failures have been analyzed for the Oconee Unit 1 nuclear plant. The instrument and control system power distribution circuits were analyzed to define a comprehensive set of 19 single-point failure modes. For each power supply failure, the failed and operating control system signal inputs were propagated through the partially energized control system circuits as well as the energized and deenergized output control devices to evaluate the initial plant response. In addition, the effects of the power supply failures on the principal control room parameter displays were combined with the initial plant response to the automatic control circuits to evaluate possible control room operator responses. Plant responses to the defined power supply failures are described in detail. The automatic responses of the plant to the instrument and control system power supply failures were not found to be severe. Possible operator responses to spurious control room displays generally did not result in significant transients. Improved automatic transfer of control system input circuits to operable power supplies, automatic trip of feedwater pumps on loss of certain power supply branch circuits, and suppression of spurious alarms have been identified as possible ways to further limit the effects of transients.

NUREG/CR-3992: COLLECTION AND EVALUATION OF COMPLETE AND PARTIAL LOSSES OF OFF-SITE POWER AT NUCLEAR POWER PLANTS. BATTLE,R.E. Oak Ridge National Laboratory. February 1985. 63pp. 8502280542. ORNL/TM-9384. 29169:202.

Events involving loss of off-site power that have occurred at nuclear power plants through 1983 are described and categorized as complete or partial losses. The events were identified as plant-centered or grid-related failures. In addition, the causes of the failures were classified as weather, human error, design error, or hardware failure. The plant-centered failures were usually of shorter duration than the weather-related grid failures. For this reason, the weather-related events were reviewed in detail. Design features that may be important factors affecting off-site power system reliability were tabulated for most of the operating nuclear power plants. The tabulated information was provided to NRC for a statistical analysis to determine the importance of these design features for losses of off-site power. The frequency of losses of off-site power versus duration were estimated for three time periods. The frequency of loss of off-site power was estimated to be 0.09/reactor-year based on industry-wide data for the years 1959 through 1983.

NUREG/CR-3998 V02: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,April-June 1984. SHACK,W.J. Argonne National Laboratory. April 1985. 92pp. 8504220361. ANL-84-60. 29946:069.

This progress report summarizes the Argonne National Laboratory work performed during April, May, and June 1984 on water reactor safety problems related to out-of-core materials. The research and development areas covered are Environmentally Assisted Cracking in Light Water Reactors, Long-Term Embrittlement of Cast Duplex Stainless Steels in LWR Systems, and Nondestructive Evaluation and Leak Detection.

NUREG/CR-3998 V03: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,October-December 1984. SHACK,W.J. Argonne National Laboratory. October 1985. 75pp. 8512120141. ANL-84-60. 33873:039.

This progress report summarizes the Argonne National Laboratory work performed during October, November and December 1984 on water reactor safety problems related to out-of-core materials. The research and development areas covered are Environmentally Assisted Cracking in Light Water Reactors and Long-Term Embrittlement of Cast Duplex Stainless Steels in LWR Systems.

NUREG/CR-3999: ELECTRICALLY HEATED EX-REACTOR PELLET-CLADDING INTERACTION (PCI) SIMULATIONS UTILIZING IRRADIATED ZIRCALOY CLADDING. BARNER,J.O.; FITZSIMMONS,D. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1985. 104pp. 8503120445. PNL-5245. 29340:190.

In a program sponsored by the Fuel Systems Research Branch of the U.S. Nuclear Regulatory Commission, a series of six electrically heated fuel rod simulation tests were conducted at Pacific Northwest Laboratory primarily to determine the susceptibility of irradiated pressurized-water reactor Zircaloy-4 cladding to failures caused by pellet-cladding mechanical interaction (PCMI). A secondary objective was to acquire kinetic data (e.g., ridge growth or relaxation rates) that might be helpful in the interpretation of in-reactor performance results and/or the modeling of PCMI. No cladding failures attributable to PCMI occurred during the six tests. This report describes the testing methods, testing apparatus, fuel rod diametral strain measuring device, and test matrix. Test results are presented and discussed.

NUREG/CR-4003: CLOSEOUT OF IE BULLETIN 79-04:INCORRECT WEIGHTS FOR SWING CHECK VALVES MANUFACTURED BY VELAN ENGINEERING CORPORATION. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. Parameter, Inc. June 1985. 29pp. 8507030702. IE-143. 31314:067.

IE Bulletin 79-04 was issued March 30, 1979 as a result of reports from three facilities that Velan Engineering Corporation had provided incorrect weights for swing check valves. The reason for concern was the possibility that these incorrect weights had been used in analyses of Seismic Category I piping systems at a large number of plants. Evaluation of utility responses and NRC/IE inspection reports shows that the bulletin can be closed out for 117 (92%) of the 127 current facilities on the basis of specific criteria. Followup items for the remaining 10 current facilities are proposed for use by NRC/IE. Incorrect weights reported for valves other than Velan swing check valves are identified as Remaining Areas of Concern. This bulletin has served its purpose and can be closed out. A final check of valve weights will be made per later IE Bulletin 79-14 on seismic analyses for as-built safety-related piping systems.

NUREG/CR-4004: CLOSEOUT OF IE BULLETIN 79-25:FAILURES OF WESTINGHOUSE BFD RELAYS IN SAFETY-RELATED SYSTEMS. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. Parameter, Inc. April 1985. 44pp. 8505010088. IEB-79-25. 30112:260.

Robinson 2 submitted LER 78-29 December 19, 1978 to report sticking of a normally energized Westinghouse BFD relay. After reviewing this problem, Westinghouse issued Service Letter TS-E-412 to recommend that BFD relays in safety-related systems be replaced with later NBFD relays. During installation and testing of the new NBFD relays, Robinson 2 found some with marginal or unsatisfactory armature overtravel. Because of this new problem, Westinghouse issued Technical Bulletin NSD-TB-79-05 to recommend prompt checking of certain models of NBFD relays and returning those with inadequate overtravel for rework or replacement. IE Bulletin 79-25, with extracts of the Westinghouse service letter and technical bulletin enclosed, was issued November 2, 1979 to require responses and specific actions by all licensees and holders of construction permits with respect to BFD and NBFD relays in safety-related systems. Evaluation of utility responses and NRC/IE inspection reports indicates that the bulletin can be closed out for 121 (94%) of the 129 current facilities on the basis of specific criteria. Proposed followup items for the remaining 8 facilities are presented in Appendix C for use by NRC/IE. Because followup of corrective action is ensured, IE Bulletin 79-25 is considered closed.

NUREG/CR-4005: CLOSEOUT OF IE BULLETIN 80-12:DECAY HEAT REMOVAL SYSTEM OPERABILITY. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. Parameter, Inc. June 1985. 59pp. 8507030675. IE-146. 32106:149.

On April 19, 1980, decay heat removal (DHR) capability was lost at Davis-Besse 1 for approximately two and one-half hours in a refueling mode. Typically for that mode, many systems and components were out of service for maintenance and testing or were deactivated to preclude inadvertent actuation. IE Bulletin 80-12 was issued May 8, 1980 for action by licensees of operating pressurized water reactors (PWRs); it was issued for information to nuclear power facilities other than operating PWRs. The intent of the bulletin was to improve nuclear plant safety by reducing the likelihood of losing DHR capability in PWRs, especially when some DHR components are unavailable because of maintenance activities during refueling and cold shutdown modes of operatic i. A related NRR Generic Letter was issued June 11, 1980 to licensees of operating PWRs, requesting amendment of technical specifications to ensure long-term maintenance of DHR capability. Evaluation of utility responses and NRC/IE inspection reports indicates that the bulletin can be closed out per specific criteria for 33 (75%) of the 44 affected facilities.

NUREG/CR-4006: CLOSEOUT OF IE BULLETIN 81-01:SURVEILLANCE OF MECHANICAL SNUBBERS. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. Parameter, Inc. August 1985. 82pp. 8508260307. IE-145. 32368:219.

In the period from August 1974 to May 1980, failures of mechanical snubbers were described in event reports issued for nine facilities and in a NRC/IE study of the DOE Fast Flux Test Facility. In most failures, the snubbers were frozen and would not permit free piping motions during thermal transients. In some cases, the failed snubbers no longer provided seismic shock restraint. Because of concern about the reported failures of mechanical snubbers, standard technical specification revisions for snubber surveillance were issued by NRC/DL on November 20, 1980. IE Bulletin 81-01 was issued January 27, 1981 to require examination and testing of mechanical snubbers in safety-related systems at licensed facilities and at selected facilities under construction. Evaluation of utility responses and NRC/IE inspection reports indicates that the bulletin can be closed out per specific criteria for 73 (95%) of the 77 facilities to which it was issued for action. Followup items are proposed for use by NRC/IE to ensure satisfactory completion of corrective action at the remaining four (4) facilities.

NUREG/CR-4008: GENERAL EXTRAPOLATION MODEL FOR AN IMPORTANT CHEMICAL DOSE-RATE EFFECT. GILLEN,K.T.; CLOUGH,R.L. Sandia National Laboratories. January 1985. 51pp. 8503010341. SAND84-1948. 29186:186.

In order to extrapolate material accelerated aging data, methodologies must be developed based on sufficient understanding of the processes leading to material degradation. One of the most important mechanisms leading to chemical dose-rate effects in polymers involves the breakdown of intermediate hydroperoxide species. A general model for this mechanism is derived based on the underlying chemical steps. The results lead to a general formalism for understanding dose rate and sequential aging effects when hydroperoxide breakdown is important. We apply the model to combined radiation/temperature aging data for a PVC material and show that this data is consistent with the model and that model extrapolations are in excellent agreement with 12-year real-time aging results from an actual nuclear plant. This model and other techniques discussed in this report can aid in the selection of appropriate accelerated aging methods and can also be used to compare and select materials for use in safety-related components. This will result in increased assurance that equipment qualification procedures are adequate.

NUREG/CR-4009: HUMAN RELIABILITY DATA BANK.Evaluation Results. COMER,M.K.; DONOVAN,M.D.; GADDY,C.D.; et al. General Physics Corp. April 1985. 75pp. 8505070505. SAND85-7150. 30210:295.

The U.S. Nuclear Regulatory Commission and Sandia National Laboratories conducted a three-year research program to develop a human reliability data bank specifically tailored to support human reliability analysis segments of probabilistic risk assessments for nuclear power plants. Previous efforts of the program include a review of existing human performance data banks (NUREG/CR-2744, Vol. 1) and a concept and system description (NUREG/CR-2744, Vol. 2). Subsequent to the system description, a detailed specification for implementing the data bank was developed. An evaluation of this specification was conducted and is described in this report. The evaluation consisted primarily of an Operability Demonstration and Evaluation of the data processing procedures and personnel required, and a Data Review Demonstration and Evaluation involving members of the potential user population. The conclusions of this study were used to modify and improve the detailed implementation specification. The revised specification is published as NUREG/CR-4010, and it describes all the necessary materials, personnel, procedures, definitions, and data taxonomies to implement the data bank.

NUREG/CR-4010: SPECIFICATION OF A HUMAN RELIABILITY DATA BANK FOR CONDUCTING HRA SEGMENTS OF PRAS FOR NUCLEAR POWER PLANTS. COMER,M.K.; DONOVAN,M.D. General Physics Corp. * Sandia National Laboratories. April 1985. 400pp. 8505060499. SAND85-7151. 30190:331.

This document specifies the personnel, resources, policies, and procedures for implementing and operating a human reliability data bank specifically tailored to support human reliability analysis (HRA) segments of probabilistic risk assessments (PRAs) for nuclear power plants. The report concludes a three-year research program conducted by the U.S. Nuclear Regulatory Commission and Sandia National Laboratories. Previous efforts of the program include a review of existing human performance data banks (NUREG/CR-2744, Vol. 1), a concept and system description (NUREG/CR-2744, Vol. 2), and a peer evaluation study (NUREG/CR-4009). This report specifies the administrative organization of the data bank functional groups and their proposed interaction. Detailed procedures are included that specify how to process submitted data, how to classify and store the data, and how to combine similar data when appropriate. Included within the report is the skeleton data manual, which is a prototype, hardcopy, data manual that would be used to disseminate data to the user population. It describes the data taxonomy, procedures for retrieving data of interest, and presents several sample data retrieval problems. Definitions are supplied for all technical and behavioral terms used in the taxonomic structure. As its name implies, the skeleton data manual embodies the data manual in structure, but is void of emperical data.

NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1. CORWIN,W.R.; ROBINSON,G.C.; NANSTAD,R.K.; et al. Oak Ridge National Laboratory. April 1985. 103pp. 8505230524. ORNL/TM-9390. 30549:090.

The HSST stainless steel cladding evaluations were initiated to study the interaction of stainless cladding with flaws initiated in and propagating in base metal of reactor pressure vessels. A complicating factor in understanding the role of stainless cladding in this setting is its toughness as a function of radiation dose and fabrication process. The initial phase of this study addressed this question by testing the response of specimens clad with single-wire submerged-arc weld overlay in varying toughness levels. The tests completed under the initial phase of this study indicate that cladding of moderate Charpy toughness has only limited capabilities to stop running cracks. This was a limited set of experiments, and the upper and lower bounds of cracks arrest capabilities are not yet determined. The fabrication techniques employed for this first series of tests have resulted in conditions that have prevented close control of the stress state at pop-in of the hydrogen-charged EB welds. Consequently, the arrest toughness of the stainless cladding was not closely bounded. General modifications are proposed for incorporation in a second series of tests to provide more comprehensive conditions of testing and materials of interest, to eliminate some undesirable test conditions that existed in the first series, and to provide an improved geometry for analytical interpretations.

NUREG/CR-4016 V01: APPLICATION OF SLIM-MAUD:A TEST OF AN INTERACTIVE COMPUTER-BASED METHOD FOR ORGANIZING EXPERT ASSESSMENT OF HUMAN PERFORMANCE AND RELIABILITY.Volume I:Main Report. ROSA,E.A.; HUMPHREYS,P.C.; SPETTELL,C.M.; et al. Brookhaven National Laboratory. September 1985. 54pp. 8512270232. BNL-NUREG-51828. 34084:323.

The U.S. Nuclear Regulatory Commission (NRC) has been conducting a mutli-year research program to investigate different methods for using expert judgments to estimate human error probabilities (HEPs) in nuclear power plants. One of the methods investigated, derived from multi-attribute utility theory, is the Success Likelihood Index Methodology implemented through Multi-Attibute Utility Decomposition (SLIM-MAUD). This report describes a systematic test application of the SLIM-MAUD methodology. The test application is evaluated on the basis of three criteria: practicality, acceptability, and usefulness. Volume I of this report presents an overview of SLIM-MAUD, describes the procedures followed in the test application, and provides a summary of the results obtained. Volume II consists of technical appendices to support in detail the materials contained in Volume I, and the user's package of explicit procedures to be followed in implementing SLIM-MAUD. The results obtained in the test application provide support for the application of SLIM-MAUD to a wide variety of applications requiring estimates of human errors.

NUREG/CR-4020: HMS:A COMPUTER PROGRAM FOR TRANSIENT,THREE-DIMENSIONAL MIXING GASES. TRAVIS,J.R. Los Alamos Scientific Laboratory. February 1985. 52pp. 8503210474. LA-10267-MS. 29479:114.

A numerical technique has been developed for calculating the full three-dimensional time-dependent equations of motion with multiple species transport. The method is a modified form of the Implicit Continuous-fluid Eulerian (ICE) technique to solve the governing equations for low Mach number flows where pressure waves and local variations in compression and expansion are not significant. Large density variations, due to thermal and species concentration gradients, are accounted for without the restrictions of the classical Boussinesq approximation. Example calculations of the EPRI/HEDL standard problems verify the feasibility of using this finite-difference technique for analyzing hydrogen transport and mixing within LWR containments.

NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT. BALL,D.G.; CHEVERTON,R.D.; FLANAGAN,G.F.; et al. Oak Ridge National Laboratory. September 1985. 702pp. 8512270369. ORNL/TM-9408. 34074:038.

An evaluation of the risk of pressurized thermal shock (PTS) resulting in a through-the-wall crack in a reactor pressure vessel was performed for the Calvert Cliffs Unit 1 nuclear power plant. The information presented in this report covers one of three plant-specific studies performed for NRC. The other two studies, for Oconee Unit 1 and H. B. Robinson Unit 2, are documented in NUREG/CR-3770 and NUREG/CR-4183, respectively. The specific objectives of the Calvert Cliffs study were (1) to further refine the methodology for evaluating the risk of PTS, (2) to provide a best estimate of the frequency of a through-the-wall crack for the Calvert Cliffs Unit 1 vessel, (3) to determine the dominant PTS sequences for the unit, and (4) to evaluate the effectiveness of potential corrective measures. The examination of tens of thousands of transients indicated that PTS was not an important core melt initiator for Calvert Cliffs Unit 1. The dominant risk sequences were determined to be small-break LOCAs at low core decay-heat conditions which led to total loop flow stagnation.

NUREG/CR-4023: FIELD PERFORMANCE ASSESSMENT OF SYNTHETIC LINERS FOR URANIUM TAILINGS POND.A Status Report. MITCHELL,D.H.; SPANNER,G.E. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1985. 78pp. 8502040788. PNL-5005. 28716:096.

This report presents the status of Pacific Northwest Laboratory's program through the end of FY-83 assessing the performance of synthetic liners used in uranium tailings ponds. Synthetic liner failure mechanisms, impoundment design, installation, and inspection techniques are presented from information collected from consultants, mill operators, and the synthetic liner industry. Progress is reported on laboratory tests on accelerated aging of liners, physical properties of aged materials, and nondestructive examination of seams.

NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER.(Final Report) FRUCHTER,J.S.; COWAN,C.E.; ROBERTSON,D.E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1985. 56pp. 8504030419. PNL-5299. 29605:123.

For the past several years, data on radionuclide migration in ground water at a low-level disposal site were collected. Most of the radionuclides were removed in the disposal basin and trench by either precipitation or adsorption mechanisms. However, three radionuclides (60)Co, (106)Ru, and (125)Sb showed somewhat greater than expected mobility. The elements of these three isotopes were found to be in either anionic or nonionic charge-forms. Complexes with both natural and man-made organics were implicated in the increased mobility, particularly in the case of (60)Co. Characterization studies of the organic fraction were performed. Ruthenium-103, (60)Co, and (125)Sb were found to be associated with the higher molecular weights greater than 1000. Studies were also performed that proved the hypothesis that the adsorption behavior of (235)Np on soils of the site is dominated by adsorption on iron hydroxide. Finally, geochemical modeling of the chemical and charge form data showed the ground water to be in equilibrium with several solids that could be important in controlling the concentrations of trace elements and radionuclides.

NUREG/CR-4031 V02: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutronics Calculations." WILLIAMS,L.; REMEC,I.; KAM,F.B. Oak Ridge National Laboratory. May 1985. 42pp. 8505170010. ORNL/TM-9423/V2. 30484:001.

A series of calculations has been completed to compute dosimeter activation in the Oak Ridge Research Reactor (ORR) HSST Simulator Experiment. A comparison of calculated and experimental results shows that calculations underpredict dosimeter activities on the average of about 15%. The C/E values indicated the now familiar tendency to become lower as more iron is penetrated. The dosimeters in front of the simulator (and behind the thermal shield) typically have C/E values of 0.9-1.0, while those at the back of the simulator have values of 0.7-0.85. The calculations also show shifted axial distribution relative to the measurements: the C/E values near the bottom of the core are about 15% to 20% higher than those near the top. This is probably due to a discrepancy in the axial power distribution computed with VIPOR/VENTURE. The axial distribution of fuel obtained from the correlation in VIPOR could possibly be causing the power shift, although this speculation has not been verified. There is also, perhaps, a slight tendency for the C/E values to increase along the x (the coordinate parallel to the reactor face) traverse from the reactor centerline to a point near the core boundary; however, this variation is much less than in the axial direction. The results obtained with different dosimeters appear generally to be reasonably consistent, except that the (46)Ti C/E values seem to be consistently lower than for the other dosimeters. The systematic nature of the discrepancies in these calculations will be adjusted by the least squares procedure to produce an accurate representation of the flux distribution.

NUREG/CR-4031 V03: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutron Exposure Parameters." REMEC,I.; STALLMANN,F.W.; KAM,F.B. Oak Ridge National Laboratory. May 1985. 31pp. 8505160647. ORNL/TM-9423/V3. 30457:181.

This is the third volume of a three-volume report which describes the simulator experiments of the fifth series of HSST irradiation experiments which are sponsored by the U.S. Nuclear Regulatory Commission (NRC). The purpose of these three volumes is to document, in detail, the experimental and calculational methodology which will be used in determining the neutron-exposure parameters for the fifth and subsequent series of HSST irradiation experiments at ORNL. The methodolgy was also used in the fourth series of HSST irradiation experiments and represents the current state-of-the-art procedures developed in the Light Water Reactor Pressure Vessel Simulation Project which is a part of NRC's Surveillance Dosimetry Improvement Program. The neutron-exposure data from the fifth and subsequent series will be documented in a loose-leaf NUREG/CR report as the data become available. In this volume, the best estimates for the values and spatial distribution of fluence rate (0) (E > 1.0 MeV), fluence rate (0) (E > 0.1 MeV), and displacements per atom per second (dpa/s) are determined using LSL-M2, a least squares logarithmic spectrum adjustment procedure with input values taken from dosimetry data from Vol. 1 and neutronics calculations from Vol. 2. These estimates have an overall uncertainty of less than 20% relative standard deviation. This volume is essential to the metallurgist for defining the irradiation strategy to meet his objective(s).

NUREG/CR-4033: THE ROLE OF PERSONAL AIR SAMPLING IN RADIATION SAFETY PROGRAMS AND RESULTS OF A LABORATORY EVALUATION OF PERSONAL AIR-SAMPLING EQUIPMENT. RITTER,P.D.; HUNTSMAN,B.L.; NOVICK,V.J.; et al. EG&G, Inc. May 1985. 80pp. 8505230534. EGG-2352. 30549:256.

Recommended applications for personal air sampling in NRC licensee radiation protecting programs are presented. The recommendations are based on performance tests of currently available samplers, a review of research and regulatory literature, and a survey of current licensee air-sampling programs. The performance tests show that personal air samplers are available which can provide a reliable, convenient means for breathing-zone sampling of workers in practically any work environment which might be encountered in the licensee industries. The research literature emphasized that estimates of an individual's exposure may be greatly underestimated if based on general area air samples, as is common practice in current licensee programs, due to the unpredictable variability of airborne-activity concentrations in the worksite. A conclusion which may be drawn from the literature and from experimental results is that in most situations, personal air sampling (or more generally, true breathing-zone sampling) is the only means to reliably estimate the airborne activity to which a worker has been exposed (MPC h). Research concerning the applicability of air-sampling measurements for estimating intake, uptake, and internal dose was also reviewed.

NUREG/CR-4035: A HIGHWAY ACCIDENT INVOLVING RADIOPHARMACEUTICALS NEAR BROOKHAVEN,MISSISSIPPI ON DECEMBER 3,1983. MOHR,P.B.; MONT,M.E.; SCHWARTZ,M.W. Lawrence Livermore National Laboratory. April 1985. 52pp. 8505070560. UCRL-53587. 30210:169.

A rear-end collision occurred between a passenger automobile and a luggage trailer carrying 84 packages, 76 of which contained radiopharmaceuticals, on U.S. Highway 84 near Brookhaven, Mississippi on the afternoon of December 3, 1983. The purpose of this report is to document the mechanical circumstances of the accident, confirm the nature and quantity of radioactive materials involved, and assess the nature of the physical environment to which the packages were exposed and the response of the packages. The report consists of three major sections. The first deals with the nature and circumstances of the accident and findings of fact. The second gives an accounting and description of the materials involved and the consequences of their exposure. The third gives an assessment and analysis of the mechanisms of damage and the conclusions which may be drawn from the investigation.

NUREG/CR-4037: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-5. OSBORNE,M.F.; COLLINS,J.L.; LORENZ,R.A.; et al. Oak Ridge National Laboratory. July 1985. 75pp. 8508090712. ORNL/TM-9437. 32104:055.

The fifth in a series of high-temperature fission product release tests was conducted for 20 min at 1700 degrees centigrade in flowing steam. The test specimen, a 15.2-cm-long section of a fuel rod which had been irradiated to a burnup of 38.3 MWd/kg, was heated in an induction furnace under simulated LWR accident conditions in a hot cell. Posttest inspection showed severe oxidation and fragmentation of the fuel specimen, but no cladding melting was apparent. Analyses of test components showed total releases from the fuel of 19.9% for (85)Kr, 22.4% for (129)I, 18.0% for (110m)Ag, and 20.3% for (137)Cs. A smaller fraction of the (125)Sb (0.326%) was released from the fuel, and 99% of the (110m)Ag and (125)Sb was retained in the furnace. Posttest analysis of the fuel specimen indicated a (134)Cs release of 24.5%, which is reasonably good agreement with the (137)Cs data. These releases were less than half those in test HI-2, where more oxidation and a large axial crack probably were significant factors in the release of fission products.

NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE. KOCHER,D.C.; WARD,R.C.; KILLOUGH,G.G.; et al. Oak Ridge National Laboratory. May 1985. 247pp. 8507250201. ORNL-6114. 31793:025.

This report presents a study of the sensitivity of reactor accident consequences predicted by the CRAC2 computer code to uncertainties in selected models and parameters used in the code. The sources of uncertainty that were investigated include (1) the model for plume rise, (2) the model for wet deposition, (3) the meteorological bin-sampling procedure for selecting weather sequences involving rain, (4) the dose conversion factors for inhalation as they are affected by uncertainties in the physical and chemical form of the released radionuclides, (5) the weathering half-time for external ground-surface exposure, and (6) the transfer coefficients for terrestrial foodchain pathways. The most important sources of uncertainty in our analyses were the choice of wet-deposition model, the dose conversion factors for inhalation, and the weathering half-time for ground-surface exposure. The choice of plume-rise model, the use of an alternative bin-sampling procedure, and uncertainties in terrestrial foodchain pathways usually had insignificant effects on CRAC2 prediction.

NUREG/CR-4039: GAMMA-RAY CHARACTERIZATION OF THE TWO-YEAR IRRADIATION EXPERIMENT PERFORMED AT THE POOLSIDE FACILITY. MAERKER,R.E. Oak Ridge National Laboratory. January 1985. 21pp. 8502220412. ORNL/TM-9440. 29073:230.

Average gamma-ray group fluence rates are calculated for each of the three exposures in the two-year metallurgical blind test experiment at the ORR-Poolside Facility in Oak Ridge, thus completing the characterization of the radiation field for this experiment, which is intended to serve as an international metalurgical benchmark. Heating rates in the steel derived from these calculations varied from about 0.23 watts/gram in the simulated surveillance capsule to 1.4 milliwatts/gram at the three-quarters depth location in the simulated pressure vessel capsule, with secondaries arising from non-fission reactions in the core and ex-core steel contributing between seventy-seven and ninety-three percent of the total. Contributions from photofission to fission foil activities are estimated to be less than five percent of those previously calculated arising from neutron-induced fission.

NUREG/CR-4040: OPERATIONAL DECISIONMAKING AND ACTION SELECTION UNDER PSYCHOLOGICAL STRESS IN NUCLEAR POWER PLANTS. GERTMAN,D.I.; JENKINS,J.P.; HANEY,L.N.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1985. 68pp. 8507020109. EGG-2387. 31313:236.

An extensive review of literature on individual and group performance and decisionmaking under psychological stress was conducted and summarized. Specific stress-related variables relevant to reactor operation were pinpointed and incorporated in an experiment to assess the performance of reactor operators under psychological stress. The decisionmaking performance of 24 reactor operators under differing levels of workload, conflicting information, and detail of available written procedures was assessed in terms of selecting immediate, subsequent, and nonapplicable actions in response to 12 emergency scenarios resulting from a severe seismic event at a pressurized water reactor. Specific personality characteristics of the operators suggested by the literature to be related to performance under stress were assessed and correlated to decisionmaking under stress.

NUREG/CR-4041: SYSTEM ANALYSIS HANDBOOK. LARSON,J.R. EG&G, Inc. January 1985. 75pp. 8502210404. EGG-2354. 29057:349.

This handbook provides simple procedures for calculating the behavior of light water reactors during a variety of incidents. It provides an additional tool for assessment of ongoing and postincident behavior. The handbook consists of a main body describing generic procedures, an appendix providing specific design data for a limited number of plants for application with the procedures, and an appendix listing existing and planned BWR and PWR plants by containment types and thermal-hydraulic parameters. The procedures are currently limited to break flow rate, decay heat power and integrated power, steam generation from decay heat, mass balance, shutdown margin, natural circulation, noncondensable gas generation, dose estimates, and DNB evaluation void formation in the upper head, and torus heatup.

NUREG/CR-4041 R01: SYSTEM ANALYSIS HANDBOOK. LARSON,J.R. EG&G, Inc. November 1985. 97pp. 8512050433. EGG-2354. 33776:154.

See NUREG/CR-4041 abstract.

NUREG/CR-4042: A 3-DIMENSIONAL COMPUTER MODEL TO SIMULATE FLUID FLOW AND CONTAINMENT TRANSPORT THROUGH A ROCK FRACTURE SYSTEM. HUANG,C.; EVANS,D.D. Arizona, Univ. of, Tucson, AZ. January 1985. 116pp. 8502210181. 29051:171.

A 3-dimensional fracture generating scheme is presented which can be used to simulate water flow and containment (solute) transport through fracture system of a rock. It is presently limited to water saturated conditions, zero permeability for the rock matrix, and steady state water flow, but allows for transient solute transport. The scheme creates finite planar plates of uniform thickness which represent fractures in 3-dimensional space. A given fracture (plate) has the following descriptors: center location, orientation, shape, areal extent and aperture. Each parameter can be described by an appropriate probability distribution. Individual fractures are generated to form an assemblage of a certain fracture density. All fracture intersections and boundary/fracture intersections are determined and deadend fractures are eliminated. Flow through the fracture assemblage is considered laminar and described by Poiseuille's law. The principle of mass conservation at each intersection is used to develop the global matrix equation, which is solved subject to specified boundary conditions to yield the head and flow distribution at each intersection. Solute transport is considered to be advective between intersections with complete mixing at each intersection. Solutes added to the flow system can be explicitly followed and concentration vs. time relationships can be determined anywhere in the system. Some examples are included.

NUREG/CR-4043: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-6. OSBORNE,M.F.; COLLINS,J.L.; LORENZ,R.A.; et al. Oak Ridge National Laboratory. October 1985. 65pp. 8511220329. ORNL/TM-9443. 33602:318.

The sixth in a series of high-temperature fission product release tests was conducted for 1 minute at 1950 degrees centigrade in a steam-helium atmosphere. The 15.2-cm-long test specimen was a section of fuel rod which was irradiated to 40.3 MWd/kg in the Monticello BWR. Posttest analyses showed total releases of 29.6% for (85)Kr (includes 2% that was released to the plenum during irradiation), 33.1% for (137)Cs, 24.7% for (129)I, 6.0% for (110m)Ag, and 0.06% for (125)Sb. A stainless steel thermal gradient tube was used to examine the retention of fission product cesium by stainless steel. Gamma scans showed that a significant fraction of the released cesium was released at temperatures >600 degrees centigrade where the surface had been oxidized. Cesium that was released as CsI appeared unaffected by its contact with the stainless steel. A comparison was made of Cs, I, and Kr release rate coefficients obtained in the HI and HT test series with NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," values.

NUREG/CR-4044: TRAC-PF1 LOCA CALCULATIONS USING FINE-NODE AND COARSE-NODE INPUT MODELS. DOBRANICH,D.; BUXTON,L.D.; WONG,C.N. Sandia National Laboratories. May 1985. 86pp. 8506190042. SAND84-2305. 31015:116.

TRAC-PF1 calculations of a 200% cold-leg break LOCA have been completed for a UHI plant using both fine-node (with 776 mesh cells) and coarse-node (with 320 mesh cells) input models. This study was performed to determine the effect of noding on predicted results and on computer running time. It was found that the overall sequence of events and the important trends of the transient were predicted to be nearly the same with both the fine-node and coarse-node models. There were differences in the time-dependent behavior of the cold-leg accumulator injection, and the predicted PCT for the coarse-node calculation was about 75 K less than that of the fine-node calculation. The higher PCT of the fine-node calculation is attributed primarily to three-dimensional flow effects in the core. The complete (steady state plus transient) coarse-node calculation required 13.5 hours of CYBER 76 computer time compared to 68.3 hours for the fine node calculation, yielding an overall factor of five decrease in running time. Thus, we conclude that for any large break LOCA analyses in which only the overall trends are of concern, the loss of accuracy resulting from use of such a coarse-node model will normally be inconsequential compared to the savings in resources that are realized. However, if the objective of the analyses is the investigation of the effects of multi-dimensional flows on clad temperatures, then a detailed model is required.

NUREG/CR-4045: LITERATURE REVIEW ON AEROSOL-SAMPLING DEVICES FOR RESPIRATORY FIELD STUDIES. SUTCLIFFE,C.R. Los Alamos Scientific Laboratory. February 1985. 58pp. 8502250844. LA-9977-MS. 29095:216.

As part of the first phase of a Respirator Field Performance project for the Occupational Safety and Health Administration/Nuclear Regulatory Commission, a critical review of the literature available on respirator protection studies was completed. Little information was available on experimental conditions, and when the information was available, each study was different in how the aerosol measurements were made and in which parameters were controlled. Under these conditions, it is difficult to compare results obtained from different investigators. The literature was also surveyed for characteristics desirable in an aerosol-sampling inlet in order to representatively sample respirable particles. Available ambient aerosol samplers were critically reviewed for their performance characteristics. Recommendations are made to avoid the pitfalls present in many respirator field studies and to help standardize these studies.

NUREG/CR-4046: DETERMINING CRITICAL FLOW VALVE CHARACTERISTICS USING EXTRAPOLATION TECHNIQUES. JARRELL,D.B. EG&G, Inc. March 1985. 28pp. 8504030456. EGG-2357. 29605:220.

This report presents the methodology and documentation of the calibration of the Loss-of-Fluid Test (LOFT) power-operated relief and safety relief valve (PORV + SRV) for the L9-3 anticipated transient without scram (ATWS) experiment. A multiposition globe valve was calibrated to produce scaled high-pressure flow rates using a low-pressure calibration facility and a simple RELAP5 critical flow model to extrapolate the calibration data to expected operating pressures. It was demonstrated that an accurate high pressure, multiphase flow calibration can be performed without the necessity of actual high-pressure testing. This technique, when applied to large pressurized water reactor (LPWR) safety and relief valves, represents a potentially large savings in the capacity qualification procedure of full-scale pressure reduction valves.

NUREG/CR-4050: A REVIEW OF THE SHOREHAM NUCLEAR POWER STATION PROBABILISTIC RISK ASSESSMENT.Internal Events And Core Damage Frequency. ILBERG,D.; SHIU,K.; HANAN,N.; et al. Brookhaven National Laboratory. November 1985. 330pp. 8512190024. BNL/NUREG-51836. 33966:065.

A review of the Probabilistic Risk Assessment of the Shoreham Nuclear Power Station was conducted with the broad objective of evaluating its risks in relation to those identified in the Reactor Safety Study (WASH-1400). The scope of the review was limited to the "front end" part, i.e., to the evaluation of the frequencies of states in which core damage may occur. Futhermore, the review considered only internally generated accidents, consistent with the scope of the PRA. The review included an assessment of the assumptions and methods used in the Shoreham study. It also encompassed a re-evaluation of the main results within the scope and general methodological framework of the Shoreham PRA, including both qualitative and quantitative analyses of accident initiators, data bases, and accident sequences which result in initiation of core damage. Specific comparisons are given between the Shoreham study, the results of the present review, and the WASH-1400 BWR, for the core damage frequency. The effect of modeling uncertainties was considered by a limited senstivity study so as to show how the results would change if other assumptions were made. This review provides an independently assessed point value estimate of core damage frequency and describes the major contributors, by frontline systems and by accident sequences.

NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS. SAARI,L.M.; MELBER,B.D.; WHITE,A.S.; et al. Battelle Human Affairs Research Centers. April 1985. 77pp. 8505010277. PNL-5303. 30114:352.

This report identifies job-related educational qualifications for the nuclear power plant licensed operator positions of reactor operator (RO), senior reactor operator (SRO), and shift supervisor (SS). The extent to which college engineering curriculum covers job-related academic knowledge was assessed. The approach used involved systematically comparing college engineering programs to knowledge needed on the job by having subject matter experts in the filed of general and nuclear engineering curriculum: (1) assess the coverage of specific academic knowledge identified by a job analysis as necessary for licensed operators in existing college engineering degree programs, and (2) make judgments concerning levels of formal engineering education necessary for application of knowledge on the job, based on job samples from a job analysis of activities under selected normal and emergency operating sequences. The major conclusions of the report are: a substantial amount (approximately 2/3) of job-related academic knowledge is covered in college engineering curriculum; college engineering curriculum provides considerable material beyond that identified as necessary for licensed operators; higher level operator positions (SS relative to SRO, SRO relative to RO) were judged as needing higher levels of education to perform the job.

NUREG/CR-4055: THE D10 EXPERIMENT:COOLABILITY OF UO2 DEBRIS IN SODIUM WITH DOWNWARD HEAT REMOVAL. MITCHELL,G.W.; OTTINGER,C.A.; MEISTER,H. Sandia National Laboratories. January 1985. 84pp. 8503050012. SAND84-1144. 29246:001.

The LMFBR Debris Coolability Program at Sandia National Laboratories investigates the coolability of particle beds which may form following a severe accident involving core disassembly in a nuclear reactor. The D series experiments utilize fission heating of fully enriched UO(2) particles submerged in sodium to realistically simulate decay heating. The D10 experiment is the first in the series to study the effects of bottom cooling of the debris which could be provided in an actual accident condition by structural materials onto which the debris might settle. Additionally, the D10 experiment was designed to achieve maximum temperatures in the debris approaching the melting point of UO(2). The experiment was successfully operated for over 50 hours and investigated downward heat removal in a packed bed at specific powers of 0.16 to 0.58 W/g. Dryout in the debris was achieved at powers from 0.42 to 0.58 W/g. Channels were induced in the bed and channeled bed dryout was achieved at powers of 1.06 to 1.77 W/g. Maximum temperatures in excess of 2500 degrees centigrade were attained.

NUREG/CR-4056: PARTICULATE AND GAS RELEASE FROM LIGHT-WATER-REACTOR (LWR) FUEL RODS STORED IN INERT AND DRY AIR ATMOSPHERES. OLSEN,C.S. EG&G, Inc. January 1985. 24pp. 8501210003. EGG-2359. 28497:013.

A testing program using eight commercial pressurized water reactor (PWR) and boiling water reactor (BWR) spent fuel rods was conducted to investigate their long-term stability under a variety of possible dry storage conditions. The objective of this project is to provide the Nuclear Regulatory Commission (NRC) with information to confirm or establish licensing positions for dry spent fuel rod storage with regard to long-term, low-temperature (<250 degrees centigrade), spent fuel rod behavior during dry storage and radioactive contamination arising from spallation of cladding crud. The results of the analyses of the crud, fuel particulate, and gas release from these eight fuel rods is presented, which includes weight change measurements, delayed neutron measurements, and isotopic analysis of smears used to assess the particulate release. Gas analyses of the fuel rod capsule environments were made to determine the fission gas release, and flow tests were performed to determine the extent of filter blockage from particle entrapment.

NUREG/CR-4057: RADIOLOGICAL ASSESSMENT OF THE TOWN OF EDGEMONT. JACKSON,P.O.; THOMAS,V.W.; YOUNG,J.A. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1985. 183pp. 8502040625. PNL-5320. 28727:151.

This document is the final report for radiological surveys conducted in the community of Edgemont, South Dakota for the purpose of locating residual radioactive materials from the uranium processing industry. It contains a discussion of the historical justification for the surveys, and a summary of activities during the survey, from September 1980 through November 1984. The survey protocols are presented and discussed. The results of several studies of relevance to the surveys are also included. The results of the survey are presented in tabular form.

NUREG/CR-4060: THE DC-1 AND DC-2 DEBRIS COOLABILITY AND MELT DYNAMICS EXPERIMENTS. HITCHCOCK,J.T.; KELLY,J.E. Sandia National Laboratories. July 1985. 163pp. 8508090648. SAND84-1367. 32101:142.

The DC experiment series investigates the heatup and melt of dry reactor core debris through nuclear heating of actual reactor materials in order to obtain the thermal properties of dry debris, the nature of the transition from a debris bed to a molten pool, and the thermal and kinetic behavior of molten pools. The purpose is to develop a data base in support of model development. The work is jointly sponsored by the USNRC, the PNC (Japan), and EURATOM. This report provides a description of the two experiments in the DC series and documents the configuration and the data. These tests investigated dry debris beds (2 kg) composed of pure UO(2) and mixed UO(2) and stainless steel. Heat transfer characteristics were studied at several steady-state conditions below melt. The beds were then taken into melt to observe the growth of a molten pool in the UO(2) bed and the agglomeration and migration of steel in a composite bed. The peak measured temperature in the UO(2) bed was above 3000 degrees centigrade. Approximately 50% of the urania formed a molten pool. In the mixed UO(2) and steel bed, the peak measured temperature was 2600 degrees centigrade. With about 90% of the steel molten, material migration occurred resulting in a significant increase in the gross bed thermal conductivity.

NUREG/CR-4061: LEACHATE PLUME MIGRATION DOWNGRADIENT FROM URANIUM TAILINGS DISPOSAL IN MINE STOPES. NELSON,R.W.; MCKEON,T.J.; CONBERE,W. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1985. 82pp. 8504050284. PNL-5318. 29673:228.

A method previously developed at Pacific Northwest Laboratory has been simplified and extended to better evaluate the environmental consequences of below-water-table disposal of uranium mill tailings in mine stopes. The method described uses analytical expressions for the velocity potential and examines numerically the convective transport of tailings liquor and leachate through the aquifer and into a water supply well located downgradient from the mine stope. The overall dependence of the leachate plume size and shape on the hydrologic parameters and the tailings disposal geometry are presented in graphical form for use in preliminary assessments. The graphical results are also used to set up worst-case scenarios for return of the leachate constituents to the biosphere via the pumped water supply well. The interactive computer models developed to evaluate such worst-case conditions are presented, discussed, and used to evaluate four typical situations.

NUREG/CR-4062: EXTENDED STORAGE OF LOW-LEVEL RADIOACTIVE WASTES:Potential Problem Areas. SISKIND,B.; DOUGHERTY,D.R.; MACKENZIE,D.R. Brookhaven National Laboratory. December 1985. 149pp. 8601070490. BNL-NUREG-51841. 34189:141.

If a state or state compact does not have adequate disposal capacity for low-level radioactive waste (LLRW), then extended storage of certain LLRW may be necessary. Extended storage of LLRW is considered in order to determine for the Nuclear Regulatory Commission areas of concern and actions recommended to resolve these concerns. The focus is on the properties and performance of the waste form and waste container. Storage alternatives are considered in order to characterize the likely storage environments for these wastes. The areas of concern are grouped into two categories: 1. Performance of the waste form and/or container during storage, e.g., radiolytic gas generation, radiation-enhanced degradation of polymeric materials, and corrosion. 2. Effects of extended storage on the properties of the waste form and/or container that are important after storage (e.g., radiation-induced embrittlement of high-density polyethylene and the weakening of steel containers resulting from corrosion. A discussion is given of additional information and actions required to address these concerns.

NUREG/CR-4064: STRUCTURAL RESPONSE OF LARGE PENETRATIONS AND CLOSURES FOR CONTAINMENT VESSELS SUBJECTED TO LOADINGS BEYOND DESIGN BASIS. KULAK,R.F. Argonne National Laboratory. * Sandia National Laboratories. April 1985. 109pp. 8505060514. ANL-84-41. 30193:019.

This report summarizes the analyses work performed by Argonne National Laboratory on three representative nuclear power plant penetrations for severe accident loads beyond the design basis conditions. These include analyses of an equipment hatch for a steel containment, a BWR-Mark II drywell head and a bellows connection. The objectives of the analyses were to identify the methodology required to simulate the response of the penetrations and determine their leakage potential under severe accident loads. This report provides the details of the analytical methodology used and the results obtained from the analyses.

NUREG/CR-4067: SUMMARY OF BARRIER DEGRADATION EVENTS AND SMALL ACCIDENTS IN U.S. COMMERCIAL NUCLEAR POWER PLANTS. SAILOR,V.L.; COLBERT,J.J. Brookhaven National Laboratory. March 1985. 66pp. 8504020086. BNL-NUREG-51842. 29585:135.

The experience of U.S. commercial nuclear power plants with respect to small accidents and events involving the breach of any of the various barriers to radioactive material release is reviewed and brief summaries are given of selected events. This report is intended to provide background information for the NRC staff evaluation of the proposed NRC safety goals. Included are events that resulted in the breach of one or more barriers (fuel cladding failures, primary coolant leakage, compromise of containment integrity), or in unintentional release of radioactive materials. Also included are miscellaneous small accidents or failures not resulting in radioactive releases, but which had special safety implications. The 1979 TMI-2 accident is not included. The report does not attempt to evaluate the significance of the events as potential precursors of more severe accidents (such evaluations are the subject of other studies). Rough statistics are presented on the frequency of events defined above for the period, 1974-1982. It is noted that none of the events resulted in fatalities or injuries attributable to radiological causes.

NUREG/CR-4068: SUMMARY OF HISTORICAL EXPERIENCE WITH RELEASES OF RADIOACTIVE MATERIALS FROM COMMERCIAL NUCLEAR POWER PLANTS IN THE UNITED STATES. SAILOR,V.L.; COLBERT,J.J. Brookhaven National Laboratory. March 1985. 73pp. 8503280025. BNL-NUREG-51843. 29548:007.

This report presents a summary of the historical experience concerning releases of radioactive materials from U.S. commercial nuclear power plants. The material was compiled specifically to provide background information for the Nuclear Regulatory Commission (NRC) Staff Evaluation of the proposed NRC Safety Goals. The types of available data on radioactive emissions are identified, reviewed and summarized. The annual 50-year population radiation dose commitments for the annular regions between 2 and 80 km surrounding each plant resulting from the radioactive emissions are summarized for the period, 1975-1981. These doses are compared with the annual population dose commitments from natural background radiation for the same areas, and with the proposed NRC Societal Safety Goal. The question of independent verification of licensee data on emissions is examined.

NUREG/CR-4069: ANALYSES OF SOILS FROM AN AREA ADJACENT TO THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE AT SHEFFIELD,ILLINOIS. PICIULO,P.L.; SHEA,C.E.; BARLETTA,R.E. Brookhaven National Laboratory. March 1985. 54pp. 8503200236. BNL-NUREG-51844. 29470:255.

Soil samples and field resistivity data were collected from an area adjacent to the Sheffield site. Specimens of Peoria Loess, Roxana Silt, Radnor Till, sand from the Toulon member, Hulick Till, and shale from the Pennsylvania system were collected and analyzed. Resisitivities of the soils are all greater than 2500 ohm-cm, indicating an environment which can be moderately corrosive to steel. Measurements of soil pH range from 6.2 to 8.6. Determination of the total acidity of the soils indicates an alkaline environment. The moisture content of the soils are representative of a wet site. The ion content of the soils show high levels of calcium consistent with the calcareous nature of the soils. Both the extractable and exchangeable concentrations of calcium, magnesium, potassium, and sodium in the soils are reported. The content of the following soluble anions is also given: carbonate, bicarbonate, sulfate, sulfide, and chloride.

NUREG/CR-4070 V02: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Volume 2:Current Status Of Biofouling Surveillance And Control Techniques. DALING,P.M.; JOHNSON,K.I. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1985. 68pp. 8503270015. PNL-5300. 29541:211.

This report describes the current status of techniques for detection and control of cooling-water system fouling by bivalve mollusks at nuclear power plants. The effectiveness of these techniques is evaluated on the basis of information gathered from a literature review and in interviews with nuclear power plant personnel. Biofouling detection techniques examined in this report include regular maintenance, in-service inspection, and testing. Generally, these methods have been inadequate for detecting biofouling. Recommendations for improving biofouling detection capabilities are presented. Biofouling prevention (or control) methods that are examined in this report include intake screen systems, thermal treatment, preventive maintenance, chemical treatment alternatives, and antifoulant coatings. Recommendations for improving biofouling control methods at operating nuclear power plants are presented. Additional techniques that could be implemented at future power plants or that require further research are also described.

NUREG/CR-4070 V03: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Factors That May Intensify The Safety Consequences Of Biofouling. HENAGER,C.H.; DALING,P.M.; JOHNSON,K.I. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1985. 61pp. 8504220375. PNL-5300. 29946:202.

This report describes the safety and economic consequences of bivalve fouling in raw-water systems at nuclear power plants. The report lists events that could cause a normal fouling situation to become more critical and describes scenarios in which bivalve fouling could cause unsafe or unwanted conditions such as transients and shutdowns. Several fouling events that have occurred at various nuclear plants are briefly reviewed, and recommendations are made to aid in the detection and control of bivalve fouling.

NUREG/CR-4071: EXPLORATORY TREND AND PATTERN ANALYSIS FOR 1981 LICENSEE EVENT REPORT DATA. HESTER,O.V.; GENTILLON,C.D. EG&G, Inc. April 1985. 215pp. 8505280415. EGG-2362. 30601:072.

This report presents an overview of the 1981 Sequence Coding and Search System (SCSS) data base that contains nuclear power plant operational data derived from Licensee Event Reports (LERs) submitted to the United States Nuclear Regulatory Commission. Both overall event reporting and events related to specific components, subsystems, systems, and personnel are discussed. At all of these levels of information, software is used to generate count data for contingency tables. Contingency table analysis is the main tool for the trend and pattern analysis. The tables primarily focus on faults associated with various components and other items of interest across different plants. The abstracts and other SCSS information on the LERs accounting for unusual counts in the tables were examined to gain insights from the events.

NUREG/CR-4072: THE ESTIMATION OF ATMOSPHERIC DISPERSION AT NUCLEAR POWER PLANTS UTILIZING REAL TIME ANEMOMETER STATISTICS. LI,W.W.; MERONEY,R.N. Colorado State Univ., Ft. Collins, CO. January 1985. 236pp. 8502010665. 28703:001.

Dispersion and turbulence measurements were conducted in a simulated atmospheric boundary layer. Field experiments and wind tunnel results for the behavior of lateral plume dispersion are compared to three semi-emperical expressions based on Taylor's diffusion theory. Agreement between the field data and laboratory measurements supports using wind tunnel results to simulate atmospheric transport phenomena. Eulerian space-time correlations with streamwise separations were measured for all three velocity components in the simulated boundary layer. Results were compared to previous measurements which were performed under different flow configurations. A universal shape of the Eulerian space-time correlation seems to exist when presented in a normalized time coordinate. Turbulence measurements of fixed-point Eulerian velocity statistics were employed to estimate the Lagrangian velocity statistics through the Baldwin and Johnson approach. The approach was modified to account for the uniform shear stress effect in a homogenous turbulent flow field. The estimated Lagrangian integral time scale agrees with estimates inferred from dispersion measurements within only a 20% error. Such agreement supports the methodology of using real time anemometer statistics to predict the atmospheric turbulent dispersion near a nuclear reactor site.

NUREG/CR-4073: RESULTS OF THE SEMISCALE MOD-2B STEAM GENERATOR TUBE RUPTURE TEST SERIES. LOOMIS,G.G. EG&G, Inc. January 1985. 75pp. 8502060492. EGG-2363. 28745:278.

A series of experiments was conducted in a scaled model of a pressurized water reactor (Semiscale Mod-2B) to investigate steam generator tube rupture system signature response and recovery techniques. The tube rupture was assumed to occur during normal full power operation [15.6 MPa (2262 psia) system pressure; 37 K (67 degrees fahrenheit) core differential temperature]. From the experimental results, the characteristic system signature responses for a wide range of number of tubes ruptured and rupture locations have been examined. In addition, recovery techniques requiring operator actions were examined. These recovery techniques included the use of pressurizer auxiliary spray and internal heaters, steam generator feed and steam, primary feed and bleed, and safety injection. The effectiveness of using these techniques for primary system pressure and subcooling control is discussed.

NUREG/CR-4074: THE PERFORMANCE OF DEFECTED SPENT LWR FUEL RODS IN INERT GAS AND DRY AIR STORAGE ATMOSPHERES. OLSEN,C.S. EG&G, Inc. January 1985. 35pp. 8502220293. EGG-2364. 29073:105.

A testing program using eight commercial pressurized water reactor and boiling water reactor spent fuel rods was conducted to investigate their long-term stability under a variety of possible dry storage conditions. The objective of this project was to provide the Nuclear Regulatory Commission with information to confirm or establish dry spent fuel storage licensing positions for long-term, low-temperature (<250 degrees centigrade) spent fuel rod behavior during dry storage and radioactive contamination arising from spallation of cladding crud. The results of a nondestructive examination of eight fuel rods, which included color closed-circuit television visual examinations, color photography, dimensional measurements, and neutron radiography, are presented.

NUREG/CR-4075: DESIGNING PROTECTIVE COVERS FOR URANIUM MILL TAILINGS PILES. A Review. BEEDLOW,P.A.; PARKER,G.B. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 29pp. 8505280093. PNL-5323. 30604:247.

This report reviews design considerations for protective covers for uranium mill tailings impoundments. The role of protective covers in tailings containment systems is discussed. Factors affecting the long-term stabilization of tailings (erosion, biotic intrusion, and soil moisture) are summarized. Basic elements to be considered in design of all uranium tailings covers are presented, and then quantitative techniques for designing site-specific covers are reviewed.

NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION. BEEDLOW,P.A.; CLINE,J.F.; FREEMAN,H.D.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 48pp. 8507030712. PNL-5324. 31318:250.

This report provides methods and procedures that can be used to verify compliance with Environmental Protection Agency (EPA) engineering standards for uranium mill tailings disposal sites. EPA standards for radon emissions, long-term isolation, and protection of water quality are discussed. Tailings isolation technologies are reviewed. Information the licensee needs to provide for the regulating agency to determine compliance is presented, as is the actual compliance criteria.

NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT. KITTMER,C.A.; WENSEL,R.G.; RHODES,D.B.; et al. Atomic Energy of Canada, Ltd. April 1985. 93pp. 8506170667. EGG-2365. 30979:290.

A testing program designed to provide fundamental information pertaining to the behavior of reactor coolant pump (RCP) shaft seals during a postulated nuclear power plant station blackout has been completed. The test plan was developed by EG&G Idaho personnel at the Idaho National Engineering Laboratory (INEL) and performed at the Chalk River Nuclear Laboratory, Ontario, Canada, under auspices of the U.S. Nuclear Regulatory Commission (NRC). One seal assembly, utilizing both hydrodynamic and hydrostatic types of seals, was modeled and tested. Extrusion tests were conducted to determine if seal materials could withstand predicted temperatures and pressures. A taper-face seal model was tested for seal stability under conditions when leaking water flashes to steam across the seal face. Test information was then used as the basis for a station blackout analysis. Test results indicate a potential problem with an elastomer material used for O-rings by a pump vendor; that vendor is considering a change in material specification. Test results also indicate a need for further research on the generic issue of RCP seal integrity and its possible consideration for designation as an unresolved safety issue.

NUREG/CR-4079: ANALYTIC STUDIES PERTAINING TO STEAM GENERATOR TUBE RUPTURE ACCIDENTS. KASHIWA,B.A.; MJOLSNESS,R.C. Los Alamos Scientific Laboratory. April 1985. 88pp. 8506060372. LA-10307-MS. 30773:271.

A study of thermal-hydraulic phenomena of possible steam generator tube rupture (SGTR) accidents leads to the conclusions that (1) flashing will not occur upstream of the tube rupture, so that the flow will be resistance limited rather than choked, (2) there is considerable potential for discharging the primary fluid in the form of micron-sized droplets, particularly when the fluid discharges into a vapor cavity surrounding the tube rupture, and (3) that the surrounding of the rupture site by water rather than vapor may be a means for preventing the formation of micron-sized droplets. The presence or absence of micron-sized droplets is considered to be a key issue for the damage assessment of SGTR accidents because they are currently thought to be the most likely route for radioactive iodine to be released to the atmosphere.

NUREG/CR-4080: DETERMINATION OF THE AVAILABILITY OF CORE EXIT THERMOCOUPLES DURING SEVERE ACCIDENT SITUATIONS. EDSON,J.L. EG&G Idaho, Inc. (subs. of EG&G, Inc.). September 1985. 47pp. 8510030430. EGG-2366. 32848:082.

This report presents the findings and recommendations of the Nuclear Power Plant Instrumentation Evaluation (NPPIE) program concerning signal validation methods to determine the on-line availability of core exit thermocouples during accident situations. Methods of selecting appropriate signal validation techniques are discussed and sources of error identified. This report shows that through the use of these techniques the existence of high-temperature-caused errors may be detected as they occur. Specific recommendations for application of selected signal validation techniques to core exit thermocouples and other measurement systems are made.

NUREG/CR-4081: ABSORPTION OF GASEOUS IODINE BY WATER DROPLETS. ALBERT,M.F. Oak Ridge National Laboratory. August 1985. 212pp. 8509110032. ORNL/TM-9488. 32558:154.

A new model has been developed for predicting the rate at which gaseous molecular iodine is absorbed by water sprays. The model is a quasi-steady state mass transfer model that includes the iodine hydrolysis reactions. The parameters of the model are spray drop size, initial concentration of the gas and liquid phases, temperature, pressure, buffered or unbuffered spray solution, spray flow rate, containment diameter and drop fall height. The results of the model were studied under many values of these parameters. Plots of concentration of iodine species in the drop versus time have been produced by varying the initial gas phase concentration of molecular iodine over the range of 1 x 10 (-5) moles/liter to 1 x 10 (-10) moles/liter, a buffered pH of 7 or 9, and a drop size of 1000 microns. Results from the model are compared to results available from the Containment Systems Experiments at Pacific Northwest Laboratory. The difference between the model predictions and the experimental data ranges from -120% to 68% with the closest agreement 7.7%. The new spray model is also compared to previously existing spray models. At high concentrations of gaseous molecular iodine, the new spray model is considered to be less accurate than the previous models. At low concentrations, the new model predicts results that are closer to the experimental data. Inclusion of the iodine hydrolysis reactions is shown to be important for determining the removal of molecular iodine from gas phase by water sprays for most conditions.

NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984. WILKOWSKI,G.M.; AHMAD,J.; BARNES,C.R.; et al. Battelle Memorial Institute, Columbus Laboratories. January 1985. 118pp. 8501280617. BMI-2120. 28574:202.

The objective of the Degraded Piping Program - Phase II is to develop simple engineering analyses to assess the fracture behavior of nuclear piping. Such analyses must give realistic estimates of actual fracture events. Hence this is an intensely integrated program involving laboratory material property evaluation, analytical developments, and full-scale pipe fracture experiments to verify the simple engineering analyses. Both advance fracture mechanics analyses (i.e., J/T), and limit-load analyses will be assessed. This is a 3-year program which began in March, 1984. Consequently, this first semiannual report describes work in progress rather than completed efforts.

NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985. WILKOWSKI,G.M.; AHMAD,J.; BARNES,C.R.; et al. Battelle Memorial Institute, Columbus Laboratories. July 1985. 367pp. 8508090632. BMI-2120. 32100:136.

The efforts in this report are broken into six work packages related to pipe-fracture research efforts and six work packages that are supporting research efforts. The pipe-fracture efforts involve only circumferential crack orientations. Twenty-six pipe experiments have been conducted to date, with all but two at 500F (288C). Approximately 35 additional pipe experiments from past programs were also analyzed. Analysis efforts include limit load and elastic-plastic fracture mechanics analysis. Elastic-plastic fracture-mechanics analytical efforts concentrate on J-integral estimation schemes that can predict loads and displacements (predictive J-estimation schemes), rather than those that can only be used to calculate the toughness (n-factor analysis). Finite-element analyses are conducted in selected cases. Supporting research efforts involve geometry effects on J-R curves, notch acuity effects, predicting J-R curves with large amounts of crack growth from small specimens, development of a large compliant pipe test system, evaluation of cracks in welds, and procurement of cracked pipe removed from service.

NUREG/CR-4083: ANALYSES OF SOILS FROM THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITES AT BARNWELL,SC AND RICHLAND,WA. PICIULO,P.L.; SHEA,C.E.; BARLETTA,R.E. Brookhaven National Laboratory. March 1985. 62pp. 8503290285. BNL-NUREG-51846. 29564:029.

To evaluate the performance of a buried waste form or waste container, consideration must include the interaction of the package with the burial environment. This report presents the results of physical and chemical measurements of soils from two currently operating commercial radioactive waste disposal sites; one at Barnwell, SC, and the other near Richland, WA. Soil samples believed to be representative of the soil that will contact the buried waste forms were collected and analyzed. Resistivity data given for soils from both sites indicate mildly corrosive environments. The soil acidity measurements show the Barnwell site to have acidic soil, whereas, the Richland site has soils ranging from acidic to near neutral in pH. The moisture content and the ion content of the soils from each site are presented. The extractable ion content of the soils is given for the following ions: calcium, magnesium, potassium, sodium, carbonate, bicarbonate, sulfate, sulfide, and chloride. Additionally, the exchangeable cations were measured for the soils from the two sites.

NUREG/CR-4084: DRY SPENT FUEL STORAGE TEST PLAN FOR DESTRUCTIVE FUEL ROD EXAMINATIONS. OLSEN,C.S. EG&G, Inc. April 1985. 41pp. 8504220369. EGG-2367. 29946:164.

A testing program using eight commercial pressurized water reactor and boiling water reactor spent fuel rods was conducted to investigate their long-term stability under a variety of possible dry storage conditions. The objective of this report is to provide the Nuclear Regulatory Commission with information to confirm or establish dry spent fuel storage licensing positions for long-term, low-temperature (<250 degrees centigrade) spent fuel rod behavior during dry storage and for radioactive contamination that might occur with spallation of cladding crud. Six of the eight commercial fuel rods will be destructively examined. This report presents the test plan for the destructive examinations.

NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis. BERGERON,K.D.; CLAUSER,M.J.; HARRISON,B.D.; et al. Sandia National Laboratories. July 1985. 354pp. 8508090656. SAND84-1204. 32099:142.

The CONTAN 1.0 computer code is an integrated analysis tool for the physical, chemical, and radiological conditions inside a containment building following the release of radioactive material from the primary system in a severe reactor accident. It can also predict the source term to the environment. The purpose of this User's Manual is to provide a basic understanding of the features and models in CONTAIN 1.0 so that users can prepare reasonable input and understand the output and its significance for particular applications. Besides input instructions, the User's Manual also contains brief descriptions of the basic features of the models. Both light-water reactors and liquid-metal reactors can be modeled with CONTAIN 1.0. The code includes atmospheric models for steam/air thermodynamics, intercell flows, condensation/evaporation on sturctures and aerosols, aerosol behavior, hydrogen burning, sodium/atmosphere chemistry, sodium-spray fires, and sodium-pool fires. It also includes models for reactor cavity phenomena such as core/concrete interactions, coolant-pool boiling, and sodium/concrete interactions. Heat conduction in structures, fission-product decay and transport, radioactive heating, and the thermal-hydraulic and fission-product decontamination aspects of engineered safety features are also modeled.

NUREG/CR-4086: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL WELDS FOR THE THIRD HSST IRRADIATION SERIES. MCGOWAN,J.J. Oak Ridge National Laboratory. May 1985. 23pp. 8505160629. ORNL/TM-9477. 30439:288.

The Heavy Section Steel Technology (HSST) Program conducted a series of experiments to investigate the effect of neutron irradiation on the fracture toughness of nuclear pressure vessel materials. Four welds of A 508 class 2 steel were examined in this Third HSST Irradiation Series. The welds were fabricated according to "early" (pre-1972) lightwater reactor weld practice (i.e., copper-coated electrodes). As part of this study, tensile properties were measured after irradiation to 2 to 10 x 10(22) neutrons/m(2) (E > 1 MeV) at temperatures between 250 and 290 degrees centigrade. Strength properties of all four welds increased with exposure to irradiation. Yield strength was more sensitive to irradiation than was ultimate strength. Tensile ductility was not affected significantly by exposure to irradiation.

NUREG/CR-4087: MEASUREMENTS OF URANIUM MILL TAILINGS CONSOLIDATION CHARACTERISTICS. FAYER,M.J. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1985. 44pp. 8503010322. PNL-5339. 29186:059.

Experiments were conducted on uranium mill tailings from the tailings pile in Grand Junction, Colorado, to determine their consolidation characteristics. Three materials (sand, sand/slimes mix, slimes) were loaded under saturated conditions to determine their saturated consolidation behavior. During a separate experiment, samples of the slimes material were kept under a constant load while the pore pressure was increased to determine the partially saturated consolidation behavior. Results of the saturated tests compared well with published data. Sand consolidated the least, while slimes consolidated the most. As each material consolidated, the measured hydraulic conductivity decreased in a linear fashion with respect to the void ratio. Partially saturated experiments with the slimes indicated that there was little consolidation as the pore pressure was increased progressively above 7 kPa. The small amount of consolidation that did occur was only a fraction of the amount of saturated consolidation. Preliminary measurements between pore pressures of 0 and 7 kPa indicated that measurable consolidation could occur in this range of pore pressure, but only if there was no load.

NUREG/CR-4088: METHODS FOR ESTIMATING RADIOACTIVE AND TOXIC AIRBORNE SOURCE TERMS FOR URANIUM MILLING OPERATIONS. HARTLEY,J.N.; GLISSMEYER,J.A.; HILL,O.F. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 69pp. 8507080192. PNL-5338. 31393:099.

Pacific Northwest Laboratory, under contract to the U.S. Nuclear Regulatory Commission (NRC), identified and evaluated methods for estimating radioactive and toxic particulate and gaseous airborne releases from uranium milling operations. Such methods need to be standardized so that all uranium mills can provide adequate data for NRC evaluation of potential environmental impacts and of compliance with 10 CFR 20, 40 CFR 190, and the National Environmental Policy Act. The general method for calculating source terms is to multiply together a normalized emission rate, contaminant content, emission control factor, and processing rate for each process being evaluated. This report describes the sources of airborne releases (ore storage area, ore crushing and grinding, ore processing, yellowcake production, and tailings impoundment) and the calculational procedures for estimating radioactive and toxic source terms. Example calculations are provided.

NUREG/CR-4089: EVALUATION OF FIELD-TESTED FUGITIVE DUST CONTROL TECHNIQUES FOR URANIUM MILL TAILINGS PILES. ELMORE,M.R.; HARTLEY,J.N. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1985. 68pp. 8502150694. PNL-5340. 28960:305.

Seventeen chemical stabilizers, rated as the most promising of those tested in earlier laboratory studies, were applied to test plots on a uranium mill tailings pile at the American Nuclear Corporation-Gas Hills Project mill site in central Wyoming. The durability of these materials when exposed to actual site conditions was evaluated over time. In addition, eight commercially available windscreens were field tested. Test panels of the eight materials were constructed at the Wyoming site to compare their relative resistance to weathering. A second test was conducted near Pacific Northwest Laboratory to evaluate the effectiveness of the windscreens at reducing wind velocity. Results of the field tests on the chemical stabilizers and windscreens are presented in this report, along with effectiveness-versus-cost information. Direct comparison of these two dust control methods is difficult due to the dependence of each on site-specific factors. However, simplified model case studies were developed to assess the cost of chemical stabilization versus windscreen systems for a hypothetical, inactive tailings pile.

NUREG/CR-4090: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Annual Summary Report - Fiscal Year 1984. MILLER,R.L.; BAUMANN,B.L.; DOERGE,D.H. United Nuclear Corp. (subs. of UNC Resources, Inc.). January 1985. 124pp. 8502010524. 28702:207.

This document summarizes work performed during the 1984 fiscal year for the Nuclear Regulatory Commission's Evaluation of Nuclear Facility Decommissioning Projects program. This report describes actual work performed during the reporting period and work planned for the future. Included as appendices to this report are drafts of the current data from the TMI-2 recovery efforts and Shippingport Atomic Power Station decommissioning.

NUREG/CR-4091: THE EFFECT OF ALTERNATIVE AGING AND ACCIDENT SIMULATIONS ON POLYMER PROPERTIES. BUSTARD,L.D.; CHENION,J.; CARLIN,F.; et al. Sandia National Laboratories. June 1985. 177pp. 8507050391. SAND84-2291. 31373:139.

The influence of accident irradiation, steam, and chemical spray exposures on the behavior of twenty-three age-preconditioned polymer sample sets (twenty-one different materials) has been investigated. The test program varied the following conditions: 1. Accident simulations of irradiation and thermodynamic (steam and chemical spray) conditions were performed both sequentially and simultaneously; 2. Accident thermodynamic (steam and chemical spray) exposures were performed both with and without air present during the exposures; 3. Sequential accident irradiations were performed both at 28 degrees centigrade and 70 degrees centigrade; 4. Age preconditioning was performed both sequentially and simultaneously; 5. Sequential aging irradiations were performed both at 27 degrees centigrade and 70 degrees centigrade; and 6. Sequential aging exposures were performed using two sequences: (1) thermal followed by irradiation and (2) irradiation followed by thermal. This report presents both general trends applicable to a majority of the tested materials as well as specific results for each polymer. The data base consists of ultimate tensile properties at the completion of the accident exposure for three XLPO and XLPE, five EPR and EPDM, two CSPE (HYPALON), one CPE, one VAMAC, one polydiallylphtalate, and one PPS material. Report bend test results at completion of the accident exposures for two TEFZEL materials and permanent set after compression results for three EPR, one VAMAC, one BUNA N, one SILICONE, and one VITON material are also presented.

NUREG/CR-4092: ORNL CHARACTERIZATION OF HEAVY-SECTION STEEL TECHNOLOGY PROGRAM PLATES 01,02,AND 03. STELZMAN,W.J.; BERGGREN,R.G.; JONES,T.N. Oak Ridge National Laboratory. April 1985. 176pp. 8506060376. ORNL/TM-9491. 30773:101.

Charpy V-notch impact, tensile, and drop-weight data are presented for three 305-mm-thick (12-in.) A 533 grade B class 1 steel plates. The effects of specimen size and orientation were examined as well as the variation of properties between different plate locations and depths. Some observations based on data obtained from an instrumented Charpy testing machine are also presented.

NUREG/CR-4093: SAFETY/SAFEGUARDS INTERACTIONS DURING SAFETY-RELATED EMERGENCIES AT NUCLEAR POWER REACTOR FACILITIES. MOUL,D.A.; PILGRIM,M.K.; SCHWEIZER,R.L.; et al. Brookhaven National Laboratory. June 1985. 229pp. 8507020094. BNL-NUREG-51848. 31308:138.

This report contains an analysis of the safety/safeguards interactions that could occur during safety-related emergencies at licensed nuclear power reactors, and the extent to which these interactions are addressed in existing or proposed NRC guidance. The safety/safeguards interaction during a series of postulated emergencies was systematically examined to identify any potential performance deficiencies or conflicts between the Operations (safety) and Security (safeguards) organizations. This examination included the impacts of coordination with offsite emergency response personnel. Duties, responsibilities, optimal methods, and procedural actions inherent in these interactions were explored.

NUREG/CR-4094: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN SULFATE LAKE ENVIRONMENTS. SIMPSON,H.J.; TRIER,R.M.; HERCZEG,A.L.; et al. Columbia Univ., New York, NY. January 1985. 72pp. 8501280372. 28572:001.

The concentrations of a number of radioisotopes of some elements (Pu, U, Th, Pa, Ac, Ra, Bi, Po, Pb, Cs, Sr, and K) were measured in a group of lakes that are dominated by SO(4)(2-) ion in their anionic composition. Only Pu and the Th show possible enhancement of solubility at high sulfate concentrations in some Saskatchewan lakes, although never to the extent as observed for alkaline lakes. Surface waters of Green Lake (meromictic) which are at saturation with respect to calcite have approximately the same radionuclide content as seawater. The deepwaters (anoxic) show a marked increase in Th and Ra. This indicates that these elements may be coupled with the redox cycle of Fe and Mn which under oxygenated conditions effectively sequester Pu, Th, and Ra as Fe(-Mn) oxyhydroxides. Another possibility for the enhanced (226)Ra in the deep water is by coprecipitation with CaCO(3) in the surface water and subsequent dissolution in the deep water thereby releasing radium into the water.

NUREG/CR-4095: TEST SERIES 2:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E EXIDE FHC-19 BATTERY CELLS. BONZON,L.L.; HENTE,D.B. Sandia National Laboratories. April 1985. 166pp. 8507020389. 31307:332.

This report, the second in a test series of an extensive seismic research program, covers the testing of 10-year old lead-calcium Exide FHC-19 cells from the Calvert Cliffs Nuclear Power Station operated by the Baltimore Gas and Electric Company. The Exide cells were tested in two configurations using a triaxial shake table: single-cell tests, both rigidly and loosely mounted; and multicell (three-cell) tests, mounted in a typical battery rack. A total of six electrically active cells was used in the two different cell configurations. None of the six cells failed in the first stage tests during the actual seismic test up to the 1.5 g ZPAs imposed. Subsequent discharge capacity tests showed, however, that only two of the cells could deliver the accepted standard of 80% of their rated electrical capacity for 3 hours. When two of the same cells were exposed to the second stage, higher g-level tests, both cells again provided instantaneous uninterrupted power. Subsequent capacity tests showed both of these cells to have capacities well below the accepted standard of 80%. Four of the cells were disassembled for examination and metallurgical analyses. The examination showed the active material on the positive plates was hard and cracked and that the positive bus bar material was corroded and brittle.

NUREG/CR-4096: TEST SERIES 3:SEISMIC FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E C&D LCU-13 BATTERY CELLS. BONZON,L.L.; HENTE,D.B. Sandia National Laboratories. April 1985. 170pp. 8507020413. SAND84-2629. 31309:159.

This report, the third in a test series of an extensive seismic research program, covers the testing of 10-year old lead-calcium C&D LCU-13 cells from the North Anna Nuclear Power Station operated by the Virginia Electric and Power Company. The C&D cells were tested in two configurations using a triaxial shake table: single-cell tests, both rigidly and loosely mounted; and multicell (three-cell) tests, mounted in a typical battery rack. A total of seven electrically active cells was used in the two different cell configurations. None of the seven cells failed in the first stage tests during the actual seismic test up to the 1.5 g ZPAS imposed. Subsequent discharge capacity tests showed that while these cells suffered some loss of discharge capacity, all cells could deliver the accepted standard of 80% of their rated electrical capacity for 3 hours. When two of the same cells were exposed to the second stage, higher g-level tests, both cells again provided instantaneous uninterrupted power. Subsequent capacity tests showed both of these cells to have capacities well below the accepted standard of 80%. Four of the cells were disassembled for examination and metallurgical analyses. The examination showed that all plates and separators were in very good condition.

NUREG/CR-4097: TEST SERIES 4:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED EXIDE EMP-13 BATTERY CELLS. BONZON,L.L.; HENTE,D.B. Sandia National Laboratories. April 1985. 119pp. 8507020430. SAND84-2630. 31309:321.

This report, the fourth in a test series of an extensive seismic research program, covers the testing of 27-year old lead antimony Exide EMP-13 cells from the recently decommissioned Shippingport Atomic Power Station. The Exide cells were tested in two configurations using a trixial shake table: single-cell tests, rigidly mounted; and multicell (five-cell) tests, mounted in a typical battery rack. A total of nine electrically active cells was used in the two different cell configurations. None of the nine cells failed during the actual seismic tests when a range of ZPAs up to 1.5 g was imposed. Subsequent discharge capacity tests of five of the cells showed, however, that none of the cells could deliver the accepted standard of 80% of their rated electrical capacity for 3 hours. In fact, none of the 5 cells could deliver more than 33% capacity. Two of the seismically tested cells and one untested, low capacity well were disassembled for examination and metallurgical analyses. The inspection showed the cells to be in poor condition. The negative plates in the vicinity of the bus connections were extremely weak, the positive buses were corroded and brittle, negative and positive active material utilization was extremely uneven, and corrosion products littered the cells.

NUREG/CR-4100: EVALUATION OF INSTRUMENTAL METHODS FOR THE MEASUREMENT OF YELLOWCAKE EMISSIONS. LEPEL,E.A.; THOMAS,V.W. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1985. 35pp. 8503200121. PNL-5350. 29470:306.

An evaluation of current sampling and analysis methods used for monitoring yellowcake emissions from uranium mill exhausts was performed by Pacific Northwest Laboratory. The representatives of once per quarter sampling was felt to be questionable. A more representative sample would be obtained by a continuous sampling system. The analysis could be performed by relatively newer instrumental methods. Direct-spectrometric and isotopically excited x-ray fluorescence instrumental analysis methods were evaluated. Because of a redirection in funding, the evaluation was not completed in terms of identifying instrumental interferences and field testing of the chosen methods. However, in light of readily available technology, a preferred method for sampling and analysis of yellowcake from uranium mill exhausts is proposed. This method would sample the exhaust stacks continuously using a continuous, automatic, isokinetic stack sampler with deposition of the exhaust gas particulates onto filter paper. The deposited particulates would then be analyzed by x-ray fluorescence using (57)Co as an excitation source. It is also recommended that a paper-tape sampler that houses an isotopic excitation source and detector be interfaced to a continuous stack sampler. This system would require evaluation and field testing after development.

NUREG/CR-4101: ASSAY OF LONG-LIVED RADIONUCLIDES IN LOW-LEVEL WASTES FROM POWER REACTORS. CLINE,J.E.; NOYCE,J.R.; COE,L.J.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). April 1985. 615pp. 8505100034. 30271:001.

The 10 CFR Part 61 waste classification system includes several nuclides which are difficult to assay without expensive radiochemical methods. In order for waste generators to classify wastes practically, NRC Staff has recommended the use of correlation factors to scale the difficult-to-measure nuclides with nuclides which can be measured more easily (i.e., gamma emitters such as (60)Co or (137)Cs). In this study, Science Applications International Corporation (SAIC) performed complete radiochemical assays for all the 10 CFR Part 61 waste classification nuclides on over 100 samples. These data, along with almost 800 other samples in the SAIC data base, were used to assess the validity of correlation factors suggested for use for nuclear power plant wastes. Specific generic correlation factors are recommended with other approaches to correlate nuclides for which generic scaling factors are not defensible.

NUREG/CR-4103: USES OF HUMAN RELIABILITY ANALYSIS PROBABILISTIC RISK ASSESSMENT RESULTS TO RESOLVE PERSONNEL PERFORMANCE ISSUES THAT COULD AFFECT SAFETY. O'BRIEN,J.N.; SPETTELL,C.M. Brookhaven National Laboratory. October 1985. 100pp. 8601020811. BNL-NUREG-51849. 34118:120.

This report is the first in a series which documents research aimed at improving the usefulness of Probabilistic Risk Assessment (PRA) results in addressing human risk issues. This first report describes the results of an assessment of how well currently available PRA data addresses human risk issues of current concern to NRC. Findings indicate that PRA data could be far more useful in addressing human risk issues with modification of the development process and documentation structure of PRAs. In addition, information from non-PRA sources could be integrated with PRA data to address many other issues.

NUREG/CR-4104: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL. Field Evaluation/Validation. SIEGEL,A.I.; WOLF,J.J.; BARTTER,W.D.; et al. Oak Ridge National Laboratory. August 1985. 86pp. 8512120145. ORNL/TM-9503. 33870:175.

This report discusses the results of efforts focused upon the evaluation of the practicality, acceptability, usefulness and validity of the Maintenance Personnel Performance Simulation (MAPPS) model. Subsequent to the completion of model development efforts, MAPPS was subjected to a number of evaluations that included table-top analyses, case study analyses, analyses involving the correlation of MAPPS output to the consensus of subject matter experts, and analyses involving the correlation of MAPPS output to directly observed empirical data gathered in the field. Results of these evaluation efforts are reported in this NUREG/CR. Overall results indicate that no unduly burdensome practicality issues were identified. IN addition, identified user groups (the U.S. Nuclear Regulatory Commission, nuclear power plant maintenance supervisors, and architect and engineering firms) rated the model as having relatively high acceptability and found the model to be very useful for a number of specific problems relating to user group interests. Predictive validity was also shown to be in good agreement with both the consensus of subject matter experts and with the empirical data observed in the field. The positive results of the accomplished evaluation efforts indicate that the model is ready for widespread application.

NUREG/CR-4105: AN ASSESSMENT OF THERMAL GRADIENT TUBE RESULTS FROM THE HI SERIES OF FISSION PRODUCT RELEASE TESTS. NORWOOD,K.S. Oak Ridge National Laboratory. May 1985. 64pp. 8505230531. ORNL/TM-9506. 30549:192.

A thermal gradient tube was used to analyze fission product vapors released from fuel heated in the HI test series. Complete deposition profiles were obtained for Cs, I, Ag, and Sb. The cesium profiles were complex and probably were dominated by Cs-S-O compounds formed by release of sulfur from furnace ceramics. The iodine profiles were simple, indicating that more than 99.5% of the released iodine behaved as a single nonvolatile species, probably CsI. Mass transfer coefficients for this species onto platinum were estimated to be 1.9 to 5.8 cm/s. Silver was probably released in elemental form, condensed to an aerosol, and captured by filters. Antimony was released as the element and reacted rapidly with platinum (or gold) as it deposited. Antimony profiles were calculated a priori with some success. A method was developed for isolating tellurium platinum and mixed fission products in a form suitable for neutron activation analysis. The platinum samples were completely dissolved in acid (HC1/HNO(3), and the tellurium was precipitated on selenium carrier by reduction. Finally, tellurium was loaded onto Dowex 1X-4 ion-exchange resin for activation and analysis. Tellurium recovery was 88%, and the theoretical sensitivity was 3 ng.

NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest. BRYAN,R.H.; BASS,B.R.; BOLT,S.E.; et al. Oak Ridge National Laboratory. April 1985. 288pp. 8506060815. ORNL-6135. 30770:024.

The first pressurized-thermal-shock test of a 148-mm-thick steel pressure vessel with a 1-m-long flaw was performed to investigate fracture behavior of a vessel under conditions relevant to a flawed nuclear reactor pressure vessel during an overcooling accident. The objectives were to observe crack arrest and stability on the ductile upper shelf and effects of warm prestressing on crack initiation. Three coordinated pressure and thermal transients were imposed on the vessel, which was preheated to 290 degrees centigrade. Two episodes of crack propagation and arrest occurred. The thermal transients were induced by coolant at -29 degrees centigrade to 15 degrees centigrade. Pressure transients were as high as 94.4 MPa. The experimental objectives were attained. The inhibiting effects of warm prestressing were definitely demonstrated. Crack propagation was nearly pure cleavage, and arrest at 30K above the onset of the Charpy upper shelf was experienced in a positive K(I) gradient and with K(I) 300 MPa square root m. Fracture-mechanics analysis of brittle fracture based on small-specimen toughness measurements was reasonably accurate. Flaw evaluation by procedures of the ASME Boiler and Pressure Vessel Code conservatively predicted vessel failure, which did not occur.

NUREG/CR-4107: SEQUENTIAL TEST PROCEDURES FOR DETECTING PROTRACTED MATERIALS LOSSES. GOLDMAN,A.S. Los Alamos Scientific Laboratory. July 1985. 51pp. 8508150033. LA-10319-MS. 32196:285.

Sequential tests are required for detecting protracted (trickle) losses of strategic special nuclear materials from a single materials control unit (MCU). We compared applicable tests including modified versions of Page's test and power-one procedures. We used simulated data from a MCU in a conversion/fabrication process that tool into account process variations, materials holdup, and measurement uncertainties. Comparisons were made over a 60-day accounting period under different loss scenarios. Some important findings include: (1) No single procedure is best for all diversion scenarios. (2) Power-one procedures are best for protracted losses that occur early in the accounting period and Page's test is best for late loss occurrence. (3) If holdup process variations are not included in the Inventory Difference model but are present in the process, then assuming steady-state conditions, flase-alarm probabilities can double.

NUREG/CR-4108: DEVELOPMENT OF MC&A ALARM RESOLUTION PROCEDURES. SMITH,B.W. Battelle Memorial Institute, Pacific Northwest Laboratories. October 1985. 53pp. 8510250531. PNL-5154. 33225:154.

The NRC has proposed reform of the material control and accounting (MC&A) requirements for facilities authorized to possess and use formula quantities of strategic special nuclear material (SSNM). The purpose of the reform is to strengthen MC&A capabilities by requiring more timely detection of possible SSNM losses and by providing for more rapid and conclusive resolution of discrepancies. This report provides guidance for developing a set of procedures to resolve alarms from the proposed near-real-time loss detection system. An alarm resolution program distinguishes between a system error and an actual loss of nuclear material. An alarm resolution program consists of procedures to identify causes of alarms, criteria for accepting resolution of alarms, and a program to monitor resolution effectiveness. Development of alarm resolution procedures consists of identifying potential causes of alarms, ordering the general elements of resolution, and determining criteria for accepting resolution. A monitoring program is performed to ensure consistent and acceptable application of the procedures.

NUREG/CR-4109: TRAC-PF1 ANALYSES OF POTENTIAL PRESSURIZED-THERMAL SHOCK TRANSIENTS AT CALVERT CLIFFS/UNIT 1,A Combustion Engineering PWR. SPRIGGS,G.D.; KOENIG,J.E.; SMITH,R.C. Los Alamos Scientific Laboratory. April 1985. 355pp. 8504220382. LA-10321-MS. 29953:331.

Los Alamos National Laboratory participated in a program to assess the risk of a pressurized thermal shock (PTS) to the reactor vessel during a postulated overcooling transient in a pressurized water reactor (PWR). We provided the thermal-hydraulic analyses of three general accident categories: steamline breaks, runaway-feedwater transients, and small-break loss-of-coolant accidents. These postulated accidents included multiple operator and equipment failures. Results were provided to Oak Ridge National Laboratory (ORNL) who plan to determine the probability of vessel failure and accident occurrence for an overall assessment of PTS risk. Our study was performed for a Combustion Engineering PWR, Calvert Cliffs/Unit 1, using the Transient Reactor Analysis Code (TRAC-PF1). We found the results of the analyses to be very sensitive to the initial conditions of the plant. If the plant was initially at hot-zero power (compared to full power), the decay heat was much less, which made it possible for the same accident initiator to produce significantly lower downcomer temperatures. However, routine operator actions may reduce the consequences of any of these simulated accidents if the prescribed pressure-temperature relationships are followed.

NUREG/CR-4110: REPOSITORY SITE DATA REPORT FOR UNSATURATED TUFF,YUCCA MOUNTAIN,NEVADA. TIEN,P.L.; SIEGEL,M.D.; UPDEGRAFF,C.D.; et al. Sandia National Laboratories. November 1985. 400pp. 8512100278. SAND84-2668. 33835:001.

The U.S. Department of Energy is currently considering the thick sequences of unsaturated, fractured tuff at Yucca Mountain, on the southwestern boundary of the Nevada Test Site, as a possible candidate host rock for a nuclear-waste repository. Yucca Mountain is in one of the most arid areas in the United States. The site is within the south-central part of the Great Basin section of the Basin and Range physiographic province and is located near a number of silicic calderas of Tertiary age. Although localized zones of seismic activity are common throughout the province, and faults are present at Yucca Mountain, the site itself is basically aseismic. No data are available on the composition of ground water in the unsaturated zone at Yucca Mountain. It has been suggested that the composition is bounded by the compositions of water from wells USW-H-3, UE25p-1, J-13, and snow or rain. There are relatively few data available from Yucca Mountain on the moisture content and saturation, hydraulic conductivity, and characteristic curves of the unsaturated zone. The available literature on thermomechanical properties of tuff does not always distinguish between data from the saturated zone and data from the unsaturated zone. Geochemical, hydrologic, and thermomechanical data available on the unsaturated tuffs of Yucca Mountain are tabulated in this report. Where the data are very sparse, they have been supplemented by data from the saturated zone or from areas other than Yucca Mountain.

NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET-GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE,AND SODIUM CHLORIDE. KERSCHNER,H.F.; ETTINGER,H.J.; DEFIELD,J.D.; et al. Los Alamos Scientific Laboratory. April 1985. 62pp. 8504300121. LA-9985-MS. 30070:220.

Respirators fitted with high-efficiency particulate (HEPA) cartridge filters are designed to remove dust, fumes, mists, and airborne particulate radionuclides. If these filters are to be reused, a Quality Assurance (QA) program must be established to ensure that filter efficiency remains greater than 99.97 per cent. The standard method for performing QA testing is to challenge the filter with a thermally generated aerosol of 0.3-m-diam di-2-ethylhexyl phthalate (DEHP). Because of potential toxicological and other problems associated with the use of monodisperse DEHP, an investigation to study measured filter efficiencies on an HEPA respirator filter population, using several recommended replacement aerosols, has been conducted. Aerosols compared in this study were thermally generated di-2-ethylexyl sebecate (DEHS), thermally generated DEHP, air-jet-generated DEHS, and air-jet-generated salt (NaC1). The study also focused on determining compatibility for parallel use of aerosols generated for respirator-fit testing for use in QA filter testing. Results indicate that a polydisperse air-jet-generated aerosol of DEHS can substitute for thermally DEHP as a method of providing QA testing of HEPA respirator filters and that equipment used in the study designed for respirator quantitative-fit testing can easily be modified to perform this function.

NUREG/CR-4112 V01: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task A:IEEE Flame Test. * Underwriters Laboratory, Inc. January 30, 1985. 105pp. 8502130459. US 75-1. 28874:250.

The flame test in the Institute of Electrical and Electronics Engineers (IEEE) Standard 383 was investigated. The investigation was to develop possible modifications in test equipment and test procedure that would increase the repeatability of results and provide additional information useful in assessing cable system performance in response to a real fire. Several fire experiments were conducted varying different test parameters. The experimental data were analyzed and modifications of both test equipment and test procedure were developed to increase repeatability. These modifications were: An enclosure for the sample, defining cable damage; cable fastening and the cable tray to be used; establishing tolerances for exhaust of the enclosure; starting temperature of the ambient air cable sample; location of the burner and the flow rates of fuel and air into the burner. Suggested also, was to report the maximum flame height versus time and the rate of heat released versus time as additional information that would be useful in assessing cable system performance.

NUREG/CR-4112 V02: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task B:Firestop Test Method. * Underwriters Laboratory, Inc. January 1985. 71pp. 8502130568. US 75-2. 28922:351.

An experimental investigation was conducted to provide data concerning the effects that changes in pressure differential, fire exposure and sample construction have on firestop performance when exposed to a standard fire test. Fifty-one fire test experiments were conducted using pressure differentials between -12 to +120 Pa, different sample constructions and two fire exposure conditions. Findings were that small changes in pressure differential did not have a significant effect on firestop materials that did not have cracks or through openings to allow passage of gases during fire exposure. If the materials allowed passage for gases through cracks or holes, such as those left open after pulling a cable, changing the pressure differential affected the firestop performance. Also, it was demonstrated that changing the size of the opening; size, location and type of the penetrating items installed through the opening; and severity of fire exposure affected the performance of the firestop.

NUREG/CR-4114: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS.II. MEYER,R.E.; ARNOLD,W.D.; CASE,F.I. Oak Ridge National Laboratory. May 1985. 53pp. 8505210576. ORNL-6137. 30521:328.

Estimation of the rates of migration of nuclides from nuclear waste repositories requires knowledge of the interaction of these nuclides with the components of the geological formations in the path of the migration. These interactions will be dependent upon the valence state and speciation of the nuclide. Experiments designed to measure interaction of multivalent nuclides and minerals must therefore include study of the speciation of the nuclides. An electrochemical method of valence state control and solvent extraction analyses of the valence states were used to study a number of reactions of interest to HLW repositories. These include the reduction of Np(V) and Tc(VII) by crushed basalt and other minerals. For the reduction of Np(V) by basalt, the experiments indicate that the sorption of basalt increases with pH and that most of the Np is reduced to Np(IV) which is very difficult to remove from the basalt even if oxygenated tracer-free solution is added to the solution. For the experiments with Tc(VII), the results are considerably more complicated. Experiments were initiated to determine the solubility of Tc(IV) oxides. The results of these experiments are used to assess some of the techniques and methods currently used in safety analyses of proposed HLW repositories.

NUREG/CR-4115: INTERNATIONAL STANDARD PROBLEM 13 (LOFT EXPERIMENT L2-5). Final Comparison Report. BURTT,J.D. EG&G, Inc. January 1985. 229pp. 8502210258. EGG-2369. 29058:109.

LOFT Experiment L2-5 was designated International Standard Problem 13 by the Organization for Economic Cooperation and Development. Comparisons between measurements from Experiment L2-5 were made with calculations from 11 international participants using five different computer codes. LOFT Experiment L2-5 simulated a double ended guillotine cold leg rupture of a primary coolant loop of a large pressurized water reactor, coupled with a loss of offsite power.

NUREG/CR-4116: NUFEGQ-NP:A DIGITAL COMPUTER CODE FOR THE LINEAR STABILITY ANALYSIS OF BOILING WATER NUCLEAR REACTORS. PENG,S.J.; PODOWSKI,M.Z.; LAHEY,R.T. Rensselaer Polytechnic Institute, Troy, NY. August 1985. 437pp. 8509060217. 32502:001.

The phenomena of nuclear-coupled density-wave oscillations are of considerable importance in boiling water nuclear reactor (BWR) stability analysis. A state-of-the-art linear frequency domain digital computer code, NUFREQ-NP, has been developed for either forced or natural circulation BWR stability analysis. The NUFREQ-NP code can be excited by many external perturbations, including system pressure perturbation. It is based on one dimensional drift-flux thermal hydraulics, and allows for subcooled boiling, arbitrary nonuniform axial and radial power shapes, distributed local losses (e.g., spacers), point or multi-dimensional neutron kinetics, and detailed fuel element dynamics. It has been compared with both out-of-core and in-core data, and good agreement has been found.

NUREG/CR-4117: FAULTING AND JOINTING IN AND NEAR SURFACE MINES OF SOUTHWESTERN INDIANA. AULT,C.H.; HARPER,D.; SMITH,C.R.; et al. Indiana Geological Survey. January 1985. 39pp. 8502070553. 28795:316.

This project was directed towards the characterization of: (1) the known large faults in southern Indiana, i.e., the Georgetown Fault in Floyd County and the newly named Crandall Fault in Harrison County; and (2) the small scale fractures endemic to southwestern Indiana. The Georgetown and Crandall Faults are normal faults that have a maximum vertical displacement of about 65 feet. They are post-Valmeyeran and pre-Pleistocene in age and are probably the result of hinge-line deformation between the subsiding Illinois Basin and the Cincinnati Arch. In contrast, abundant small-scale faults and joints are related to regional compressive lithospheric stress or to sedimentologic processes that operated penecontemporaneously with deposition of the rocks in which they are found. Structures related to regional stress include small-scale thrust faults with displacements of a few inches to a few feet and joints that are widespread in mines and outcrops in rocks of Mississippian and Pennsylvanian age. The jointing and most of the small-scale thrust faulting indicate that southern Indiana is affected by the Midcontinent Stress Province in the northern part of the study area and by another stress field in the southern part. An east-west boundary can be defined between the two stress fields.

NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES. DENHAM,D.H.; RATHBUN,L.A.; BARNES,M.G.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 31pp. 8507030713. PNL-5361. 31314:036.

Decommissioning of a uranium processing site requires radiological surveys to: 1) identify buildings, equipment, and open land areas that require cleanup; 2) verify that cleanup operations have been successful; and 3) provide a record of the radiological condition of the site following cleanup. This report describes the instruments, measurements, quality assurance, and statistical procedures that can be used to perform pre-and post-cleanup surveys. The procedures described include: 1) gamma-radiation exposure-rate measurements using micro-R-meters, 2) beta-gamma measurements using Geiger-Mueller tubes, 3) wipe tests for surface contamination, and 4) soil analyses for (226)Ra and other (238)U daughters. Locations likely to have (226)Ra concentrations that exceed standards can be identified by gamma-radiation exposure rate measurements. Samples of soil or other material from location showing elevated exposure rates can then be analyzed for (226)Ra to determine the boundaries of areas that exceed standards. Beta-gamma measurements and wipe sample analyses can be used to determine whether uranium concentrations exceed standards for either fixed or removable contamination.

NUREG/CR-4119: INTEGRITY OF CONTAINMENT PENETRATIONS UNDER SEVERE ACCIDENT CONDITIONS FY84 ANNUAL REPORT. SUBRAMANIAN,C. Sandia National Laboratories. August 1985. 37pp. 8508210429. SAND85-0016. 32334:167.

This document is an annual report for FY84 on the NRC-funded program titled "Integrity of Containment Penetrations Under Severe Accident Conditions." The purpose of this program is to evaluate the behavior of seals and gaskets and major fixed and operable penetrations. The scope and objectives of this program are discussed as well as the test matrix, test facilities, and test procedures.

NUREG/CR-4120: MATHEMATICAL MODELING OF ULTIMATE HEAT SINK COOLING PONDS. POLICASTRO,A.J.; WASTAG,M.; DUNN,W.E.; et al. Argonne National Laboratory. March 1985. 276pp. 8504050372. ANL/ES-143. 29671:054.

A general treatment of ultimate heat sink (UHS) cooling pond thermal performance is proposed through the application of a three-dimensional grid model. Validation of the model has been shown through comparisons of predictions with data from a field and laboratory pond. The advantage of the model lies in its ability to determine the detailed character of the flow field whether it be one, two, or three dimensional. Existing models require a priori knowledge of the character and dimensionality of the flow field in such ponds. Application of the model to a prototype UHS pond revealed that the balance of physical mechanisms involved in the thermal hydraulics of these ponds is quite different than for ponds used in normal cooling. The small, heavily-loaded, irregularly-shaped nature of the UHS pond should, in many cases, lead to a vertically mixed pond with only a one-dimensional (longitudinal) variation in pond temperature.

NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS. CULLEN,W.H. Materials Engineering Associates, Inc. KEMPPAINEN,M.; HANNINEN,H.; et al. Finland, Govt. of. February 1985. 49pp. 8503040021. MEA-2053. 29198:223.

Fatigue crack growth rate tests, at a load ratio of 0.2, have been conducted on steels of low, medium and high sulfur contents (0.004%, 0.013% and 0.025%) in PWR water at both low and high flow rates. Crack growth rates show no dependence on flow rate, but are strongly dependent on sulfur content, with a large proportion of environmental assistance for the highest sulfur contents. Tests of low and high sulfur content steels at a load ratio of 0.7 show relatively little environmental assistance in either case. The fractography of these specimens shows the usual brittle appearance for environmentally-assisted fatigue crack growth. In addition, the opposing fracture surfaces match perfectly, indicating that little or no dissolution of the metal matrix has occurred, and there is very little plastic flow associated with the fatigue cracking process. The X-ray photoelectron emission examination of the fracture surface oxides shows that FeS and FeS(2) coexist in the oxide layer, suggesting that the conditions within the crack enclave involved near-neutral pH and cathodic potentials.

NUREG/CR-4122: A FORTRAN 77 PROGRAM AND USER'S GUIDE FOR THE CALCULATION OF PARTIAL CORRELATION AND STANDARDIZED REGRESSION COEFFICIENTS. IMAN,R.L.; SHORTENCARIER; JOHNSON,J.D. Sandia National Laboratories. August 1985. 54pp. 8508210437. SAND85-0044. 32333:270.

This document is for users of a computer program developed by the authors at Sandia National Laboratories. The computer program is designed to be used in conjunction with sensitivity analyses of complex computer models. In particular, this program is most useful in analyzing input-output relationships when the input has been selected using the Latin hypercube sampling program developed at Sandia (Iman and Shortencarier, 1984). The present computer program calculates the partial correlation coefficients and/or the standardized regression coefficients from the multivariate input to, and output from, a computer model. These coefficients can be calculated on either the original observations or on the ranks of the original observations. The coefficients provide alternative measures of the relative contribution (importance) of each of the various inputs to the observed output variations. Relationships between the coefficients and differences in their interpretations are identified. If the computer-model output has an associated time or spatial history then the computer program will generate a graph of the coefficients over time or space for each input-variable, output variable combination of interest, thus indicating the importance of each input over time or space. The computer program is user-friendly and written in FORTRAN 77 to facilitate portability.

NUREG/CR-4123: SEISMIC FRAGILITY OF REINFORCED CONCRETE STRUCTURES AND COMPONENTS FOR APPLICATION TO NUCLEAR FACILITIES. GERGELY,P. Lawrence Livermore National Laboratory. March 1985. 107pp. 8503280022. UCID-20164. 29547:260.

The failure and fragility analyses of reinforced concrete structures and elements in nuclear reactor facilities within the Seismic Safety Margins Research Program (SSMRP) at the Lawrence Livermore National Laboratory are evaluated. Uncertainties in material modeling, behavior of low shear walls, and seismic risk assessment for nonlinear response receive special attention. Problems with ductility-based spectral deamplification and prediction of the stiffness of reinforced concrete walls at low stress levels are examined. It is recommended to use relatively low damping values in connection with ductility-based response reductions. The study of static nonlinear force-deflection curves is advocated for better nonlinear dynamic response predictions. Several details of the seismic risk analysis of the Zion plant are also evaluated.

NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984. KUPPERMAN,D.S.; CLAYTOR,T.N.; PRINE,D.W. Argonne National Laboratory. April 1985. 39pp. 8505060509. ANL-85-5. 30190:287.

This progress report summarizes work performed by the Argonne National Laboratory and GARD, Inc. (Division of Chamberlain Mfg. Corp.) as subcontractor on NDE of stainless steel and on-line monitoring of LWRs during the twelve months from October 1983 to September 1984.

NUREG/CR-4125 V01: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 1:Guidelines For Utility Organization And Administration Plan. THURBER,J.A.; OLSON,J.; OSBORN,R.N.; et al. Battelle Human Affairs Research Centers. August 1985. 41pp. 8508230324. PNL-5374. 32358:304.

This report is a partial response to the requirements of Item I.B.1.1 of the "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0660, and is designed to serve as a basis for replacing the earlier NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources." The Guidelines are intended to provide guidance to the user in preparing a written plan for a proposed nuclear organization and administration. The purpose of the Workbook is to guide the NRC reviewer through a systematic review and assessment of a proposed organization and administration. It is the NRC's intention to incorporate these Guidelines and Workbook into a future revision of the Standard Review Plan (SRP), NUREG-0800. However, at this time the report is being published so that the material may be used on a voluntary basis by industry to systematically prepare or evaluate their organization or administration plans. Use of the report by the NRC would not occur until after it has been incorporated in the SRP.

NUREG/CR-4125 V02: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 2:Workbook For Assessment Of Organization And Management. THURBER,J.A.; OLSON,J.; OSBORN,R.N.; et al. Battelle Human Affairs Research Centers. August 1985. 92pp. 8508260008. PNL-5374. 32363:272.

See NUREG/CR-4125,V01 abstract.

NUREG/CR-4127 V01: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 1:Numerical Methods. HECK,C.L.; ANDERSEN,J.G. General Electric Co. November 1985. 66pp. 8512050366. EPRI NP-3987. 33769:032.

A complete technical basis for implementation of the 3-D fast numerics in TRACB04 is presented. The 3-D numerics is a generalization of the predictor/corrector method previously developed for the 1-D components in TRACB.

NUREG/CR-4127 V02: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 2:Models. CHU,K.H.; ANDERSEN,J.G.; CHEUNG,Y.K.; et al. General Electric Co. November 1985. 64pp. 8512050467. EPRI NP-3987. 33769:286.

TRAC-BWR (Transient Reactor Analysis Code) is a computer code for best estimate analysis of the thermal hydraulic conditions in a Boiling Water Reactor system. In this report, the development of new models and the implementation of the balance of plant models leading to the creaction of the TRACB04 version of the code, is described. The new models include an improved model for boron transport which accounts for non-uniform mixing and stratification, and a model for the interfacial heat transfer at two-phase levels. The balance of plant models (turbine, containment and heat exchanger) developed at INEL were evaluated, adapted, and implemented into TRACB04.

NUREG/CR-4127 V03: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 3:Development Assessment For Plant Application. CHEUNG,Y.K.; ANDERSEN,J.G.; CHU,K.H.; et al. General Electric Co. November 1985. 90pp. 8512050464. EPRI NP-3987. 33769:196.

The TRACB04 computer code has been developed under the model development tasks in the FIST Program. This report describes two developmental assessment calculations performed on BWR plants with TRACB04. A BWR/2 Design Basis Accident (DBA) including the containment response and a BWR/4 DBA with Low Pressure Coolant Injection (LPCI) water injected into the lower plenum were calculated and results of these cases were documented. These cases serve to test some of the new features of the TRACB04 (air field, containment model, "water packing" fixes and faster numerics in the three dimensional vessel component) and to demonstrate that the code has been assembled properly. They also provide best estimate LOCA results for the two plant types.

NUREG/CR-4128: BWR FULL INTEGRAL SIMULATION TEST (FIST) PHASE II TEST RESULTS AND TRAC-BWR MODEL QUALIFICATION. SUTHERLAND,W.A.; ALAMGIR,M.; FINDLAY,J.A.; et al. General Electric Co. October 1985. 200pp. 8511220149. EPRI NP-3988. 33604:051.

A full height BWR system simulator built under the Full Integral Simulation Test (FIST) program is used to investigate system responses for various transients. The test program consists of two test phases. This report provides a summary, discussion, highlights, and conclusions of the FIST Phase II tests. The Phase I tests are reported in NUREG/CR-3711, EPRI NP-3602, GEAP-30496. Eight matrix tests were conducted in the FIST Phase I. These tests investigated the large break, small break and steamline break LOCA's, as well as natural circulation and power transients. There are nine tests in Phase II of the FIST program. They include the following LOCA tests: BWR/6 LPCI line break, BWR/6 intermediate size recirculation break, and a BWR/4 large break. Steady state natural circulation tests with feedwater makeup performed at high and low pressure, and at high pressure with HPCS makeup, are included. Simulation of a transient without rod insertion, and with controlled depressurization, was performed. Also included is a simulation of the Peach Bottom turbine trip test. The final two tests simulated a failure to maintain water level during a postulated accident. A FIST program objective is to assess the TRAC code by comparisons with test data. Two post-test predictions made with TRACB04 are compared with Phase II test data in this report. These are for the BWR/6 LPCI line break LOCA, and the Peach Bottom turbine trip test simulation.

NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS. OWCZARSKI,P.C.; SCHRECK,R.I.; WINEGARDNER,W. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1985. 93pp. 8510040377. PNL-5379. 32857:295.

This report describes the technical bases and use of computer code ICEDF. ICDEF was developed to serve as a tool for calculating particle retention in pressurized water reactor (PWR) ice compartments during severe accidents. This report also serves as a complete user's guide for the most recent standalone version of ICDEF. A complete code description, code operating instructions, code listing, examples of the use of ICEDF, and a summary of a parameter sensitivity study support the use of code ICEDF.

NUREG/CR-4131: INVESTIGATION OF ALTERNATIVE MEANS TO ACCOMPLISH THE GOALS OF BIENNIAL ION CHAMBER CALIBRATION. CAMERON,J.R.; DEWERD,L.A.; GOETSCH,S.J.; et al. Wisconsin, Univ. of, Madison, WI. May 1985. 38pp. 8506030101. 30707:220.

The research described in this report was performed to investigate the feasibility of a mailed dosimetry system as an alternative method of achieving the goals of the present U.S. Nuclear Regulatory Commission requirement that ionization chambers used for calibration of cobalt-60 teletherapy units be calibrated every two years. Both thermoluminescent dosimeters (TLD's) and a diode detector unit was used in this study. A total of 20 hospitals in the states of Illinois, Iowa and Wisconsin participated in a program in which this dosimetry package was sent to each institution on three separate occasions. The physicist, physician or chief technologist was asked to deliver 1.50 Gray (150 rads) to the device, assuming the device was equivalent in radiation adsorption characteristics to human tissue. A treatment field size of 10cm by 10cm was chosen and the institution was requested to use their clinical source-to-surface distance. The accuracy of the beam localization as indicated by the coincidence of the light field with the radiation field was measured as well. The criterion for accuracy of dose delivery was plus minus 3.0mm. Only two hospitals during the course of the study had both a disagreement of more than 3mm between the light field and the radiation field. It is recommended that such a mailed dosimetry package be considered as an alternative to the present NRC requirement for biennial calibration of ionization chambers used to calibrate cobalt-60 teletherapy sources.

NUREG/CR-4133: NUCLEAR POWER SAFETY REPORTING SYSTEM IMPLEMENTATION AND OPERATIONAL SPECIFICATIONS. NEWTON,R.D.; IMS,J.R.; FINLAYSON,F.C. Aerospace Corp. November 1985. 117pp. 8512050321. ATR-85(5818)2. 33776:037.

This report is the last in a series investigating the feasibility of adapting a voluntary, anonymous, non-punitive, third-party managed reporting concept in a U.S. commercial nuclear industry/regulatory environment. Such a system is intended for use in identifying and quantifying, in an uninhibited manner, the factors that contribute to the occurrence of significant safety incidents which elicit either positive or negative responses from humans in U.S. nuclear power plants. This report specifies the elements of a Nuclear Power Safety Reporting System (NPSRS), along with operating procedures and forms to be used for accepting, integrating and processing reports submitted to the system. Also included is a taxonomy for collating and storing reports received from a variety of sources addressing myriad safety-related topics. A companion NUREG/CR-4132 presents the results of a limited evaluation of the technical specifications contained in this report.

NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES. CLAIRBORNE,H.C.; CROFF,A.G.; GRIESS,J.C.; et al. Oak Ridge National Laboratory. May 1985. 130pp. 8506130358. ORNL/TM-9522. 30867:350.

This document provides specifications for a model/methodology and approach that could be employed in determining post-closure repository environmental parameters relevant to high-level waste package performance for the Basalt Waste Isolation Project (BWIP). Guidance is provided on (1) the identity of the relevant repository environmental parameters (groundwater characteristics, temperature, radiation, and pressure), (2) the models/methodologies employed to determine the parameters, and (3) the input data base for the model/methodologies. Supporting studies included are (1) an analysis of potential waste package failure modes leading to identification of the relevant repository environmental parameters, (2) an evaluation of the credible range of the repository environmental parameters for the BWIP situation, and (3) a summary review of existing models/methodologies currently employed in determining repository environmental parameters relevant to waste package performance.

NUREG/CR-4136: SMOKE:A Data Reduction Package For Analysis Of Combustion Experiments. RATZEL,A.C.; KEMPKA,S.N.; SHEPHERD,J.E.; et al. Sandia National Laboratories. September 1985. 131pp. 8511190556. SAND83-2657. 33533:235.

A suite of computer codes, collectively referred to as SMOKE, has been developed to expedite the processing and analysis of data obtained from transient combustion tests. Instrumentation signals which can be processed using SMOKE include pressure sensors, gas and wall thermocouples, and different types of calorimetry such as Sandia-developed thin-film gauges, capacitance-type slug calorimeters, and commercial Gardon-type heat flux gauges. This package has been used to analyze data from combustion experiments conducted at the Sandia National Laboratories FITS facility and in hydrogen dewar at the Nevada Test Site. In this report, we discuss the theory and models used in the computer codes comprising SMOKE. Details of the data files, signal preparation, and processing procedures for executing SMOKE are provided. Sample data files and representative results using SMOKE are included.

NUREG/CR-4137: PRETEST PREDICTIONS FOR THE RESPONSE OF A 1:8-SCALE STEEL LWR CONTAINMENT BUILDING MODEL TO STATIC OVERPRESSURIZATION. CLAUSS,D.B. Sandia National Laboratories. July 1985. 53pp. 8508120552. SAND85-0175. 32147:107.

The analyses used to predict the behavior of a 1:8-scale model of a steel LWR containment building to static overpressurization are described and results are presented. Finite strain, large displacement, and nonlinear material properties were accounted for using finite element methods. Three-dimensional models were needed to analyze the penetrations, which included operable equipment hatches, personnel lock representations, and a constrained pipe. It was concluded that the scale model would fail due to leakage caused by large deformations of the equipment hatch sleeves.

NUREG/CR-4138: DATA ANALYSES FOR NEVADA TEST SITE (NTS) PREMIXED COMBUSTION TESTS. RATZEL,A.C. Sandia National Laboratories. July 1985. 179pp. 8507250130. SAND85-0135. 31794:341.

This report provides results from an in-depth analysis of twenty-one premixed large-scale combustion experiments sponsored by the NRC and EPRI and conducted by EG&G at the Nevada Test Site (NTS). These experiments were performed in a 2048 m(3) spherical vessel (hydrogen dewar) with mixtures of hydrogen, steam, and air ignited by glow plugs or heated resistance coils. Hydrogen concentrations ranged from 5 to 13% (by volume) and steam concentrations from 4 to 40%. Several tests also incorporated spray systems and/or fans which enhanced the combustion rate and significantly altered the postcombustion gas cooling. Data provided by EPRI from instrumentation designed to characterize the thermal environment in the dewar during and following combustion have been evaluated. The data reduction package SMOKE has been used to process data from thin-film gauges, commercial heat flux gauges, capacitance calorimeters, gas and wall thermocouples, and pressure sensors. Local measurements of the heat transfer are provided from the calorimetry, and global averages are inferred from the pressure. Instrumentation "goodness" for each test is assessed based on the raw data and on comparisons of local and global results. Graphical and tabular results are provided for each test, and trends observed from the results are reported.

NUREG/CR-4139: THE MAILED SURVEY:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL. MCGUIRE,M.V.; WALSH,M.E.; MORISSEAU,D.S.; et al. Battelle Human Affairs Research Centers. May 1985. 87pp. 8505100041. PNL-5381. 30270:125.

This report describes a mailed survey of operations personnel at a sample of commercial nuclear power plants. The survey was conducted for the U.S. Nuclear Regulatory Commission (NRC) as part of the Operator Feedback Project. The survey sought to collect information on topics of concern to the NRC and to assess the feasibility of a mailed survey on an information collection mechanism. Participants in the survey were 520 personnel at 26 nuclear power plants representing all five NRC regions. The individual participants completed and returned by mail a ten-page questionnaire. This contained questions about operations crew practices, including work and shift schedules, operations shift crew staffing, the shift technical advisor position, respondents' own backgrounds, the questionnaire, and other information-collection techniques. Results of the survey offer some insight on operations crew practices at the plants participating in the survey. Survey results also suggest that the mailed survey is an information-collection technique that can be used effectively to obtain feedback for the NRC from operations personnel.

NUREG/CR-4140: DOMINANT ACCIDENT SEQUENCES IN OCONEE-1 PRESSURIZED WATER REACTOR. DEARING,J.F.; HENNINGER,R.J.; NASSERSHARIF,B.; et al. Los Alamos Scientific Laboratory. April 1985. 112pp. 8506240647. LA-10351-MS. 31149:230.

A set of dominant accident sequences in the Oconee-1 pressurized water reactor was selected using probabilistic risk analysis methods. Because some accident scenarios were similar, a subset of four accident sequences was selected to be analyzed with the Transient Reactor Analysis Code (TRAC) to further our insights into similar types of accidents. The sequences selected were loss-of-feedwater, small-small break loss-of-coolant, loss-of-feedwater-initiated transient without scram, and interfacing systems loss-of-coolant accidents. The normal plant response and the impact of equipment availability and potential operator actions were also examined. Strategies were developed for operator actions not covered in existing emergency operator actions not covered in existing emergency operator guidelines and were tested using TRAC simulations to evaluate their effectiveness in preventing core uncovery and maintaining core cooling.

NUREG/CR-4141: CONTAINMENT PURGE AND VENT VALVE TEST PROGRAM FINAL REPORT. STEELE,R.; WATKINS,J.C. EG&G Idaho, Inc. (subs. of EG&G, Inc.). September 1985. 79pp. 8511010188. EGG-2374. 33304:164.

This report presents the results of the containment purge and vent valve test program, conducted under the sponsorship of the United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. The test program investigated valve functionality and leak integrity. Three nuclear designed butterfly valves typical of those used in domestic nuclear power plant containment purge and vent applications were tested. For a comparison of response, two valves of the same size with differing internal designs were tested. For extrapolation insights, a larger sized valve was also tested. The valve experiments were performed with various piping configurations and valve disc orientations to the flow to simulate various installation options in field applications. As a standard for comparing the effects of the installation options, testing was also performed in a standard ANSI test section. Dynamic flow tests were performed over the range of a design basis accident. Leak integrity testing was also performed and extended into severe accident conditions. Analysis of the test results produced a technical basis to assess industry purge and vent valve closing torque extrapolation methodology and quantified the influence of worst case piping geometry on valve torque response. It was also determined that some valve designs will leak in single isolation when exposed to design basis and severe accident environments.

NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences. KHATIB-RAHBAR; PRATT,W.; LUDEWIG,H.; et al. Brookhaven National Laboratory. September 1985. 75pp. 8510020257. BNL-NUREG-51907. 32829:198.

A technical review and evaluation of the Millstone Unit 3 Probabilistic Safety Study has been performed. It was determined that: (1) long-term damage indices (latent fatalities, person-rem, etc.) are dominated by late failure of the containment, (2) short-term damage indices (early fatalities, etc.) are dominated by bypass sequences for internally initiated events, while severe seismic sequences can also contribute significantly to early damage indices. These overall estimates of severe accident risk are extremely low compared with other societal sources of risk. Furthermore, the risks for Millstone-3 are comparable to risks from other nuclear plants at high population sites. Seismically induced accidents dominate the sever accident risks at Millstone-3. Potential mitigative features were shown not to be cost-effective for internal events. Value-impact analyses for seismic events showed that a manually actuated containment spray system might be cost-effective.

NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS. DAVIS,T.C.; SHAFAGHI,A.; KURTH,R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1985. 69pp. 8504220341. PNL-5389. 29946:001.

This report presents a method for focusing additional research on aging phenomena that affects nuclear power plant components. Specifically, the method ranks components using a risk aging sensitivity measure that describes the change in risk due to changes in component failure rate. Describing the aging phenomena and the resulting time-dependent component failure rate changes is beyond the scope of this study. The applications use average components unavailability equations currently employed in PRAs to calculate the risk aging sensitivity. A more exact calculation is possible by using unavailability equations that include the time-dependent characteristics of component failure rates; however, these time-dependent characteristics are not well-known. The risk aging sensitivity measure presented here is, therefore, segregated from these time-dependent effects and addresses only the time-independent portion of aging phenomena. The results identify the component types that show the most potential for risk change due to aging phenomena. Future research on the time-dependent portion of aging phenomena for these component types is needed to completely describe the risk impact due to component aging.

NUREG/CR-4145: EARTHQUAKE RECURRENCE INTERVALS AT NUCLEAR POWER PLANTS: ANALYSIS AND RANKING. HILEMAN,J.A.; KNOPOFF,L.; MANN,N.R.; et al. Earth Technology Co. March 1985. 142pp. 8503200110. 29471:001.

Five methods for estimating earthquake recurrence were ranked. The methods represent those used, or proposed, in nuclear power plant studies through 1982 and include Log Linear Poisson, Extreme Value, Semi-Markov, Bayesian, and Uniform Hazard Method. Ranking focused on recurrence estimates for earthquake sources, excluding attenuation and site response. Scores were assigned to each method for 12 criteria such as accuracy, use of geologic data, and subjective judgment. Criteria scores were weighted by their importance and summed. Different scoring and weighting schemes were used to identify any sensitivities. To aid in scoring statistical criteria, methods were tested on synthetic earthquake catalogs with known statistics, and natural catalogs were tested against theoretical magnitude distributions. The uniform Hazard Method scored high because, in principal, expert judgement draws upon all seismologic knowledge. The Bayesian Method scored low because data requirements are severe for practical cases. The other methods were intermediate. These observations seem insensitive to scorer, scoring approach, or weighting scheme. The semi-Markov Method scores were sensitive to the weighting scheme.

NUREG/CR-4146: SIMULATION OF AN EPRI-NEVADA TEST SITE (NTS) HYDROGEN BURN TEST AT THE CENTRAL RECEIVER TEST FACILITY. DANDINI,V.J.; ARAGON,J.J. Sandia National Laboratories. October 1985. 77pp. 8510280470. SAND85-0205. 33228:002.

In order to augment results obtained from the hydrogen burn tests performed by the Electric Power Research Institute (EPRI) at the Nevada Test Site (NTS), a series of tests was conducted at the Sandia National Laboratories Central Receiver Test Facility (CRTF). The CRTF tests simulated a 13 volume-percent burn from the EPRI-NTS series. During the tests, the responses of several specimens of nuclear power plant safety-related equipment were monitored when subjected to the simulated hydrogen burn. The specimens were pressure transmitters, solenoid valves, and single and multiconductor electric cables. All were nuclear service qualified. The simulation was conducted with and without steam in the vicinity of the test specimens. Prior to exposure, metallic specimens were preheated to temperature corresponding to the precombustion environment in the EPRI-NTS test vessel.

NUREG/CR-4147: THE EFFECT OF ENVIRONMENTAL STRESS ON SYLGARD 70 SILICONE ELASTOMER. BUCKALEW,W.H.; WYANT,F.J. Sandia National Laboratories. May 1985. 92pp. 8506240313. SAND85-0209. 31157:001.

Dow Corning Sylgard 170 Silicone Elastomer has been investigated to characterize its response to accelerated thermal aging, radiation exposure, and its behavior under applied compressive forces. Sylgard 170 response to accelerated thermal aging suggests the material properties are not particularly age dependent. Radiation exposures, however, produce significant, monotonic changes in both elongation and hardness with increasing absorbed radiation dose. Elastomer response to an applied compressive force was strongly dependent on environment temperature and degree of material confinement. Variations in temperature produced large changes in compressive forces applied to confined samples. Attempts to mitigate force fluctuations by means of pressure relief paths resulted in total loss of the applied compressive force. Thus, seal applications employing this elastomer in Class 1E equipment required to function during or following an accident should consider the potential loss of compressive force from long-term aging and potential LOCA-temperature transient conditions.

NUREG/CR-4149: ULTIMATE PRESSURE CAPACITY OF REINFORCED AND PRESTRESSED CONCRETE CONTAINMENT. SHARMA,S.; WANG,Y.K.; REICH,M. Brookhaven National Laboratory. May 1985. 95pp. 8506130467. BNL-NUREG-51857. 30901:328.

This report presents the results of an in-depth evaluation of current modeling techniques and analysis procedures for determining ultimate pressure capacity of reinforced and prestressed concrete containments. The material models used for describing the nonlinear material behavior of concrete and steel are reviewed in detail. Special attention is focused on post-cracking behavior of concrete which controls one of the containment failure modes, i.e., the shear failure. Various finite element idealizations utilized for containment analysis are reviewed. The effects of major assumptions pertaining to containment geometry, basemat restraint, finite element mesh, rebar locations and orientations, are evaluated. Finally, failure analyses of two selected reinforced and prestressed concrete containments are performed and results are compared with those presented in the literature.

NUREG/CR-4150: EPICOR-II RESIN DEGRADATION RESULTS FROM FIRST RESIN SAMPLES OF PF-8 AND PF-20. MCCONNELL,J.W.; SANDERS,R.D. EG&G Idaho, Inc. (subs. of EG&G, Inc.). July 1985. 52pp. 8508150017. EGG-2376. 32190:267.

The 28 March 1979 accident at Three Mile Island Unit 2 released approximately 560,000 gallons of contaminated water to the Auxiliary and Fuel Handling Buildings. The water was decontaminated using a demineralization system called EPICOR-II developed by Epicor,Inc. The Low-Level Waste Data Base Development--EPICOR-II Resin/Liner Investigation Project, funded by the U.S. Nuclear Regulatory Commission, is studying the chemical and physical conditions of the synthetic ion exchange resins found in several EPICOR-II prefilters. The work is being done by EG&G Idaho,Inc. at the Idaho National Engineering Laboratory. This report summarizes results and analyses of the first sampling of ion exchange resins from EPICOR-II prefilters PF-8 and -20. Results are compared with baseline data from tests performed on unirradiated Epicor, Inc. resins to determine if degradation has occurred due to the high internal radiation dose received by the EPICOR-II resins. Results also are compared with recent findings on resin degradation by Battelle Columbus Laboratories and Brookhaven National Laboratory.

NUREG/CR-4151: INTEGRATION OF EMERGENCY ACTION LEVELS WITH COMBUSTION ENGINEERING EMERGENCY OPERATING PROCEDURES.By Use Of Combustion Engineering Owners Group Emergency Operating Procedure Technical Guidelines. FALETTI,D.W.; JAMISON,J.D. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1985. 137pp. 8509250160. PNL-5392. 32753:295.

Pacific Northwest Laboratory, under contract to the U.S. Nuclear Regulatory Commission developed a method for integrating Emergency Action Levels (EALs) with plant-specific Emergency Operating Procedures (EOPs) using the Combustion Engineering Owner's Group Emergency Operating Procedure Technical Guidelines (CEOG EOPTGs). Using the Combustion Engineering Owner's Group Technical Guidelines document, a set of emergency class definitions and criteria were developed based on the status of the three main fission product barriers (fuel cladding, primary coolant system and containment). The EOPTGs were then annotated to point out where, in a symptom/function-based EOP patterned after the EOPTG, the inferred plant condition is such that a specific EAL may have been exceeded. After the EOPTGs have been annotated, the proposed method was tested by applying it to the reactor accident sequences that were shown in the reactor safety study to dominate accident risk to determine if an EAL set linked to the EOP annotations would produce timely and accurate classification of the risk-dominant sequences. Additional annotations and additions to the EOPTGs were developed and the revised annotations were shown to produce timely and accurate event classifications for all the accident sequences.

NUREG/CR-4152: AN INDEPENDENT SAFETY ORGANIZATION. KATO,W.Y.; WEINSTOCK,E.V.; CAREW,J.F.; et al. Brookhaven National Laboratory. February 1985. 327pp. 8502260121. BNL-NUREG-51858. 29109:001.

Brookhaven National Laboratory has conducted a study on the need and feasibility of an independent organization to investigate significant safety events for the Office for Analysis and Evaluation of Operational Data, USNRC. This is being carried out in response to a Congressional request to the NRC for such a study. The study consists of three parts: the need for an independent organization to investigate significant safety events, alternative organizations to conduct investigations, and legislative requirements. The determination of need was investigated by reviewing current NRC investigation practices, comparing aviation and nuclear industry practices, and interviewing a spectrum of representatives from the nuclear industry, the regulatory agency, and the public sector. The advantages and disadvantages of alternative independent organizations were studied, namely, an Office of Nuclear Safety headed by a director reporting to the Executive Director for Operations (EDO) of NRC; an Office of Nuclear Safety headed by a director reporting to the NRC Commissioners; a multi-member NTSB-type Nuclear Safety Board independent of the NRC. The costs associated with operating a Nuclear Safety Board were also included in the study. The legislative requirements, both new authority and changes to the existing NRC legislative authority, were studied. These legislative requirements were based upon the Edwards-Udall Bill H.R. 6390 introduced in the 96th Congress and study of the NRC Organization Act.

NUREG/CR-4153: APPLICATIONS OF FOREIGN PROBABILISTIC SAFETY ASSESSMENT EXPERIENCE TO THE U.S. NUCLEAR REGULATORY PROCESS. ANDREWS,W.B. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1985. 163pp. 8503130128. PNL-5388. 29359:207.

This report is a summary of applications of probabilistic safety assessment (PRA) in the United States and foreign countries. It is intended to stimulate discussion on the applicability of foreign experience to the United States, provide information on foreign safety technology development and focus the United States goals for future participation in the activities of the Committee for the Safety of Nuclear Installations (CSNI), Principle Working Group 5 (PWG5). Results indicate that the United States leads the surveyed countries in the completion and application of comprehensive PSAs of public safety impacts from nuclear power plants. European experience has focused on the use of reliability analyses in support of design and operational decisions. It is recommended that use of probabilistic analyses be expanded in the United States for engineering applications based on the success in European countries.

NUREG/CR-4155: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:NORTHWESTERN UNIVERSITY PERFORATED-PLATE CCFL TESTS. DOBRANICH,D. Sandia National Laboratories. April 1985. 42pp. 8505060503. SAND85-0172. 30190:241.

The TRAC-PF1/MOD1 independent assessment project at Sandia is part of an overall effort funded by the NRC to determine the ability of various systems codes to predict the detailed thermal/hydraulic response of LWRs during accident and off-normal conditions. As part of this effort, calculations for some of the Northwestern University perforated-plate CCFL tests have been performed. Two input models were constructed to represent the rectangular test channel: a two-dimensional model and a faster-running one-dimensional model. The results of both models indicate that for high water flow rates, with the water injected vertically above a perforated plate, TRAC overpredicts the steam flow rate necessary for complete weeping (CCFL). However, for flow conditions more typical of PWR transients, TRAC provides a reasonable prediction of weeping.

NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS. SUBUDHI,M.; BURNS,E.L.; TAYLOR,J. Brookhaven National Laboratory. June 1985. 150pp. 8511180647. BNL-NUREG-51861. 33506:297.

This report provides an aging assessment of electric motors and was conducted under the auspices of the NRC Nuclear Plant Aging Research Program (NPAR). The objectives of this program are to identify concerns related to the aging and service wear of equipment operating in nuclear power plants, to assess their possible impact on plant safety, to identify effective inspection, surveillance and monitoring methods and to recommend suitable maintenance practices for mitigating aging related concerns and diminish the rate of degradation due to aging and service wear. Motor design and materials of construction are reviewed to identify age-sensitive components. Operational and accidental stressors are determined, and their effect on promoting aging degradation is assessed. Failure modes, mechanisms, and their effect on promoting aging degradation is assessed. Failure modes, mechanisms, and causes have been reviewed from operating experiences and existing data banks. The study has also included consideration for the seismic correlation of age-degraded motor components. The aforementioned reviews and assessments were assimilated to characterize the di-electric, rotational, and mechanical hazards on motor performance and operational readiness. The functional indicators which can be monitored to assess motor component deterioration due to aging or other accidental stressors are identified. Conforming with the NPAR strategy as outlined in the program plan, the study also includes a preliminary discussion of current standards and guides, maintenance programs, and research activities pertaining to nuclear power plant safety-related electric motors.

NUREG/CR-4157: A SCIENTIFIC CRITIQUE OF AVAILABLE MODELS FOR REAL-TIME SIMULATIONS OF DISPERSION. LEWELLEN,W.S.; SYKES,R.I. Aeronautical Research Associates of Princeton. March 1985. 180pp. 8503200126. ARAP 472. 29468:191.

This report provides an evaluation of several available dispersion models to determine their suitability for providing the capability for estimating the effects of accidental discharges of radioactive material at nuclear power plants. A critique of the assumptions involved and a review of existing verification studies are made for models ranging from the Gaussian plume with straight line winds to models which attempt a complete solution of the primitive equations of motion. It is demonstrated that although even the simple models are capable of providing reasonably accurate predictions under ideal conditions, there are reasons to expect relatively severe limits on plume predictability when certain emission conditions are combined with certain meteorological conditions. The usefulness of a real-time dispersion model is thus likely to be dependent on a complementary estimate of the variability expected about the mean dispersion for the conditions existing at that time. This report is one of a set of three dealing with real-time dispersion models. The other two deal with the uncertainties involved in the deposition module of dispersion models and the results of testing some of the dispersion models reviewed in this report by comparing them with the data collected at the Idaho National Engineering Laboratory in July, 1981 during an NRC sponsored field test.

NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING. LEWELLEN,W.S.; VARMA,A.K.; SHENG,Y.P. Aeronautical Research Associates of Princeton. April 1985. 79pp. 8504250168. ARAP NO. 504. 30032:015.

The current generation of dispersion models contains very simple parameterizations of deposition processes. The analysis here looks at the physical mechanisims governing these processes in an attempt to see if more valid parameterizations are available and what level of uncertainty is involved in either these simple parameterizations or any more advanced parameterization. The report is composed of three parts. The first, on dry deposition model sensitivity, provides an estimate of the uncertainty existing in current estimates of the deposition velocity due to uncertainties in independent variables such as meteorological stability, particle size, surface chemical reactivity and canopy structure. The range of uncertainty estimated for an appropriate dry deposition velocity for a plume generated by a nuclear power plant accident is three orders of magnitude. The second part discusses the uncertainties involved in precipitation scavenging rates for effluents resulting from a nuclear reactor accident. The conclusion is that major uncertainties are involved both as a result of the natural variability of the atmospheric precipitation process and due to our incomplete understanding of the underlying process. The third part involves a review of the important problems associated with modeling the interaction between the atmosphere and a forest. It gives an indication of the magnitude of the problem involved in modeling dry deposition in such environments.

NUREG/CR-4159: COMPARISON OF THE 1981 INEL DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS. LEWELLEN,W.S.; SYKES,R.I.; PARKER,S.F. Aeronautical Research Associates of Princeton. May 1985. 202pp. 8505310421. ARAP NO. 505. 30670:001.

Results from simulations by 12 different dispersion models are compared with observations from an extensive field experiment at the Idaho National Engineering Laboratory in July, 1981. Comparisons were made based on hourly ground-level SF(6) samples, out to approximately 10 km from the 46 m release tower, both during and following 7 different 8-hour releases. Comparisons are also made for total integrated doses collected out to approximately 40 km. Within the limited range appropriate for Class A models this data comparison shows that neither the puff models or the transport and diffusion models agree with the data any better then the simple Gaussian plume models. The puff and transport and diffusion models do show a slight edge in performance in comparison with the total dose over the extended range appropriate for Class B models. The best model results for the hourly samples show approximately 40% calculated within a factor of two when a 15 degree uncertainty in plume position is permitted, and it is assumed that higher data samples may occur at stations between the actual sample sites. This is increased to 60% for the 12 hour integrated dose and 70% for the total integrated dose. None of the models reproduce the observed patchy dose patterns. This patchiness appears to be consistent with the inherent uncertainty associated with time averaged plume observations.

NUREG/CR-4160: HISTORICAL SUMMARY OF OCCUPATIONAL RADIATION EXPOSURE EXPERIENCE IN U.S. COMMERCIAL NUCLEAR POWER PLANTS. MOELLER,M.P.; STOETZEL,G.A.; MUNSON,L.H. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1985. 65pp. 8505070439. PNL-5404. 30214:293.

This report organizes existing data on occupational radiation exposure experience for consideration in the safety goal evaluation program. The report includes a review of occupational radiation exposures incurred by workers at commercial U.S. nuclear power plants. In addition, occupational radiation exposure information is presented for work performed at commercial U.S. nuclear power plants to meet regulatory actions and required backfits. This information identifies specific operations performed as part of these requirements. Where possible, actual radiation exposure histories are provided. A brief history of radiation dose limits and a review of the biological and health effects attributable to radiation exposure is included to provide a perspective on the development of radiation protection regulations.

NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt. BINNALL,E.P.; WOLLENBERG,H.A.; BENSON,S.M.; et al. Lawrence Berkeley Laboratory. May 1985. 95PP. 8506060810. UCID-20092. 30769:292.

This report addresses critical parameters specific to a repository in basalt, using the Columbia River Basalt Group as the principal example. For the purposes of this report, a parameter is considered to be a physical property whose value helps determine the characteristics or behavior of a repository system. Parameters which are defined as critical are those essential to evaluate and/or monitor leakage of radionuclides from the repository and to evaluate the need for retrieval. The parameters are considered with respect to the disciplines of geomechanics, geology, hydrology, and geochemistry and are rank ordered in terms of importance. The specific role of each parameter, specific factors affecting the measurement of each parameter, and the interrelationships between the parameters are considered in detail.

NUREG/CR-4164: DATA REPORT FOR THE TPFL TEE/CRITICAL FLOW EXPERIMENTS. ANDERSON,J.L.; OWCA,W.A. EG&G Idaho, Inc. (subs. of EG&G, Inc.). November 1985. 118pp. 8512270266. EGG-2377. 34081:188.

A series of experiments have been performed investigating the phenomena of liquid entrainment and vapor pull-through at a tee junction between a horizontal pipe and a small branchline. These experiments were performed under conditions of stratified steam-water flow at 3.4, 4.4, and 6.2 MPa in the 28.4cm diameter mainline, and critical flow through a nozzle installed in the branchline. Two orientations of the branchline were investigated: horizontal and vertical downflow. This report documents the experimental program, presents the data obtained, and discusses correlations for predicting the levels at which the onset of vapor pull-through and liquid entrainment occur and correlations for predicting the flow quality into the branchline.

NUREG/CR-4166: ANALYSIS OF FLECHT-SEASET 163-ROD BLOCKED BUNDLE DATA USING COBRA-TF. PAIK,C.Y.; HOCHREITER,L.E. Westinghouse Electric Corp. KELLY,J.M.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. October 1985. 692pp. 8511010008. EPRI NP-4111. 33288:150.

Flow blockage and spacer grid heat transfer models for rod bundle arrays have been developed for a two-phase flow situation characteristic of a PWR reflood. These models have been incorporated into COBRA-TF, which is a three-dimensional, three-field, two-fluid mechanistic two-phase flow subchannel computer code. Comparison of the predicted flow blockage heat transfer in large rod bundle arrays with test data indicates that the blockage and grid heat transfer models used with the COBRA-TF code agree quite well with the measured data. Bias plots of the predicted and measured temperatures rises from different tests indicate that, in general, the computer code calculations tend to underpredict the heat transfer improvement observed to have been caused by grids and blockage in the experiments. The principal reason for heat transfer improvement due to blockage and grids is the breakup of the entrained liquid droplets in the superheated steam flow above the quench front. The breakup of these entrained drops results in a population of much smaller drops, which are more easily evaporated in the superheated vapor. The enhanced heat transfer observed in and downstream of blockages and grids is also attribute to increased turbulence caused by the droplets in the steam flow. The resulting computer models and methods of modeling both grids and blockages, which are described in this report, are believed to be applicable to PWR safety analysis. Application of such models is expected to significantly reduce or eliminate the calculated peak clad temperature penalty due to flow blockage for a hypothetical PWR LOCA, using the Appendix K criteria.

NUREG/CR-4167: FLECHT SEASET PROGRAM.Final Report.NRC/EPRI Westinghouse Report Number 16. HOCHREITER,L.E. Westinghouse Electric Corp. November 1985. 200pp. 8512300016. EPRI NP-4112. 34092:177.

This report presents the highlights and main findings of the USNRC, EPRI, and Westinghouse cooperative FLECHT SEASET program. The report indicates areas in which results of the program can contribute to revising the current licensing requirements for Loss of Coolant (LOCA) safety analysis for PWRs. Also identified are several technical areas in which the new FLECHT SEASET data and analysis can lead to improved safety analysis modeling, and thereby to predicted PWR response for postulated accident scenarios. Significant progress has been made in the modeling areas of nonequilibrium dispersed two-phase flow during reflood. Improved models and understanding of this rod bundle cooling regime are summarized in this report. Another important result of the FLECHT SEASET program arises from the natural circulation test series, which investigated significant single-phase, two-phase, and reflux condensation cooling modes of scaled PWR undersmall-break LOCA conditions. The tests and subsequent analysis constitute one of few complete sets of data for these cooling modes in which full-height, multitube steam generators with sufficient instrumentation were used to examine primary-to-secondary heat transfer in the generators. It is believed that the natural circulation test data will be extremely useful to benchmark the improved post-TMI small-break LOCA computer codes.

NUREG/CR-4168: GT2F:A COMPUTER CODE FOR ESTIMATING LIGHT WATER REACTOR FUEL ROD FAILURES. WILLIFORD,R.E.; LANNING,D.D.; BEYER,C.E. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 277pp. 8506060567. PNL-5354. 30771:002.

This report describes the development, benchmarking and results of a computer code designed to permit comparison of BWR and PWR fuel rod failure behaviors during postulated reactor off-normal events such as control rod withdrawal errors. The code is called GT2F, and was developed from the GAPCON-THERMAL-2 code by the addition of new models for calculating transient temperatures, fission gas release, mechanical interaction between fuel and cladding, and Zircaloy cladding fracture behavior. Results indicate that for the conservatively severe overpower transient scenarios assumed, a full length commercial BWR fuel rod has a failure probability between 1% and 4.5% at 27 MWd/kgM when the transient begins from high operating power. A full length commercial PWR fuel rod has a failure probability between 2% and 11% at 28 MWd/kgM when the transient begins from low power. Failure probabilities are substantially smaller at lower burnups and for less extreme transient conditions.

NUREG/CR-4169: AN APPROACH TO TREATING RADIONUCLIDE DECAY HEATING FOR USE IN THE MELCOR CODE SYSTEM. OSTMEYER,R.M. Sandia National Laboratories. June 1985. 33pp. 8507050426. SAND84-1404. 31371:175.

A new code system is being developed for use in assessment of nuclear reactor accident risks. The code system, termed MELCOR, will treat thermal-hydraulic and fission product behavior jointly. A part of its treatment of thermal-hydraulic processes, the code system will evaluate decay heating from fission product inventories contained within the reactor core debris and compartments that are defined for the reactor system and containment. A simple approach to treating radionuclide decay heating is proposed for use in MELCOR. The proposed approach uses a table-lookup to estimate element decay powers as a function of time after reactor shutdown (start of accident). Decay power for each element in a compartment of the reactor system is found by multiplying the mass of the element in the compartment by the element's decay-heat rate per unit mass which is a function of time after reactor scram. The approach assumes that daughter products are transported along with the parent radionuclide during the accident. The validity of this assumption is discussed. In addition, methods for apportioning the decay energy between the walls and the gases in a compartment are also discussed. The proposed approach is based on SANDIA-ORIGEN calculations for a 3412 MWt PWR and a 3578 MWt BWR.

NUREG/CR-4170: AN ULTRA-HIGH SPEED RESIDUE PROCESSOR FOR SAFT INSPECTION SYSTEM IMAGE ENHANCEMENT. POLKY,J.N.; MILLER,D.D. Sigma Research, Inc. March 1985. 42pp. 8504030453. 29605:178.

The Phase-I feasibility study of residue number system (RNS) image processing for SAFT inspection has successfully determined that an advanced inspection system may be built using a correlation-reconstruction SAFT algorithm, implemented with RNS techniques and off-the-shelf electronic components. Images are reconstructed in a number theoretic transform domain with simple pointwise multiplication of the A-scan data volume by a custom point spread function (PSF), all in a highly parallel computational architecture. These methods also allow image enhancement to be easily performed for improved flaw visualization, and with negligible speed reduction. It has been determined that high resolution three dimensional flaw images may be generated and that a commercially viable product could result through development of a prototype real-time RNS processor. The hardware is expected to be made up of 100 nsec bit slice microprocessor components and large RAM storage units. Based on the performance estimates of the Phase-I effort, this new image processing system has the potential to acquire and focus the equivalent of the 145 A-scans per second, which translates into more than 1000 cubic inches per min. inspection rate for typical pressure vessel specimens.

NUREG/CR-4172: A USER'S GUIDE FOR MERGE. FREEMAN-KELLY; JUNG,R.G. Battelle Memorial Institute, Columbus Laboratories. March 1985. 41pp. 8504040006. BMI-2121. 29629:233.

The MERGE code acts as the interface between the MARCH-2 code, which is used to determine overall accident progression, and the TRAP-MELT code, which is used to evaluate reactor coolant system fission product transport and deposition. MERGE uses MARCH-calculated core exit flows and temperatures to perform a detailed gas-to-structures heat transfer analysis for the control volumes in the flow path through the reactor coolant system and converts these results into a form required as input to TRAP-MELT. MERGE can treat up to nine control volumes, containing up to five structures each. Required inputs include descriptions of the control volumes and their flow connections, as well as initial conditions.

NUREG/CR-4173: CORSOR USER'S MANUAL. KUHLMAN,M.R.; LEHMICKE,D.J. Battelle Memorial Institute, Columbus Laboratories. MEYER,R.O. NRC - No Detailed Affiliation Given. March 1985. 58pp. 8504040423. BMI-2122. 29618:312.

The CORSOR code simulates the release of fission products and structural materials from a reactor core during the in-vessel period of a severe accident in a light water reactor. The code is a simple, empirically based treatment of release and does not treat detailed mechanisms for release from high temperature fuel. The first-order release rate coefficients for the species considered are presented, the input requirements of the code are described, and an example input and output stream is supplied in an appendix.

NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984. DAEMEN,J.J.; GREER,W.B.; ADISOMA,G.S.; et al. Arizona, Univ. of, Tucson, AZ. March 1985. 384pp. 8504090456. 29739:193.

This report describes experimental borehole plugging performance assessments performed, started, or planned during June 1983 - May 1984. Results are given from field flow tests on three cement plugs installed in vertical boreholes in basalt and on one nearly horizontal cement plug. The horizontal plug and one vertical plug seal very well. Hydraulic conductivity of two vertical field plugs has been relatively high. Remedial action is described. Laboratory simulations of dynamic loading of cement plugs show no detrimental effects. Drying of cement plug, especially for months, at elevated temperatures, increases the hydraulic conductivity of the plugs severely, and reduces their bond strength along the interface. Microscopic inspection, strength and flow tests identify the drilling-induced damaged zone in basalt. While such a damaged zone exists, and has typical features (e.g. fracture density, size, location, orientation), it is thin and not likely to be a preferential flowpath. Engineering characteristics tests on bentonite plugs, chemical analysis and swelling tests. Experimental designs are given for the study of size and of thermal effects on plug performance. Preliminary results are presented. Results are included from ongoing cement push-out tests and swelling measurements.

NUREG/CR-4176: EMISSION CONTROL TECHNOLOGY AND

QUALITY ASSURANCE NEEDS AT URANIUM MILLING FACILITIES.Includes Supporting Methods For Testing,Operating,And Maintaining Air Pollution Control Devices. LUDWICK,J.D. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 55pp. 8507030684. PNL-5386. 31318:311.

Pacific Northwest Laboratory, under contract to the U.S. Nuclear Regulatory Commission, conducted an investigation of particulate emission control devices for application to process exhausts at uranium milling facilities. The scope of this investigation included devices now in use, as well as those devices that have potential application for milling sites. This report presents the results of the study. Emission control devices are categorized and described, including high-efficiency and moderate-efficiency devices as well as other (some novel) devices useful in specific situations. Preoperational considerations discussed include selecting devices, instrumentation, and testing programs. Operational and maintenance considerations related to dry and wet removal processes are described. Quality assurance documents and topics are also discussed.

NUREG/CR-4177 V01: MANAGEMENT OF SEVERE ACCIDENTS.Perspectives On Managing Severe Accidents In Commercial Nuclear Power Plants. DISALVO,R.; LEONARD,M.; MANAHAN,M.; et al. Battelle Memorial Institute, Columbus Laboratories. May 1985. 105pp. 8506130369. BMI-2123. 30868:176.

Accident management is examined from several related perspectives. The emphasis is on the role of the operating crew and the technical support provided to them before, during, and after an accident. The relationship among accident management, risk management and emergency management is examined. The roles played by industry, regulation, and research are reviewed. Finally, the results of viewing accident management from these various perspectives are reflected in the articulation of issues and some proposals for their resolution.

NUREG/CR-4177 V02: MANAGEMENT OF SEVERE ACCIDENTS.Extending Plant Operating Procedures Into The Severe Accident Regime. WREATHALL,J.; LEONARD,M.; DISALVO,R. Battelle Memorial Institute, Columbus Laboratories. May 1985. 75pp. 8506130132. BMI-2123. 30867:108.

This study examines the feasibility and value/impact of extending emergency operating procedures into the severe accident regime. It reviews the types of knowledge needed to develop such procedures and the applicability of existing regulatory review criteria. A method is developed and illustrated in two cases. This study concludes that it is feasible to develop procedures for operators to mitigate the consequences of accidents progressing past the onset of core damage. A preliminary value/impact assessment indicates a significant likelihood of there being an overall net positive benefit of developing mitigative procedures. A phased program has been proposed. First a pilot study should develop the application of the methods used in this feasibility study and provide more precise information for a detailed value-impact assessment. Based on the results of the pilot study, extension to a greater population of plants may be justified.

NUREG/CR-4178 DRFT: AN EVALUATION OF SELECTED LICENSEE EVENT REPORTS PREPARED PURSUANT TO 10 CFR 50.73.Draft Report. ANDERSON,B.S.; MILLER,C.F.; VALENTINE,B.M. EG&G Idaho, Inc. (subs. of EG&G, Inc.). March 1985. 28pp. 8512120149. EGG-2381. 33873:011. This report describes an evaluation of an industry-wide sample of Licensee Event Reports (LERs) that was performed to determine whether or not these LERs were prepared in accordance with the requirements set forth in 10 CFR 50.73. It was determined from this evaluation that many of the LERs failed to meet all of the requirements. This report presents the methodology that was used to evaluate the LERs, the conclusions reached concerning problem areas in the reports, and suggestions as to how the overall quality and completeness of reports can be improved. In addition, plant specific information is provided that will permit an assessment of each licensee's performance.

NUREG/CR-4180: STATE-OF-THE-ART OF SOLID-STATE MOTOR CONTROLLERS. JAROSS,R.A.; MULCAHEY,T.P.; KOEHL,E.R. Argonne National Laboratory. April 1985. 118pp. 8504180201. ANL-84-102. 29921:264.

The state-of-the-art of solid-state motor controllers (SSMCs) is assessed in terms of use, probability of Class 1E qualification, failure rate experience, and reliability prediction. Surveys of commercial availability, nuclear and nonnuclear electric utility experience, and architect-engineering use were made relative to the suitability of SSMCs for nuclear service. Reasons for the limited use of SSMCs in nuclear plants are given. Available failure rate data are meager, and are augmented by data on other solid-state power electronic devices that are shown to have subcomponents similar to those found in SSMCs. In addition to large nonnuclear solid-state adjustable-speed motor drives, the reliability of nuclear plant inverter systems and high-voltage solid-state DC transmission line converters is assessed. Class 1E environmental qualification experience with nuclear plant converter/inverters and battery chargers in shown to be directly applicable to SSMCs. No problems are expected in qualifying them. Actual reliability predictions of two typical commercial SSMCs are given, together with predictions of improvements possible with use of high-quality parts and manufacturing procedures.

NUREG/CR-4181: LEACHABILITY OF RADIONUCLIDES FROM CEMENT SOLIDIFIED WASTE FORMS PRODUCED AT OPERATING NUCLEAR POWER REACTORS. CRONEY,S.T. EG&G Idaho, Inc. (subs of EG&G, Inc.). April 1985. 130pp. 8505070173. EGG-2355. 30217:161.

Different sized samples of cement-solidified liquid wastes were collected from two nuclear power plants, a pressurized water reactor (PWR) and a boiling water reactor (BWR), to correlate radionuclide leaching from small and full sized waste forms. Diffusion-based model analysis of measured radionuclide leach data from small samples and full size samples indicated that leach data from small samples could be used to determine leachability indexes for full size waste form. The leachability indexes for cesium, strontium, and cobalt isotopes were determined for waste samples from both nuclear plants according to models used in ANS 16.1. The leachability indexes for the PWR samples were 6.4 for cesium, 7.1 for strontium, and 10.4 for cobalt. The leachability indexes for BWR samples were 6.5, 8.6, and 11.1 for cesium, strontium, and cobalt, respectively.

NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS. MILLER,C.A.; COSTANTINO,C.J.; PHILIPPACOPOULO; et al. Brookhaven National Laboratory. July 1985. 182pp. 8508090676. BNL-NUREG-51893. 32102:161.

Soil-structure interaction (SSI) methods currently used by industry to evaluate the seismic response of nuclear power plant facilities are reviewed with the aim of evaluating those areas of uncertainty which still exist in the analytic approaches. The primary methodologies used by various agencies generally can be grouped into three areas, namely, lumped parameter methods, finite element methods of combined soil/structure systems, and substructuring methods of analysis. Each of these are discussed in the report. In general, it was found that lumped parameter approaches yield reasonable results provided that the site is relatively uniform and the seismic inputs are low enough such that nonlinear effects are unimportant. The finite element results are reasonable provided that extreme care is taken in determining mesh geometry, boundary conditions, 3D effects, etc. Similar conclusions can be applied to the structuring approaches.

NUREG/CR-4185: AN ASSESSMENT OF DOSIMETRY DATA FOR ACCIDENTAL RADIONUCLIDE RELEASES FROM NUCLEAR REACTORS. RUNKLE,G.E.; OSTMEYER,R.H. Sandia National Laboratories. September 1985. 61pp. 8510010219. SAND85-0283. 32833:329.

This report reviews dosimetry models for estimating the absorbed dose from external and internal exposure to radionuclides. Important modeling parameters and assumptions are described. Recommendations for the dosimetry data to be used in the MELCOR health and economic consequence model are made. For estimating the dose from cloudshine and groundshine, the models for external exposure developed by Kocher are recommended. The ICRP-30 models and metabolic parameters are recommended for estimating the dose from radionuclides deposited internally via inhalation and ingestion. Dose conversion factors calculated with these models for a variety of radionuclides, clearance classes, particle sizes and integration periods were obtained from Oak Ridge National Laboratory for use in the MELCOR health and economic consequence model. Sources and magnitude of uncertainty in dose factors were evaluated. Recommendations are made for assessing the uncertainty in estimated consequences due to uncertainty in dose conversion factors.

NUREG/CR-4189: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT.Semiscale MOD-2A Feedwater-Line Break (S-SF-3) And Steam-Line Break (S-SF-5) Tests. DOBRANICH,D. Sandia National Laboratories. November 1985. 144pp. 8512270342. SAND85-0576. 34078:347.

The TRAC-PF1/MOD1 independent assessment project at Sandia is part of an overall effort funded by the NRC to determine the ability of various systems codes to predict the detailed thermal/hydraulic response of LWRs during accident and off-normal conditions. As part of this effort, calculations for Semiscale Mod-2A test S-SF-3, a feedwater-line break test, and S-SF-5, a steam-line break test, were performed with TRAC-PF1/MOD1. Most aspects of both the S-Sf-3 and S-SF-5 steady-state calculations were found to be in good agreement with data. However, the need for a better steam separator model was identified from the S-SF-3 calculation. Overall, the qualitative behavior of both transients was calculated reasonably well; however, there were some discrepancies in the prediction of the quantitative behavior. The results for the S-SF-3 transient calculation indicate that the primary-to-secondary heat transfer degradation began too early. This was possibly due to overprediction entrainment in the steam generator boiler, leading to an incorrect calculation of the secondary-side fluid distribution during the steady state. However, there was insufficient data to verify this. Results for the S-SF-5 transient calculation indicate that the primary-side fluid temperature response to a steam-line break was in reasonable agreement with data but the pressure response did not coincide with the data. Uncertainties in the data are sufficient to prevent us from determining the exact cause of this discrepancy, but there is indirect evidence that the calculated rate of phase change in the pressurizer was incorrect.

NUREG/CR-4190: CALIFORNIA OFFSHORE SURVEY OF LICENSEES USING RADIOACTIVE MATERIAL. WONG,K.S.; BROWN,J. California, State of. May 1985. 22pp. 8506060807. 30770:316.

This report is an account of offshore radioactive material activities and was prepared to provide information about their safe use in the marine environments beyond California's jurisdiction. The report supplies the essential information called for and (a) identifies licensees with radioactive nuclide utilization programs, (b) describes the licensees' work stations, (c) identifies and/or describes radionuclide, quantities and their applications, and (d) describes the radiation safety concerns and existing methods for their resolution. Finally, three offshore sites were inspected in a typical compliance manner and the findings reported. Enclosed photographs of the work stations, during source and equipment use, illustrate conditions and the licensees' operations. It is concluded from observations during onsite visits to these unusual work environments, that periodic onsite compliance inspections are necessary to assure radiation protection for all concerned.

NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES. BOLAND,J.F.; BROOKSHIRE,R.L.; DANIELSON,W.F.; et al. Argonne National Laboratory. April 1985. 225pp. 8505100194. ANL-85-13. 30268:213.

This document presents the results of a survey of Licensee control-room-habitability practices. The survey is part of a comprehensive program plan instituted in August 1983 by the NRC to respond to ongoing questions from the Advisory Committee on Reactor Safeguards (ACRS). The emphasis of this survey was to determine by field review the control-room habitability practices at three different plants, one of which is still under construction and scheduled to receive an operating license in 1986. The other two plants are currently operating, having received operating licenses in the mid-1970's and early 1980's. The major finding of this survey is that despite the fact that the latest control-room-habitability systems have become large and more complex than earlier systems surveyed, the latest systems do not appear to be functionally superior. The major recommendation of this report is to consolidate into a single NRC document, based upon a comprehensive systems engineering approach, the pertinent criteria for control-room-habitability design.

NUREG/CR-4192: THE ANALYSIS OF DRAINAGE AND CONSOLIDATION AT TYPICAL URANIUM MILL TAILINGS SITES. FAYER,M.J.; CONBERE,W. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 56pp. 8506190031. PNL-5421. 31017:324.

The computer code TRUNC was used to analyze three aspects of uranium mill tailings dewatering: the coupling of consolidation and fluid flow, drainage design, and cover load. One-dimensional simulations of the effects of consolidation on fluid flow within a tailings pile of either slimes or a sand/slimes mix showed that drainage flux was greater for a consolidating system early in the simulation. After days 1,400 and 160 of the simulations for the slimes and sand/slimes mix, respectively, however, the fluxes for the nonconsolidating systems were greater. In the sand/slimes mix, the nonconsolidating system had a cumulative flux by day 5,000 that was 93% of that of the consolidating system. At the same time, in the slimes tailings piles the nonconsolidating system had a cumulitive flux of only 34% of that of the consolidating system, which indicates that consolidation and fluid flow should not be decoupled for the slimes. Two-dimensional simulations of an actual tailings pile drainage design showed that a sand blanket drain increased the rate of drainage and settlement. The sand blanket drain also significantly reduced differential settlement across the pile. This indicates that the use of a sand blanket drain could enable earlier placement of the cover system after tailings emplacement. In simulations of covered and uncovered tailings piles, nearly the same quantity of water was removed from each, but drainage occurred much more slowly without the cover; hence, surface settlement was slower when the tailings pile was not covered.

NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984. MCCRAY,J.G.; NOWATZKI,E.A.; ARMSTRONG,G.; et al. Arizona, Univ. of, Tucson, AZ. May 1985. 219pp. 8506240003. 31149:012.

This is the final report on a three year study to evaluate trench cap designs, trench construction and trench loading by accelerating the creation of void space to simulate waste degradation in order to apply stress conditions on the trench in a relative short period of time. Eight trenches were initially constructed and instrumented, four in a semi-arid region and four in a more humid mountainous region. After the first year the semi-arid site was abandoned due to cap failures. A new trench incorporating an improved soil slab design with a wick was constructed at the humid site. Conclusions from these experiments are: 1. Controlled compaction is not sufficient to mitigate long term surface subsidence. 2. Single sheet geotextile reinforcement is not adequate trench cap reinforcement. 3. Geotextile wrapped soil slab attenuates surface subsidence and surface water infiltration. 4. A steel-reinforced soil-cement slab appears to meet the requirements necessary for long term stability. 5. If the crown and cap remain stable so does the trench. 6. Aliphatic tracers performed well and dye type of tracers poorly. 7. Tracers are feasible and effective as a trench monitoring tool. 8. Narrow designed trenches improve trench cap stability. This report recommends a design for enhanced isolation disposal trench providing improved monitoring capabilities.

NUREG/CR-4195: OVERVIEW OF TRAC-PD2 ASSESSMENT CALCULATIONS. WATERMAN,M.E. EG&G Idaho, Inc. (subs. of EG&G, Inc.). November 1985. 70pp. 8512270237. EGG-2380. 34081:311.

A summary of Transient Reactor Analysis Code Version PD2(TRAC-PD2) calculations performed at the Idaho National Engineering Laboratory (INEL) is presented in this report as part of the U.S. Nuclear Regulatory Commission's (NRCs) overall assessment program of TRAC-PD2. The calculated and measured parameters summarized in this report are break mass flow rate, primary coolant system pressure, reactor core flow rates, and fuel rod cladding temperatures. The data were obtained from seven tests that were performed at two test facilities. The tests were conducted to study the various aspects of cold leg break transients, including the effects of large and small breaks, and core reflood phenomena. User experience gained from the various calculations is also summarized.

NUREG/CR-4196: OVERVIEW OF TRAC-BD1 (VERSION 12) ASSESSMENT STUDIES. KULLBERG,C.M. EG&G Idaho, Inc. (subs. of EG&G, Inc.). April 1985. 55pp. 8506060796. EGG-2382. 30771:279.

A series of simulations were performed at Idaho National Engineering Laboratory to continue the advancement of Boiling Water Reactor (BWR) safety research, with the TRAC-BD1 (Version 12) computer code. The principal motivation for performing these simulations was to assess the code's capability to calculate Loss-of-Coolant Accident (LOCA) related phenomena. The results of a number of TRAC-BD1 (Version 12) simulations, which cover a broad range of conditions during different types of LOCA scenarios, are summarized in this document. Selected comparisons between calculated and measured results are presented. Conclusions derived from those comparisons are given.

NUREG/CR-4197: SAFETY GOAL SENSITIVITY STUDIES. BURKE,R.P.; BLOND,R.M. Sandia National Laboratories. June 1985. 50pp. 8507020415. SAND85-0634. 31313:301.

This study presents the results of analyses performed as part of the two-year evaluation program for the NRC safety goals. Analyses are performed to demonstrate the sensitivities of the quantitative design objective calculations to changes in input parameters and assumptions. Results are presented which show the influence of parameter changes on the health risk quantitative design objectives and on cost-benefit calculations. The alternative design objective risk measures are compared with alternative measures of the health impacts of LWR accidents. The results of this study provide background information and input to be used in the NRC staff evaluation of the safety goals and quantitative design objectives.

NUREG/CR-4198: FRACTURE IN GLASS/HIGH LEVEL WASTE CANISTERS. MARTIN,D.M. Iowa State Univ., Ames, IA. April 1985. 81pp. 8504170534. 29906:243.

The release rate of radionuclides from a vitrified waste form due to aqueous leaching by ground water will depend, among other factors, on the waste form's surface area. Large castings of glass will almost certainly be used as the waste form for high level nuclear wastes and such castings tend to fracture as a result of transient and residual stresses induced by the casting process; such fractures increase the surface area available for aqueous leaching of radionuclides from the HLW glass. The primary focus of this study was on achieving an understanding of the dependence of fracture surface area on glass properties and processing variables for both in-can melts and castings. The maximum fracture surface area per unit volume of glass observed in this study was about 7.1/cm (an equivalent spherical particle diameter of 0.85 cm) for a water quenched in-can melt. The processing parameter which appears to most strongly affect the extent of fracture surface area for both castings and in-can melts is the dimensionless Biot modulus (thermal film coefficient x radius/waste form thermal conductivity).

NUREG/CR-4199: A DEMONSTRATION UNCERTAINTY/SENSITIVITY ANALYSIS USING THE HEALTH AND ECONOMIC CONSEQUENCE MODEL CRAC2. ALPERT,D.J.; IMAN,R.L.; HELTON,J.C.; et al. Sandia National Laboratories. June 1985. 59pp. 8507050415. SAND84-1824. 31372:210.

A demonstration uncertain/sensitivity analysis was performed for the reactor accident consequence model CRAC2 using techniques compiled as part of the NRC-sponsored MELCOR program. The principal objectives of the study were: 1) to demonstrate the use of the uncertain/sensitivity analysis techniques, 2) to test the computer models that implement the techniques 3) to identify possible difficulties in performing such an analysis, and 4) to explore alternative means of analyzing, displaying, and describing the results. Seventeen CRAC2 input variables thought to contribute significantly to uncertainty in estimated consequences were selected for analysis; subjective estimates of ranges, distributions, and correlations for these variables were made. Latin hypercube sampling, a modified Monte Carlo technique, was used to generate two multivariate samples of size 50 from the distributions assigned to the 17 input variables. A total of 100 CRAC2 runs, 50 for each sample, was performed. The results of the two samples were similar. A regression analysis was performed to estimate the contribution of each variable to uncertainty in estimated consequences. The study was first performed with the magnitude of the source term as one of the 17 variables. A second analysis was performed with a fixed source term. Only one sample of size 50 was generated in the second analysis. The uncertainty/sensitivity analysis techniques compiled for MELCOR appear well suited for use with a health and economic consequence model. Alternative methods for displaying and describing the results are presented. The insights gained from performing the analysis are reviewed and major conclusions summarized. A comparison of the results of this study with current point estimates of health and economic consequences is presented.

NUREG/CR-4200: BIODEGRADATION TESTING OF SOLIDIFIED LOW-LEVEL WASTE STREAMS. PICIULO,P.L.; SHEA,C.E.; BARLETTA,R.E. Brookhaven National Laboratory. May 1985. 46pp. 8506140593. BNL-NUREG-51868. 30936:219.

The NRC Technical Position on Waste Form (TP) specifies that waste should be resistant to biodegradation. The methods recommended in the TP for testing resistance to fungi, ASTM G21, and for testing resistance to bacteria, ASTM G22, were carried out on several types of solidified simulated wastes, and the effect of microbial activity on the mechanical strength of the materials tested was examined. The tests are believed to be sufficient for distinguishing between materials that are susceptible to biodegradation and those that are not. However, it is concluded that failure of these tests should not be regarded of itself as an indication that the waste form will biodegrade to an extent that the form does not meet the stability requirements of 10 CFR Part 61. In the case of failure of ASTM G21 or ASTM G22 or both, it is recommended that additional data be supplied by the waste generator to demonstrate the resistance of the waste form to microbial degradation.

NUREG/CR-4201: THERMAL STABILITY TESTING OF LOW-LEVEL WASTE FORMS. PICIULO,P.L.; CHAN,S.F. Brookhaven National Laboratory. May 1985. 48pp. 8506060814. BNL-NUREG-51869. 30769:247.

The NRC Technical Position (TP) on Waste Form specifies that waste forms should be resistant to thermal degradation. The thermal cycle testing procedure outlined in the TP on Waste Form was carried out and is believed adequate for demonstrating the thermal stability of solidified waste forms. The inclusion of control samples and the monitoring of sample temperature are recommended additions to the test. An outline for reporting thermal cycling test results is given. To produce a data base on the applicability of the thermal cycling test, the following simulated laboratory-scale waste forms were prepared and tested: boric acid and sodium sulfate evaporator bottoms, mixed bed bead resins, and powdered resins each solidified in asphalt, cement and vinyl ester-styrene.

NUREG/CR-4203: A CALCULATIONAL METHOD FOR DETERMINING BIOLOGICAL DOSE RATES FROM IRRADIATED RESEARCH REACTOR FUEL. SCHNITZLER,B.G. EG&G Idaho, Inc. (subs. of EG&G, Inc.). April 1985. 65pp. 8506060811. EGG-2383. 30775:237.

This report describes a calculational method for the determination of biological dose rate from irradiated research reactor fuels. The calculational method is implemented in a computer program for quick and convenient assessment of multigroup gamma and beta dose rates resulting from an arbitrary (user-supplied) irradiation history. The FUELDR program calculates dose rates at a fixed dose point using built-in fission product impulse source functions and precalculated gamma and beta transport factors. The fixed dose point is located on the axial mid-plane at a distance of 91.44 cm (3 ft) from the fuel element. Transport factors are included for sixteen unique (235)U fuel types in use at thirteen nonpower reactor facilities.

NUREG/CR-4204: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS:Annual Report,October 1983 - September 1984. CHOPRA,O.K.; CHUNG,H.M. Argonne National Laboratory. April 1985. 33pp. 8506140599. ANL-85-20. 30935:223.

This progress report summarizes work performed by Argonne National Laboratory during the twelve months from October 1983 to September 1984 on long-term embrittlement of cast duplex stainless steels used in light-water reactors.

NUREG/CR-4205: TRAP-MELT2 USER'S MANUAL. JORDAN,H.; KUHLMAN,M.R. Battelle Memorial Institute, Columbus Laboratories. May 1985. 74pp. 8506190036. BMI-2124. 31017:100.

The TRAP-MELT 2 code is a development of the previously issued TRAP-MELT code which simulates the transport and deposition of aerosol particles and certain vapors in the reactor coolant system under hypothetical accident conditions in a light water reactor. This manual contains a brief description of the models of the processes treated in the code and of the code organization. The input to the code for a sample run are presented and output from a run are presented as well.

NUREG/CR-4206: A SELECT REVIEW OF THE RECENT (1979-1983) BEHAVIORAL RESEARCH LITERATURE ON TRAINING SIMULATORS. LAUGHERY,K.R. Oak Ridge National Laboratory. May 1985. 51pp. 8506130489. ORNL/TM-9445. 30901:249.

Report summarizes some selected reports of behavioral research performed in years 1979-1983 on training simulator application technology, and discusses findings related to nuclear power plant operators' simulator training. Findings are organized as related to the design, testing, and use of training simulators. Topics include Simulator Fidelity vs. Training Effectiveness, Operator Performance Measurement, Measuring Simulator Effectiveness, and Simulator Utilization Practices. Reviews 89 references.

NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS:Application To Establishing Values Of f1 For Soluble Plutonium. BHATTACHARYYA; LARSEN,R.P.; OLDHAM,R.D.; et al. Argonne National Laboratory. May 1985. 99pp. 8507050425. ANL-85-21. 31371:207.

The gastrointestinal (GI) absorption of plutonium was measured in mice, rats, and dogs under conditions relevant to setting drinking water standards. The fractional GI absorption of Pu (VI) in adult mice was 2 x 10(-4) (0.02%) in fed mice and 2 x 10(-3) (0.02%) in fasted mice. The GI absorption of plutonium was independent of plutonium oxidation state, administration medium, and plutonium concentration; absorption was dependent upon animal species, state of animal fasting, state of Pu(IV) hydrolysis, and age of the animal. Fractional GI absorption values ranged from 3 x 10(-5) (0.003%) for hydrolyzed Pu(IV) administered to fed adult mice to 7 x 10(-3) (0.7%) for Pu(VI) administered to fed neonatal rats. From analysis of our data, we suggested values of f(1) (the fraction transferred from gut to blood in humans) for use in establishment of oral limits of exposure to plutonium. For an acute exposure in the occupational setting, we proposed one value of f(1) for fed (2 x10(-4) and one for fasted (2 x 10(-3) individuals. For the environmental setting, we developed two approaches to obtaining values of f(1); suggested values were 6 x 10(-4) and 4 x 10(-3), respectively. Both approaches took into account effects of animal age and fasting. We discussed uncertainties in proposed values of f(1) and made recommendations for further research.

NUREG/CR-4209: COMPARISON OF ANALYTICAL PREDICTIONS AND EXPERIMENTAL RESULTS FOR A 1:8-SCALE STEEL CONTAINMENT MODEL PRESSURIZED TO FAILURE. CLAUSS,D.B. Sandia National Laboratories. September 1985. 68pp. 8512060008. SAND85-0679. 33789:045.

Predictions for the response of a 1:8-scale model of a steel nuclear containment building to overpressurization are compared to experimental results. Finite element analyses were used to predict the model's response. Strains, displacements, and leak rate measurments were made at 21 different pressure levels. Comparisons of the pressure histories for strain and displacement at a point, and the spatial variation of strain and displacement are made. In addition, comparisons of a more global nature, such as the capacity of the model and the failure mode, are discussed. An evaluation of the predictive capabilities and the failure criteria is made in the light of these comaprisons.

NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS. BAYBUTT,P.; RAGHURAM,S.; AVCI,H.I. Battelle Memorial Institute, Columbus Laboratories. April 1985. 62pp. 8505080375. BMI-2125. 30218:271.

A new computer code called MATADOR (Methods for the Analysis of Transport And Deposition Of Radionuclides) has been developed to replace the CORRAL computer code which was written for the Reactor Safety Study (WASH-1400). This report contains a detailed description of the models used in MATADOR. A companion report provides a User's Manual for the code. MATADOR is intended for use in system risk studies to analyze radionuclide transport and deposition in reactor containments. The principal output of the code is information on the timing and magnitude of radionuclide releases to the environment as a result of severely degraded core accidents. MATADOR considers the transport of radionuclides through the containment and their removal by natural deposition and the operation of engineered safety systems such as sprays. The code requires input data on the source term from the primary system, the geometry of the containment, and the thermal-hydraulic conditions in the containment.

NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL. AVCI,H.I.; RAGHURAM,S.; BAYBUTT,P. Battelle Memorial Institute, Columbus Laboratories. April 1985. 75pp. 8505080373. BMI-2126. 30218:192.

A new computer code called MATADOR (Methods for the Analysis of Transport And Deposition of Radionuclides) has been developed to replace the CORRAL-2 computer code which was written for the Reactor Safety Study (WASH-1400). This report is a User's Manual for MATADOR. A companion report describes in detail the models used in the code. MATADOR is intended for use in system risk studies to analyze radionuclide transport and deposition in reactor containments. The principal output of the code is information on the timing and magnitude of radionuclide releases to the environment as a result of severely degraded core accidents. MATADOR considers the transport of radionuclides through the containment and their removal by natural deposition and by engineered safety systems such as sprays. It is capable of analyzing the behavior of radionuclides existing either as vapors or aerosols in the containment. The code requires input data on the source terms into the containment, the geometry of the containment, and thermal-hydraulic conditions in the containment.

NUREG/CR-4212: IN-PLACE THERMAL ANNEALING OF NUCLEAR REACTOR PRESSURE VESSELS. SERVER,W.L. EG&G Idaho, Inc. (subs. of EG&G, Inc.). April 1985. 250pp. 8505070548. EGG-MS-6708. 30211:101.

Radiation embrittlement of ferritic pressure vessel steels changes the toughness properties. A thermal anneal cycle well above the normal operating temperature of the vessel can restore most of the original properties. The Army SM-1A test reactor vessel was wet annealed in 1967, and wet annealing of the Belgian BR-3 reactor vessel has recently taken place. An industry survey indicates that dry annealing of a reactor vessel in-place is feasible, but solvable engineering problems exist. Limited toughness data available for five high copper content welds were reviewed. The review suggested that significant recovery results from annealing at 454 degrees centigrade (850 degrees fahrenheit) for one week, but scatter in the data makes assessment of recovery and reembrittlement response difficult to quantify. A thermal and structural analysis of a reactor vessel undergoing an annealing treatment found no problems with the reactor vessel itself, but did indicate a rotation at the nozzle region of the vessel which would plastically deform the attached primary piping. Further analytical studies attempted to solve this problem, but they were not successful. An American Society for Testing and Materials (ASTM) task group is upgrading and revising guide ASTM E 509-74 with emphasis on the materials and surveillance aspects of annealing.

NUREG/CR-4213: SETS REFERENCE MANUAL. WORRELL,R.B. Sandia National Laboratories. July 1985. 250pp. 8508090642. SAND83-2675. 32097:137.

The Set Equation Transformation System (SETS) is used to achieve the symbolic manipulation of Boolean equations. Symbolic manipulation involves changing equations from their original forms into more useful forms -- particularly by applying Boolean identities. The SETS program is an interpreter which reads, interprets, and executes SETS user programs. The user writes a SETS user program specifying the processing to be achieved and submits it, along with the required data, for execution by SETS. Because of the general nature of SETS, ie., the capability to manipulate Boolean equations regardless of their origin, the program has been used for many different kinds of analysis.

NUREG/CR-4214: HEALTH EFFECTS MODEL FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Part I:Introduction,Integration & Summary.Part II:Scientific Basis For Health Effects Models. EVANS,J.S.; MOELLER,D.W.; COOPER,D.W.; et al. Harvard Univ., Cambridge, MA. August 1985. 357pp. 8509190140. SAND85-7185. 32681:109.

Analysis of the radiological health effects of nuclear power plant accidents requires models for predicting early health effects, cancers and benign thyroid nodules, and genetic effects. Since the publication of the Reactor Safety Study, additional information on radiological health effects has become available. This report summarizes the effort of a program designed to provide revised health effects models for nuclear power plant accident consequence modelling. The new models for early effects address four causes of mortality and nine categories of morbidity. The models for early effects are based upon two parameter Weibell functions. They permit evaluation of the influence of dose protraction and address the issues of variation in radiosensitivity among the population. The piecewise-linear dose-response models used in the Reactor Safety Study to predict cancers and thyroid nodules have been replaced by linear and linear-quadratic models. The new models reflect the most recently reported results of the follow-up of the survivors of the bombings at Hiroshima and Nagasaki and permit analysis of both morbidity and mortality. The new models for genetic effects allow prediction of genetic risks in each of the first five generations after an accident and include information on the relative severity of various classes of genetic effects. The uncertainty in modelling radiological health risks is addressed by providing central, upper and lower estimates of risks. An approach is outlined for summarizing the health consequences of nuclear power plant accidents.

NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA. MACKENZIE,D.R.; VASLOW,F.; DOUGHERTY,D.R.; et al. Brookhaven National Laboratory. May 1985. 77pp. 8506140405. BNL-NUREG-51873. 30908:116.

This report provides technical support to NRC in connection with the regulation 10 CFR Part 61 and NRC's Technical Position (TP) on waste form. Six specific areas are addressed, namely: the technical basis for limiting containers of radioactive gases to atmospheric pressure and 100 curies; the requirements to demonstrate that a stable waste would be recognizable for 300 or 500 years; the feasibility of achieving less than 5% deformation in buried wastes; the adequacy of ASTM tests G21 and G22 for testing for biodegradability; the adequacy of ASTM test B553 for testing for thermal degradation; and the basis for determining if a waste is explosive or pyrophoric. The principal conclusions of the report follow. A maximum pressure of 1.5 atmospheres for radioactive gases is acceptable, but the radioactivity limit should depend on the isotope, the quality of the container and the properties of the site. Site and package qualities and a wet/dry cycling test are suggested that appreciably increase the probability of indicating whether a waste would have long-term recognizability. Achieving deformation of buried waste of <5% would not be feasible using current solidification methods with either metal or polyethylene containers. ASTM tests G21 and G22, with modifications are suitable for biodegradability testing. A modified form of ASTM B553 is adequate for thermal testing. Required information on pyrophoric and explosive materials is provided.

NUREG/CR-4217: A STATISTICAL ANALYSIS OF NUCLEAR POWER PLANT VALVE FAILURE-RATE VARIABILITY--SOME PRELIMINARY RESULTS. BECKMAN,R.J.; MARTZ,H.F. Los Alamos Scientific Laboratory. July 1985. 70pp. 8508090719. LA-10396-MS. 32105:076.

Valve failure data from the In-Plant Reliability Data System (IPRDS) are statistically analyzed using the Failure Rate Analysis Code (FRAC). Data from the five failure modes, four of which are time related and the other demand related, are analyzed to determine which of the factors--operating system, valve size, valve type, operating type, and operating mode--most affect valve failure rates. A separate analysis is given for each of two plants, a pressurized water reactor (PWR) and a boiling water reactor (BWR). For both plants and each failure mode, multiplicative adjustments for the mean are obtained for categories, such as nuclear or containment systems, of the various factors. These multipliers indicate whether a particular category of a factor has a corresponding failure rate that is less than the average failure rate (a multiplier less than one) or greater than average (a multiplier greater than one). Based on the multiplicative adjustments, ball valves are shown to be the most reliable valves for the PWR plant. Globe and gate valves have the highest failure rates for this plant. The average failure rate for the BWR plant is found to be half that of the PWR plant for three of the five failure modes studied. In addition to the multipliers, point estimates and confidence intervals on the failure rates are given for selected valve factor combinations. These estimates and intervals are compared with several other estimates.

NUREG/CR-42.8: LOCA SIMULATION IN THE NATIONAL RESEARCH UNIVERSAL REACTOR PROGRAM.Postirradiation Examination Results For The Third Materials Test (MT-3) - Second Campaign. HABERMAN,J.H. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 62pp. 8506260395. PNL-5433. 31244:326.

A series of in-reactor experiments were conducted using full-length 32-rod pressurized water reactor (PWR) fuel bundles as part of the Loss-of-Coolant Accident (LOCA) Simulation Program by Pacific Northwest Laboratory (PNL). The third materials test (MT-3) was the sixth experiment in a series of thermal-hydraulic and materials deformation/rupture experiments conducted in the National Research Universal (NRU) Reactor, Chalk River, Ontario, Canada. The MT-3 experiment was jointly funded by the U.S. Nuclear Regulatory Commission (NRC) and the United Kingdom Atomic Energy Authority (UKAEA) with the main objective of evaluation ballooning and rupture during active two-phase cooling at elevated tempertures. All 12 test rods in the center of the 32-rod bundle failed with an average peak strain of 55.4%. At the request of the UKAEA, a destructive postirradiation examination (PIE) was performed on 7 of the 12 test rods. The results of this examination were presented in a previous report. Subsequently, and at the request of UKAEA, PIE was performed on three additional rods along with further examination of one of the previously examined rods. Information obtained from the PIE included cladding thickness measurements, cladding metallography, and particle size analysis of the fractured fuel pellets. This report describes the additional PIE work performed and presents the results of the examinations.

NUREG/CR-4219 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985. PUGH,C.E. Oak Ridge National Laboratory. July 1985. 216pp. 8507250168. ORNL/TM-9593/V1. 31783:146.

The Heavy-Section Steel Technology (HSST) 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 cracklike flaws. Current work is organized into ten tasks: (1) program management, (2) fracture-methodology and analysis, (3) material characterization and properties, (4) environmentally assisted crack growth studies, (5) crack arrest technology, (6) irradiation effects studies, (7) cladding evaluations, (8) intermediate vessel tests and analysis, (9) thermal-shock technology, and (10) pressurized thermal-shock technology.

NUREG/CR-4220: RELIABILITY ANALYSIS OF CONTAINMENT ISOLATION SYSTEMS. PELTO,P.J.; GALLUCCI,R.H.; AMES,K.R. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 222pp. 8506260362. PNL-5432. 31245:199.

This report summarizes the results of the Reliability Analysis of Containment Isolation System Project. Work was performed in five basic areas: design review, operating experience review, related research review, generic analysis and plant specific analysis. Licensee Event Reports (LERs) and Integrated Leak Rate (ILRT) Test reports provided the major sources of containment performance information used in this study. Data extracted from the LERs were assembled into a computer data base. Qualitative and quantitative information developed for containment performance under normal operating conditions and design basis accidents indicate that there is room for improvement. A crude estimate for overall containment unavailability for relatively small leaks which violate plant technical specifications is 0.3. An estimate of containment unavailability due to large leakage events is in the range of 0.001 to 0.01. These estimates are dependent on several assumptions (particularly on event duration times) which are documented in the report.

NUREG/CR-4221: AN EVALUATION OF STRESS CORROSION CRACK GROWTH IN BWR PIPING SYSTEMS. KASSIR,M.; SHARMA,S.; REICH,M.; et al. Brookhaven National Laboratory. May 1985. 80pp. 8506130175. BNL-NUREG-51874. 30867:183.

This report presents the results of a study conducted to evaluate the effects of stress intensity factor and environment on the growth behavior of integranular stress corrosion cracks in type 304 stainless steel piping systems. Most of the detected cracks are known to be circumferential in shape, and initially start at the inside surface in the heat affected zone near girth welds. These cracks grow both radially in-depth and circumferentially in length and, in extreme cases, may cause leakage in the installation. The propagation of the crack is essentially due to the influence of the following simultaneous factors: (1) The action of applied and residual stress, (2) Sensitization of the base metal in the affected zone adjacent to firth weld and (3) The continuous exposure of the material to an aggressive environment of high temperature water containing dissolved oxygen and some levels of impurities. Each of these factors and their effects on the piping systems is discussed in detail in text of the report. The report also evaluates the time required for hypothetical cracks in BWR pipes to propagate to their critical size. The pertinent times are computed and displayed graphically. Finally, parametric study is performed in order to assess the relative influence and sensitivity of the various input parameters (residual stress, crack growth law, diameter of pipe, initial size of defect, etc.) which have bearing on the growth behavior of the intergranular stress corrosion cracks in type 304 stainless steel. Cracks in large-diameter as well as in small-diameter pipes are considered and analyzed.

NUREG/CR-4225: SUMMARY OF EFFICIENCY TESTING OF STANDARD AND HIGH-CAPACITY HIGH-EFFICIENCY PARTICULATE AIR FILTERS SUBJECTED TO SIMULATED TORNADO DEPRESSURIZATION AND EXPLOSIVE SHOCK WAVES. SMITH,P.R.; GREGORY,W.S. Los Alamos Scientific Laboratory. April 1985. 25pp. 8507020407. LA-10401-MS. 31309:007.

Pressure transients in nuclear facility air cleaning systems can originate from natural phenomena such as tornadoes or from accident-induced explosive blast waves. This study was concerned with the effective efficiency of high-efficiency particulate air (HEPA) filters during pressure surges resulting from simulated tornado and explosion transients. The primary objective of the study was to examine filter efficiencies at pressure levels below the point of structural failure. Both standard and high-capacity 0.61-m by 0.61-m HEPA filters were evaluated, as were several 0.2-m by 0.2-m HEPA filters. For a particular manufacturer, the material release when subjected to tornado transients is the same (per unit area) for both the 0.2-m by 0.2-m and the 0.61-m by 0.61m filters. For tornado transients, the material was on the order of micrograms per square meter. When subjecting clean HEPA filters to simulated tornado transients with aerosol entrained in the pressure pulse, all filters tested showed a degradation of filter efficiency. For explosive transients, the material release from preloaded high-capacity filters was as much as 340 g. When preloaded high-capacity filters were subjected to shock waves approximately 50% of the structural limit level, 1 to 2 mg of particulate was released.

NUREG/CR-4226: NEW MADRID SEISMOTECTONIC STUDY.Activities During Fiscal Year 1983. BUSCHBACH,T.C. St. Louis Univ., St. Louis, MO. April 1985. 153pp. 8505070552. 30209:297.

The purpose of the New Madrid Seismotectonic Study is to identify the earthquake mechanisms within a 200-mile radius of New Madrid, Missouri. During 1983 there was more awareness of the significance of current regional stress patterns and the local concentration of stresses by basement structures and inhomogeneities. The program continued to concentrate on defining boundaries of a proposed rift complex in the area, as well as establishing the relationships of the east-west trending fault systems with the northeast-trending faults of the Wabash Valley and New Madrid areas. There were 204 earthquakes located by the Saint Louis University microearthquake network in 1983. In addition, the earthquake swarm in north-central Arkansas continued throughout the year, and 45,000 earthquakes have been recorded there since January, 1982. Trenching data from Late Cenozoic terrace deposits along the Kentucky River Fault System suggest that there was post-terrace deformation along some of the faults. Thermal and chemical data from groundwaters in the Mississippi Embayment appear to be useful in localizing deep faults that cut through the aquifers. Early indications from studies of jointing in Indiana are that the direction of major joint sets will be useful in determining regional stress directions. No Quaternary faulting was found in the Indiana or Illinois fault studies.

NUREG/CR-4227: HUMAN ENGINEERING GUIDELINES FOR THE EVALUATION AND ASSESSMENT OF VIDEO DISPLAY UNITS. GILMORE,W.E. EG&G Idaho, Inc. (subs. of EG&G, Inc.). July 1985. 535pp. 8508150085. EGG-2388. 32193:075.

This report provides the Nuclear Regulatory Commission with a single source that documents known guidelines for conducting formal Human Factors evaluations of Visual Display Units (VDUs). The handbook is a "cookbook" of acceptance guidelines for the reviewer faced with the task of evaluating VDUs already designed or planned for service in the control room. The areas addressed are visual displays, controls, control/display integration, and workplace layout. Guidelines relevant to each of those areas are presented. The existence of supporting research is also indicated for each guideline. A comment section and Method for Assessment section are provided for each set of guidelines.

NUREG/CR-4228: REVIEW OF THE VOGTLE UNITS 1 AND 2 AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS. FRESCO,A.; YOUNGBLOOD,R.; PAPAZOGLOU,I.A. Brookhaven National Laboratory. October 1985. 95pp. 8511120038. BNL-NUREG-51876. 33440:228.

This report presents the results of the review of the Auxiliary Feedwater System reliability analysis for the Vogtle Electric Generating Plant (VEGP) Units 1 and 2. The objective of this report is to estimate the probability that the Auxiliary Feedwater System will fail to perform its mission for each of three different initiators: (1) loss of main feedwater with offsite power available, (2) loss of offsite power, (3) loss of all ac power except vital instrumentation and control 125-V dc/120-V ac power. The scope, methodology, and failure data are prescribed by NUREG-0611, Appendix III. The results are compared with those obtained in NUREG-0611 for other Westinghouse plants.

NUREG/CR-4229: EVALUATION OF CURRENT METHODOLOGY EMPLOYED IN PROBABILISTIC RISK ASSESSMENT (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS. RUGER,C.; BOCCIO,J.L.; AZARM,M.A. Brookhaven National Laboratory. May 1985. 47pp. 8506190103. BNL-NUREG-51877. 31017:277.

The report presents a general evaluation of the current methodology used by industry for the probabilistic assessment of fire events in nuclear power plants. The basis for this evaluation, in which the strengths and weaknesses of the methods are identified, stem from reviews of several, industry-sponsored, full-scope Probabilistic Risk Assessments (PRAs) and various deterministic/probabilistic approaches used by industry to judge their compliance with or used to seek exemptions from the fire-protection requirements enumerated in Appendix R to 10 CFR 50. In performing this evaluation of the current methodologies, state-of-the-art literature on the modeling of propagation/detection/suppression, input parameters, and modeling uncertainties are utilized. Areas are identified where recently-developed, more accurate and complete techniques can be implemented to reduce the state-of-knowledge uncertainties that presently exist. Recommendations are also made which could be the basis for a more suitable and complete fire-risk methodology.

NUREG/CR-4230: PROBABILITY-BASED EVALUATION OF SELECTED FIRE PROTECTION FEATURES IN NUCLEAR POWER PLANTS. AZARM,M.A.; BOCCIO,J.L. Brookhaven National Laboratory. May 1985. 93pp. 8506180415. BNL-NUREG-51878. 30985:281.

A probabilistic approach for the evaluation of major fire protection measures in nuclear power plants is described. The methods developed are applied to two representative fire areas -- one similar to a cable routing room and the other typical of a diesel generator room. The fire areas chosen for application, the fire scenarios described, and the various fire-damage states specified in the two illustrative examples are used to evaluate those fire-protection guidelines which deal with automatic/manual fire detection and suppression systems, rated barriers, divisional separation, drainage systems, dampers, and fire rating of electrical cables. Tabular results are presented, which reflect the relative merits of these systems/features in terms of conditional probabilities of achieving various room-damage states. The conclusions drawn and the lessons learned through the course of this study are discussed, and the areas that may need further investigation are identified.

NUREG/CR-4231: EVALUATION OF AVAILABLE DATA FOR PROBABILISTIC RISK ASSESSMENTS (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS. SAMANTA,P.K.; BOCCIO,J.L. Brookhaven National Laboratory. KRASNER,L.M.; et al. Factory Mutual Research Corp. May 1985. 71pp. 8506190077. BNL-NUREG-51879. 31017:205.

Several crucial parameters are needed in the assessment of fire risk in nuclear power plants. Among those that need to be developed from a data base are: (1) fire frequency, (2) fire detection time, and (3) fire suppression time. Currently, that data base for nuclear power plants is not large enough to develop these parameters, considering fuel location, fuel geometry, combustion properties, enclosure geometry, etc. This study attempts to augment the nuclear data base by investigating the usefulness of other nonnuclear data bases which contain fire incident loss experience of occupancy classes having somewhat similar physical features and fire protection engineering systems normally found in nuclear power plants. This study has found that indeed some useful information can be gleaned from nonnuclear sources; in particular, detection and suppression times. However, other fire-risk data needs such as fire frequency and fire size would require other forms of data searches and data analyses that at this stage can only be conceptualized.

NUREG/CR-4232: THE RESPONSE OF VENTILATION DAMPERS TO LARGE AIRFLOW PULSES. GREGORY,W.S.; SMITH,P.R. Los Alamos Scientific Laboratory. July 1985. 71pp. 8507250121. LA-10413-MS. 31794:051.

The results of an experimental program to evaluate the response of ventilation system dampers to simulated tornado transients are reported. Relevant data, such as damper response time, flow rate and pressure drop, and flow/pressure vs blade angle, were obtained, and the response of one tornado protective damper to simulated tornado transients was evaluated. Empirical relationships that will allow the data to be integrated into flow dynamics codes were developed. These flow dynamics codes can be used by safety analysts to predict the response of nuclear facility ventilation systems to tornado depressurizations.

NUREG/CR-4233: DISTRIBUTION OF CORBICULA FLUMINEA AT NUCLEAR FACILITIES. COUNTS,C.L. Delaware, Univ. of, Lewes, DE. November 1985. 86pp. 8512270228. 34084:171.

A review of the zoogeographic records for the exotic Asian clam, CORBICULA FLUMINEA (Muller, 1774), reveals its presence in 27 states where nuclear powered electric generating plants are either operating or under construction. Nineteen plant sites reported infestations of varying severity in facilities or source waterbodies immediately adjacent to the facility by populations of C. FLUMINEA. Thirteen plant sites are located within the zoogeographic limits of C. FLUMINEA but have a low risk of infestation due to either salt water cooling systems or locations a great distance from known populations. Eighteen plant sites are located wholly outside of the known zoogeographic range of C. FLUMINEA. Thirty plant sites are located in close proximity to known populations of C. FLUMINEA and therefore should maintain surveillance of the source water body and within plant water systems for possible infestations by these bivalves.

NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS. GREENSTREET,W.; MURPHY,G.A.; EISSENBERG,D.M. Oak Ridge National Laboratory. July 1985. 124pp. 8507250149. ORNL-6170/V1. 31808:069.

This is the first in a series of three reports on electric motor-operated valves (MOVs) to be produced under the U.S. Nuclear Regulatory Commission's Nuclear Plant Aging Research program. This program addresses the evaluation and identification of practical and cost-effective methods for detecting, monitoring, and assessing the severity of time-dependent degradation (aging and service wear) of MOVs in nuclear plants. These methods are to provide capabilities for establishing degradation trends prior to failure and developing guidance for effective maintenance. This report examines failure modes and causes resulting from aging and service wear, manufacturer-recommended maintenance and surveillance practices, and measurable parameters (including functional indicators) for use in assessing operational readiness, establishing degradation trends, and detecting incipient failure. The results presented are based on information derived from operating experience records, nuclear industry reports, manufacturer-supplied information, and input from architect-engineer firms and plant operators.

NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984. KELMERS,A.D.; SEELEY,F.G.; ARNOLD,W.D.; et al. Oak Ridge National Laboratory. October 1985. 59pp. 8511180625. ORNL/TM-9614. 33504:334.

Geochemical information relevant to the retention of radionuclides by candidate high-level nuclear waste geologic repositories being characterized by Department of Energy (DOE) projects is being evaluated by Oak Ridge National Laboratory (ORNL) for the Nuclear Regulatory Commission (NRC). During this report period, emphasis was given to evaluation of published sorption and solubility information for key radionuclides which is relevant to the Hanford Site in the Columbia River basalts. The removal of neptunium from solution by basalt/groundwater systems under anoxic redox conditions at 60 degrees centigrade proved to be sensitive to the basalt particle size and the test contact time. It was not possible to establish if the neptunium removal from solution was due to sorption or precipitation processes. In studies of uranium solubility, sodium boltwoodite was shown to be the U(VI)- containing phase that precipitates from synthetic groundwater at 60 degrees centigrade. The precipitation of sodium boltwoodite, rather than schoepite which is predicted by geochemical modeling, shows the importance of identifying the solid phase in radionuclide experiments and highlights the weaknesses of the actinide thermodynamic data bases used in geochemical modeling calculations. An evaluation was made of the information developed by DOE on the native copper deposits of Michigan as a natural analog for the possible emplacement of copper canisters in repository in basalt. The similarity in bulk chemistry of the basalts, relied upon heavily by DOE in their analysis, cannot be used to unequivocally conclude that similar geochemical controls, particularly controls on the geochemical conditions, exist within the basalt/water systems at Michigan and the Hanford Site. Thus, the DOE analysis is insufficient to conclude, with reasonable assurance, that copper will be stable at the Hanford Site.

NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985. KELMERS,A.D.; SEELEY,F.G.; ARNOLD,W.D.; et al. Oak Ridge National Laboratory. December 1985. 52pp. 8601070466. ORNL/TM-9614. 34155:078.

Geochemical information relevant to the retention of radionuclides by the Hanford Site (in basalt) and the Yucca Mountain site (in tuff), candidate high-level nuclear waste geologic repositories being developed by U.S. Department of Energy (DOE) projects, is being evaluated by Oak Ridge National Laboratory (ORNL) for the U.S. Nuclear Regulatory Commission (NRC). Our evaluation of the sorption of technetium by basalt/groundwater systems was essentially completed this quarter and the results summarized; we conclude that the experimental methodology and results reported by the DOE for the Hanford Site have not conclusively established that significant retardation of technetium migration may be provided by phases present in the basalts of the Hanford Site. We have shown that sodium boltwoodite is the saturating uranium solid phase in two basalt/groundwater systems. Because thermodynamic data are not available for sodium boltwoodite, calculated solubilities for uranium are erroneous in these systems. Results of radionuclide solubility/speciation calculations, published by the DOE for the Yucca Mountain site, were evaluated this quarter under our geochemical modeling task. We express concerns relative to the inherent limitations of such calculations. Samples of Yucca Mountain tuff and J-13 well water were received for use in our planned radionuclide sorption/solubility experiments. These Yucca Mountain materials will be used to evaluate radionuclide sorption and apparent concentration limit values published by the Nevada Nuclear Waste Storage Investigation (NNWSI) project.

NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS. SIMPSON,H.J.; HERCZEG,A.L.; ANDERSON,R.F.; et al. Columbia Univ., New York, NY. April 1985. 77pp. 8505070484. 30210:219.

Concentrations of naturally occurring isotopes of uranium, thorium, radium and radon were measured in freshwaters and in sodium-chloride brines near the site of the Waste Isolation Pilot Plant (WIPP) located in southeastern New Mexico. Supplemental water chemistry analyses (chloride, alkalinity, P(CO2), CO(2), Fe, Mn, H(2)S) were made to aid in interpreting the data for natural radionuclides. Three features of radionuclide mobility are evident from the results: 1) There is a slight tendency for U and Ra concentrations to correlate with the chloride content of the water samples. Whether this tendency results from complexation by Cl- ions or cation exchange competition for adsorption sites cannot be resolved with the available information. 2) Much more dramatic than the correlation with Cl- concentration is the effect of the redox state of the waters on U and Ra concentrations. Chemically reducing groundwaters contain much lower U concentrations and much higher Ra concentrations than were measured in oxic and suboxic samples. Calculated retardation factors of 1 for Ra indicate that it can migrate freely in anoxic brines. 3) Low chemical recoveries of Th, and to a lesser extent U were observed for methods that work well with seawater samples. These elements may be present in a mobile, unreactive dissolved or colloidal complex with organic matter.

NUREG/CR-4239: ANALYSIS OF THE ABILITY OF CURRENT HEALTH PHYSICS INSTRUMENTS TO PREDICT DOSE IN EXPOSED INDIVIDUALS. ARMANTROUT,G.A. Lawrence Livermore National Laboratory. July 1985. 315pp. 8507250126. UCID-20398. 31788:001.

In this study, theoretical calculations of effective dose to body tissue using Monte Carlo simulation techniques have been performed for both gamma ray and beta ray irradiation. Similar calculations for neutron irradiation by other workers have also been reviewed. Evaluations were made of the performance of a series of the more common health physics instruments. In this evaluation, representative instruments for both gamma-ray and beta-ray survey work were evaluated using a series of calibrated radiation sources. These instrument evaluations were then compared against similar evaluations in the literature, and an evaluation of basic instrument response by type was performed. In addition, data on calculated health effects was used to evaluate the ability of these instruments to predict health effects. Key results for the gamma-ray, beta-ray, and neutron survey meters are given.

NUREG/CR-4240 V01: PHYSICS OF REACTOR SAFETY.Quarterly Report,January-March 1985. * Argonne National Laboratory. July 1985. 27pp. 8508[illegible]779. ANL-85-23 V01. 31923:219.

This quarterly progress report summarizes work done during the months of January-March 1985 in Argonne National Laboratory's Applied Physics and Components Technology Divisions for the Division of Reactor Safety Research in 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 at ANL's Components Technology Divison, emphasizing 3-dimensional code development for LMFBR accidents under natural convection conditions. An executive summary is provided including a statment of the findings and recommendations of the report.

NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION. DUCE,S.W.; CRONEY,S.T.; AKERS,D.W.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). June 1985. 895pp. 8507020396. EGG-2392. 31311:061.

This report presents data obtained at Brunswick as part of the In-Plant Source Term Measurement Program in operating light water reactors (LWRs). The work was conducted for the Office of Nuclear Regulatory Research (RES) in support of the Meteorology and Effluent Treatment Branch (METB) of the Office of Nuclear Reactor Regulation (NRR). The primary objective of this program is to provide the Nuclear Regulatory Commission (NRC) with operational data that can be used in evaluation of plant designs for liquid and gaseous radwaste treatment systems. Data presented were obtained at the Brunswick Nuclear Generating Station, operated by Carolina Power and Light, located at Southport, North Carolina. In-plant measuremens were conducted during the time period from March 1982 to November 1982. This plant is the sixth in a series of operating LWRs to be studied and the first boiling water reactor (BWR) in the series.

NUREG/CR-4248: RECOMMENDATIONS FOR NRC POLICY ON SHIFT SCHEDULING AND OVERTIME AT NUCLEAR POWER PLANTS. LEWIS,P.M. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1985. 150pp. 8508090710. PNL-5435. 32102:343.

This report contains the Pacific Northwest Laboratory's (PNL's) recommendations to the U.S. Nuclear Regulatory Commission (NRC) for an NRC policy on shift scheduling and hours of work (including overtime) for control room operators and other safety-related personnel in nuclear power plants. First, it is recommended that NRC make three additions to its present policy on overtime: 1) limit personnel to 112 hours of work in a 14-day period, 192 hours in 28 days, and 2,260 hours in one year; exceeding these limits would require plant manager approval, 2) add a requirement that licensees obtain approval from NRC if plant personnel are expected to exceed 72 hours of work in a 7-day period, 132 hours in 14 days, 228 hours in 28 days, and 2,300 hours in one year, and 3) make the policy a requirement, rather than a nonbinding recommendation. Second, it is recommended that licensees be required to obtain NRC approval to adopt a routine 12-hour/day shift schedule. Third, it is recommended that NRC add several nonbinding recommendations concerning routine 8-hour/day schedules. Finally, because additional data can strengthen the basis for future NRC policy on overtime, five methods are suggested for collecting data on overtime and its effects.

NUREG/CR-4249: PRESSURE VESSEL FRACTURE STUDIES PENETRATING TO THE PWR THERMAL-SHOCK ISSUE:EXPERIMENTS TSE-5,TSE-5A AND TSE-6. CHEVERTON,R.D.; BALL,D.G.; BOLT,S.E.; et al. Oak Ridge National Laboratory. July 1985. 255pp. 8507250152. ORNL-6163. 31791:140.

Thermal-shock experiments TSE-5, TSE-6 were conducted for the purpose of investigating the behavior of surface flaws under pressurized-water-reactor (PWR) overcooling-accident conditions. These experiments were the fifth, sixth, and seventh in a series of thermal-shock experiments conducted with large steel cylinders (A508, class-2 chemistry; 991-mm OD x 76- and 152-mm wall x 1.2-m length) as a part of the Heavy-Section Steel Technology (HSST) Program for this purpose. For each of these experiments the initial flaw was on the inner surface and extended the full length of the cylinder. The thermal shock was applied to the inner surface only, and this was accomplished by effectively dunking the test cylinder, initially at 93 degrees centigrade, into a large volume of liquid nitrogen. Results of the experiments have confirmed that (1) linear-elastic fracture mechanics (LEFM) is valid for thermal-shock loading, (2) crack arrest will take place in accordance with recently developed crack-arrest concepts, (3) the crack-arrest toughness values for rising and falling K(I) fields are the same, (4) warm prestressing is effective in preventing crack initiation, (5) thermal shock alone cannot drive a flaw all the way through the wall, (6) dynamic effects for PWR-vessel thermal-shock loading conditions are negligible, (7) in the absence of cladding and under severe thermal-shock loading conditions finite-length flaws will extend on the surface to become very long, and (8) there can be very large scatter in small-specimen fracture-toughness data.

NUREG/CR-4250: VEHICLE BARRIERS:EMPHASIS ON NATURAL FEATURES. ADAMS,K.G.; ROSCOE,B.J. Sandia National Laboratories. September 1985. 110pp. 8509300238. SAND85-0935. 32815:292.

The recent increase in the use of car and truck bombs by terrorist organizations has led NRC to evaluate the adequacy of licensee security against such threats. As part of this evaluation, one of the factors is the effectiveness of terrain and vegetation in providing barriers against the vehicle entry. The effectiveness of natural features is presented in two contexts. First, certain natural features are presented. In addition to the discussion of natural features, this report provides a discussion of methods to slow vehicles. Also included is an overview of man-made barrier systems, with particular attention to ditches.

NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions. OBERLANDER,P.L.; SKAGGS,R.L.; SHAFER,J.M. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1985. 321pp. 8509110279. PNL-5461. 32559:004.

Pacific Northwest Laboratory evaluated the feasibility of using ground-water containment mitigation techniques to control radionuclide migration following a severe commercial nuclear power reactor accident. The two types of severe commercial reactor accidents investigated are 1) containment basemat penetration of core melt debris, which slowly cools and leaches radionuclides to the subsurface environment; and 2) containment basemat penetration of sump water without full penetration of the core mass. Six generic hydrogeologic site classifications were developed from an evaluation of reported data pertaining to the hydrogeologic properties of all existng and proposed commercial reactor sites. One-dimensional radionuclide transport analyses were conducted on each of the individual reactor sites to determine the generic characteristics of a radionuclide discharge to an accessible environment. Ground-water containment mitigation techniques that may be suitable for severe power plant accidents, depending on specific site and accident conditions, were identified and evaluated. Feasible mitigative techniques and associated constraints on feasibility were determined for each of the six hydrogeologic site classifications. Three case studies were conducted at power plant sites located along the Texas Gulf Coast and the Ohio River. Mitigative strategies were evaluated for their impact on containment transport. Results show that the techniques evaluated significantly increased ground-water travel times and reduced contaminant migration rates.

NUREG/CR-4251 V02: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 2:Case Study Analysis Of Hydrologic Characterization And Mitigative Schemes. OBERLANDER,P.L.; SKAGGS,R.L.; SHAFER,J.M. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1985. 301pp. 8509100515. PNL-5461. 32538:094.

See NUREG/CR-4215,V01 abstract.

NUREG/CR-4252: INDEPENDENT ASSESSMENT OF TRAC-PD2/MOD1 CODE WITH BCL ECC BYPASS TESTS. SLOVIK,G.C.; SAHA,P. Brookhaven National Laboratory. August 1985. 78pp. 8509180076. 32668:213.

This report presents the TRAC-PD2/MOD1 independent assessment calculations performed at Brookhaven National Laboratory (BNL) using the Emergency Core Cooling (ECC) bypass experiments conducted in a 2/15-scale PWR vessel at Battelle Columbus Laboratories (BCL). Both steady-state experiments with various ECC water subcoolings and transient tests with hot wall effects were simulated. Besides the base cases, several sensitivity calculations were performed to study the effects of nodalization, particularly the relative locations of the hot leg penetrations in the downcomer. In addition, calculations were performed to determine the effect of slig[illegible] creases in the reverse core steam flow and the associated form losses due to the hot leg penetrations. Code corrections as received from the code developers at Los Alamos National Laboratory (LANL) were also incorporated into this study.

NUREG/CR-4253: REVIEW OF TRAC CALCULATIONS FOR CALVERT CLIFFS PTS STUDY. JO,J.H.; ROHATGI,U.S. Brookhaven National Laboratory. April 1985. 114pp. 8511010391. BNL-NUREG-51887. 33300:231.

Six selected transient calculations out of thirteen performed by LANL using the TRAC-PF1 code for the USNRC PTS study of the Calvert Cliffs Nuclear Power Plant have been reviewed in depth at BNL. Simple hand calculations based on the mass and energy balances have been performed to predict the temperature and pressure of the reactor system, and the results have been compared with those of TRAC. Comparison was also made between the TRAC and RETRAN calculations for two of these transients, which were performed by ENSA. In general, the results calculated by TRAC appear to be reasonable based on the comparison with RETRAN and hand calculations.

NUREG/CR-4254: OCCUPATIONAL DOSE REDUCTION AND ALARA AT NUCLEAR POWER PLANTS.Study On High-Dose Jobs,Radwaste Handling,And ALARA Incentives. DIONNE,B.J.; BAUM,J.W. Brookhaven National Laboratory. July 1985. 104pp. 8508090697. BNL-NUREG-51888. 32105:192.

The purpose of this report is to provide the NRC and the nuclear industry with information and data which will be useful for occupational dose reduction at nuclear power plants. The objectives of this effort were to: 1. identify the repetitive high-dose jobs, related collective dose ranges and applicable dose reduction techniques, 2. investigate and recommend improvements in the selection of high reliability and low maintenance equipment to assure that collective doses received during equipment repair is considered, 3. recommend improved radioactive waste handling procedure and equipment which could reduce collective dose equivalent, and 4. examine current ALARA incentives and recommend new positive steps which will provide additional dose-reduction incentives. Ten nuclear sites were visited by two Brookhaven health physicists to collect the needed dose-reduction data and information. This report summarizes the findings and recommendations on the above objectives.

NUREG/CR-4255 V01: AEROSAL RELEASE AND TRANSPORT PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985. ADAMS,R.E.; TOBIAS,M.L. Oak Ridge National Laboratory. August 1985. 56pp. 8509110010. ORNL/TM-9632/V1. 32565:009.

This report summarizes progress for the Aerosol Release and Transport Program sponsored by the Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Division of Accident Evaluation, for the period October 1984-March 1985. Topics discussed include (1) steam-only experiments in the NSPP facility; (2) tests in small vessels to study thermal output, mass generation rates, and other operating features of plasma torch aerosol generators in support of the development of the Aerosol Moisture Interaction Tests (AMIT) facility and to support the LWR Aerosol Containment Experiments (LACE) program at Hanford; (3) analysis of data from plasma torch aerosol generator tests; (4) analysis of steam behavior in the NSPP vessel in aerosol experiments and in steam only tests; and (5) a study of the feasibility of experiments for shape factor measurements.

NUREG/CR-4256: MEASUREMENT OF RESPONSE TIME AND DETECTION OF DEGRADATION IN PRESSURE SENSOR/SENSING LINE SYSTEMS. BUCHANAN,M.E.; MILLER,L.F.; THIE,J.A.; et al. Oak Ridge National Laboratory. September 1985. 107pp. 8512270230. ORNL/TM-9574. 34080:258.

A team evaluated several methods for remote measurement of the response time and detection of degradation (blockage or air in lines) of pressure sensor/sensing line systems typical of nuclear power plants. A method was developed for obtaining the response time of force-balance pressure transmitters by briefly interrupting the power supply to the transmitter. The data thus generated are then analyzed in conjunction with a model to predict transmitter response to an actual pressure perturbation. The research team also evaluated a pressure perturbation method for determining the asymptotic delay time of a pressure-sensing line and found that this method yields accurate results for essentialy unblocked sensing lines. However, these pressure perturbation tests are not recommended for use in nuclear power plants because they re difficult to implement on-line. A third method for remote measurement applied noise analysis methods that yielded accurate estimates of asymptotic delay times for blockage or air in sensing lines. Even though noise analysis methods worked well in the laboratory, it is recommended that further evaluation be performed in operating nuclear plants.

NUREG/CR-4257: INSPECTION,SURVEILLANCE,AND MONITORING OF ELECTRICAL EQUIPMENT INSIDE CONTAINMENT OF NUCLEAR POWER PLANTS--WITH APPLICATIONS TO ELECTRICAL CABLES. AHMED,S.; CARFAGNO,S.P. ARVIN/CALSPAN Advanced Technology Center. August 1985. 101pp. 8509100530. 32536:234.

The general concepts of equipment condition monitoring as applicable to the detection of age-related deterioration of safety-related equipment are described. The goal is to detect deterioration in the incipient stage, prior to in-service failure and prior to the point at which equipment can no longer be expected to perform its function when exposed to design basis accident conditions. The application of condition monitoring is discussed specifically for electrical cables. The goal of cable condition monitoring is to determine the degree of cable degradation and to predict the remaining useful life. In situ nondestructive testing and destructive laboratory testing are discussed. Interim recommendations are given for the implementation of a condition monitoring program.

NUREG/CR-4258: AN APPROACH TO TEAM SKILLS TRAINING OF NUCLEAR POWER PLANT CONTROL ROOM CREWS. DAVIS,L.T.; GADDY,C.D.; TURNEY,J.R. General Physics Corp. July 1985. 83pp. 8508200093. GP-R-123022. 32278:020.

An investigation of current team skills training practices and research was conducted by General Physics Corporation for the Office of Nuclear Reactor Regulation. The methodology used included a review of relevant team skills training literature and a workshop to collect inputs from team training practitioners and researchers from the public and private sectors. The workshop was attended by representatives from nuclear utility training organizations, the commercial airline industry, federal agencies, and defense training and research commands. The literature reviews and workshop results provided the input for a suggested approach to team skills training that can be integrated into existing training programs for control room operating crews. The approach includes five phases: (1) team skills objectives development, (2) basic team skills training, (3) team task training, (4) team skills evaluation, and (5) team training program evaluation. Supporting background information and a user-oriented description of the approach to team skills training are provided.

NUREG/CR-4259: TAILINGS NEUTRALIZATION AND OTHER ALTERNATIVES FOR IMMOBILIZING TOXIC MATERIALS IN TAILINGS.Final Report. OPITZ,B.E.; SHERWOOD,D.R.; DODSON,M.E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1985. 131pp. 8510040347. PNL-5467. 32857:159.

This final document, in a series of six, summarizes research completed since the beginning of the project. Three subtasks are included: Subtask A - Neutralization Methods Selection; Subtask B - Laboratory Analysis; and Subtask C - Field Testing. Subtask A reviews treatment processes from other industries to evaluate if current waste technology from other fields is applicable to the uranium industry. This task also identifies several reagents that were tested for their effectiveness in treating acidic tailings and tailings solution in order to immobilize the contaminants associated with the acid waste. Subtask B describes the laboratory batch and column treatment studies performed on solid waste tailings and tailings solution over the course of the project. The evaluation of several reagents identified in Subtask A was based on three criteria: 1) treated effluent water quality, 2) neutralized sludge handling and hydraulic properties, and 3) reagent costs and acid neutralizing efficiency. Subtask C presents a field demonstration plan that will evaluate the effectivenss, costs and benefits of neutralizing acidic uranium mill tailings solution to reduce the potential leaching of toxic trace metals, radionuclides and macro ions from a tailings impoundment.

NUREG/CR-4260: TORAC USER'S MANUAL.A Computer Code For Analyzing Tornado-Induced Flow And Material Transport In Nuclear Facilities. ANDRAE,R.W.; TANG,P.K.; MARTIN,R.A.; et al. Los Alamos Scientific Laboratory. July 1985. 145pp. 8507250118. LA-10435-M. 31787:001.

This manual describes the TORAC computer code, which can model tornado-induced flows, pressures, and material transport within structures. Future versions of this code will have improved analysis capabilities. In addition, it is part of a family of computer codes that is designed to provide improved methods of safety analysis for the nuclear industry. TORAC is directed toward the analysis of facility ventilation systems, including interconnected rooms and corridors. TORAC is an improved version of the TVENT code. In TORAC, blowers can be turned on and off and dampers can be controlled with an arbitrary time function. The material transport capability is very basic and includes convection, depletion, entrainment, and filtration of material. The input specifications for the code and variety of sample problems are provided.

NUREG/CR-4262 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR.Main Report. BRUSKE,S.J.; BAXTER,D.E.; RANSOM,C.B.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1985. 81pp. 8506240180. EGG-2394. 31150:280.

This report documents the evaluation of the effects of nonsafety grade control system failures on a typical boiling water reactor plant. The methods utilized in this evaluation include a system level failure modes and effects analysis, deterministic computer analysis, a review of 3 years of recorded plant occurrences, a probability analysis and a review of applicable NRC criteria pertaining to control systems. This study identified three system failures that could cause transients leading to a reactor vessel overfill and of these three failures, two could also lead to a reactor coolant cooldown of greater than 100 degrees fahrenheit per hour. This study concluded that the existing NRC criteria, concerning control systems, adequately address the potential problem areas that were identified during this evaluation. Based on the results of this study, it was recommended that the consequences and risk associated with overfill and overcool transients be further investigated.

NUREG/CR-4262 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR. Appendices. BRUSKE,S.J.; BAXTER,D.E.; RANSOM,C.B.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1985. 341pp. 8506240689. EGG-2394. 31179:036.

Safety Implications of Control Systems (A-47) was approved as an Unresolved Safety Issue (USI) by the Nuclear Regulatory Commission (NRC) in December of 1980. USI A-47 concerns the potential for transients or accidents being made more severe as a result of control system failures. This report describes the work performed on the effects of control system failures on transients and accidents at a General Electric boiling water reactor. This work was conducted for the U.S. Nuclear Regulatory Commission, Division of Safety Technology by EG&G Idaho, Inc. and is based on the Browns Ferry Nuclear Plant. This report is contained in two volumes: a main report and five appendices. The main report describes the study methodology, the major areas of work performed, and the results and conclusions. The appendices contain detailed information consisting of failure mode and effects analysis tables, a detailed description of the computer analyses and significant transient excerpts.

NUREG/CR-4263: RELIABILITY ANALYSIS OF STIFF VERSUS FLEXIBLE PIPING FINAL PROJECT REPORT. LU,S.C.; CHOU,C.K. Lawrence Livermore National Laboratory. May 1985. 78pp. 8505280086. UCRL-20410. 30604:169.

This research project is to develop a technical basis for flexible piping designs which will improve piping reliability and minimize the use of pipe supports, snubbers, and pipe whip restraints. This study indicated that piping design can be made more reliable by some reduction of rigid supports and/or snubbers. This study also confirmed that the malfunction of pipe whip restraints introduced higher thermal stresses and tended to reduce the overall piping reliability. Finally, our results indicated that supports in a flexible piping design may need to be re-evaluated and that the elimination of pipe supports which are close to components should be done with care in order to minimize the impact on the component reliability.

NUREG/CR-4264: INVESTIGATION ON HIGH-EFFICIENCY PARTICULATE AIR FILTER PLUGGING BY COMBUSTION AEROSOLS. FENTON,D.L.; GREGORY,W.S.; et al. Los Alamos Scientific Laboratory. GUNAJI,M.V. New Mexico State Univ., Las Cruces, NM. May 1985. 32pp. 8507050422. LA-10436-MS. 31375:021.

Experiments were conducted to investigate high-efficiency particulate air (HEPA) filter plugging by combustion aerosols. These tests were done to obtain empirical data to improve our modeling of filter plugging phenomena using the Los Alamos National Laboratory fire accident analysis code FIRAC. Commercially available 0.61-m by 0.61-m square filters were tested in a specially designed facility to determine how airflow resistance varies with increased filter loading by combustion aerosols. Two organic fuels normally found in nuclear fuel cycle facilities, polystyrene (PS) and polymethylmethacrylate (PMMA), were burned under varied conditions to generate combustion aerosols. The test facility included a combustor, a 23-m-long duct, and a specially designed gravimetric balance for determing the aerosol mass gain of the filters. Test results include correlations of HEPA filter resistance ratios (actual resistance/initial resistance) with aerosol mass gain. The mass gain of plugged HEPA filters was found to correlate with the airborne mass concentration of material in the size range greater that approximately 2.0 m. Also, the fuel with a smaller soot fraction, PMMA, produced filter plugging at lower accumulated aerosol mass deposits on or within the filter.

NUREG/CR-4266: STANDARD BETA-PARTICLE AND MONOENERGETIC ELECTRON SOURCES FOR THE CALIBRATION OF BETA-RADIATION PROTECTION INSTRUMENTATION. EHRLICH,M.; PRUITT,J.S.; SOARES,C.G.; et al. Commerce, Dept. of, National Bureau of Standards. August 1985. 86pp. 8509060202. NBSIR 85-3169. 32505:057.

In a project funded jointly by the National Bureau of Standards (NBS) and the Nuclear Regulatory Commission (NRC), NBS has developed a calibration facility for beta-particle instruments and sources used in radiation-protection dosimetry. The facility consists of beta-particle and nearly monoenergetic electron beams characterized in terms of absorbed-dose rates to plastic and in terms of beta-particle spectra. A second phase of the project was concerned with establishing secondary calibration laboratories for radiation-protection instruments. This final report includes a detailed discussion of (1) the determination of absorbed-dose rates to plastic for each beta-particle and nearly monoenergetic electron beam, dose-rate dependence on altitude above sea level, and an estimate of the overall uncertainties in dose-rate measurements; (2) beta-particle and nearly monoenergetic electron spectra and their dependence on source configuration; and (3) degree of achievable uniformity of beam cross sections. Included also is a review of the results of a first attempt to predict instrument response to realistic beta-particle environments from their response to monoenergetic electrons and knowledge of the approximate beta-particle spectra. Attached to the report are proposed guidelines for establishing secondary calibration laboratories for radiation-protection instruments.

NUREG/CR-4267: VESSEL INTEGRITY SIMULATION (VISA) CODE SENSITIVITY STUDY. SIMONEN,E.P.; JOHNSON,K.I.; SIMONEN,F.A. Battelle Memorial Institute, Pacific Northwest Laboratories. December 1985. 49pp. 8601070522. PNL-5469. 34186:043.

In a study conducted for the Nuclear Regulatory Commission by Pacific Northwest Laboratory, the sensitivity of through-wall crack probability to input distributions was studied. Flaw growth characteristics were evaluated for three pressurized water reactor plants (Oconee 1, Calvert Cliffs 1, and a hypothetical plant similar to H. B. Robinson 2). Three postulated pressurized thermal shock (PTS) transients were considered for each plant. This report describes the results of material and flaw distribution assumptions on calculated conditional failure probabilities for the predicted sensitivites are evaluated and are related to requirements for defining input distributions for probabilistic failure predictions.

NUREG/CR-4268: RATIO METHODS FOR COST-EFFECTIVE FIELD SAMPLING OF COMMERCIAL RADIOACTIVE LOW-LEVEL WASTES. EBERHARDT,L.L.; SIMMONS,M.A.; THOMAS,J.M. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1985. 81pp. 8508090570. PNL-5156. 32106:069.

An investigation of cost-effective methods for sampling at commercial radioactive low-level waste sites has been one goal of this project. To that end, double sampling was investigated, and we found that the method appears useful when estimating total radionuclide inventory in waste site environs. The methods are explained, decision criteria for cost effectiveness presented, and a worked example based on field data is provided. The statistical basis for the conclusion that double sampling appears to be robust and cost-effective is in separate sections. Field tests and additional estimates of "field instrument" errors are needed to substantiate the findings.

NUREG/CR-4271: RECOMMENDED SAFETY,RELIABILITY,QUALITY ASSURANCE AND MANAGEMENT AEROSPACE TECHNIQUES WITH POSSIBLE APPLICATION BY THE DOE TO THE HIGH LEVEL RADIOACTIVE WASTE REPOSITORY PROGRAM. BLAND,W.M. GeeB's, Inc. June 1985. 113pp. 8507080205. 31393:164.

Aerospace SRQA and management techniques, principally those developed and used by the NASA Lyndon B. Johnson Space Center on the manned space flight programs, have been assessed for possible application by the DOE and the DOE-contractors to the high level radioactive waste repository program that results from the implementation of the NWPA of 1982. Those techniques believed to have the greatest potential for usefulness to the DOE and the DOE-contractors have been discussed in detail and are recommended to the DOE for adoption; discussion is provided for the manner in which this transfer of technology can be implemented. Six SRQA techniques and two management techniques are recommended for adoption by the DOE; included with the management techniques is a recommendation for the DOE to include a licensing interface with the NRC in the application of the milestone review technique. These other techniques are recommended for study by the DOE for possible adaption to the DOE program.

NUREG/CR-4272: RESPONSE TREE EVALUATION:EXPERIMENTAL ASSESSMENT OF AN EXPERT SYSTEM FOR NUCLEAR REACTOR OPERATORS. NELSON,W.R.; BLACKMAN,H.S. EG&G Idaho, Inc. (subs. of EG&G, Inc.). September 1985. 68pp. 8510040397. EGG-2397. 32859:203.

The United States Nuclear Regulatory Commission (USNRC) sponsored a project performed by EG&G Idaho, Inc., at the Idaho National Engineering Laboratory (INEL) to evaluate different display concepts for use in nuclear reactor control rooms. Included in this project was the evaluation of the response tree computer-based decision aid and its associated displays. The response tree evaluation task was designed to (a) assess the merit of the response tree decision aid and (b) develop a technical basis for recommendations, guidelines, and criteria for the design and evaluation of computerized decision aids for use in reactor control rooms. Two major experiments have been conducted to evaluate the response tree system. This report emphasizes the conduct and results of the second experiment. An enhanced version of the response tree system, known as the automated response tree system, was used in a controlled experiment using trained reactor operators as test subjects. This report discusses the automated response tree system, the design of the evaluation experiment, and the quantitative results of the experiments. The results of the experiments are compared to the results of the previous experiments to provide an integrated perspective of the response tree evaluation project. In addition, a subjective assessment of the results addresses the implications for the use of advanced "intelligent" decision aids in the reactor control room.

NUREG/CR-4274: ANALYSIS AND TESTS ON SMALL-SCALE SHEAR WALLS FY-82 FINAL REPORT. ENDEBROCK,E.G.; DOVE,R.C.; DUNWOODY,W.E. Los Alamos Scientific Laboratory. September 1985. 59pp. 8512270234. LA-10443-MS. 34084:109.

The Phase-I experimental program was completed during FY 1982. This report summarizes the results of (1) quasistatic (monotonic and load-cycling) tests, (2) sinusoidal vibration tests, and (3) simulated earthquake tests conducted on small-scale, reinforced-concrete shear walls. Model construction, test methods, instrumentation, and experimental results are presented in this report. Experimental results are interpreted to investigate the effects of high-load levels (which produce cracking and failure of the walls) on stiffness, damping, and on deformation and acceleration transmissibility. The nonlinear analysis method that has been developed as part of this program has been used to aid in the interpretation of these experimental results.

NUREG/CR-4275: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM - FIVE-YEAR PLAN FY 1984-1988. * Oak Ridge National Laboratory. August 1985. 160pp. 8509110024. ORNL/TM-9654. 32558:001.

The second in an annual series of five-year program plan documents is presented for the Heavy-Section Steel Technology program. The program is carried out by the Oak Ridge National Laboratory for the Materials Engineering Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission. The program is aimed at advancing the understanding and validation of materials and structures behavior as they relate to light water reactor pressure vessel integrity. The program has nine technical tasks and a management function. A background statement and a plan-of-action is given for each. The nine technical tasks address fracture methodology and analysis, materials characterization, crack growth, crack arrest, irradiation effects, cladding evaluations, intermediate-vessel testing, thermal-shock testing, and pressurized thermal-shock experiments.

NUREG/CR-4276: VIBRATION AND WEAR IN STEAM GENERATOR TUBES FOLLOWING CHEMICAL CLEANING - SEMIANNUAL REPORT. ENDERLIN,W.I.; BAUGH,J.W. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 38pp. 8507030714. PNL-5477. 31322:273.

The Pacific Northwest Laboratory is studying the effects of increased tube/tube-support clearances in pressurized water reactor steam generators following chemical cleaning. The project purpose is to provide NRC with criteria for evaluating licensees' specific proposals for chemical cleaning of steam generators. This report describes the test and data analysis plans and procedures for the flow and accelerated wear tests to be performed in a scale-model steam generator. The flow tests will establish the forcing boundary conditions, using clearances representing various conditions following chemical cleaning. The accelerated wear tests will determine the potential wear rates possible, based on the vibrations characterized in the flow tests. The overall project status, including work completed to date and tasks planned for the remainder of FY85, is also documented.

NUREG/CR-4277: INVERTED ANNUAL FLOW EXPERIMENTAL STUDY. DE JARLAIS,G.; ISHII,M. Argonne National Laboratory. April 1985. 115pp. 8507050406. ANL-85-31. 31338:074.

Steady-state inverted annular flow of Freon 113 in up flow was established in a transparent test section. Using a special inlet configuration consisting of long aspect-ratio liquid nozzles coaxially centered within a heated quartz tube, idealized inverted annular flow initial geometry (cylindrical liquid core surrounded by coaxial annulus of gas) could be established. Inlet liquid and gas flowrates, liquid subcooling, and gas density (using various gas species) were measured and varied systematically. The hydrodynamic behavior of the liquid core, and the subsequent downstream break-in of this core into slugs, ligaments and/or droplets of various sizes, was observed. In general, for low inlet liquid velocities it was observed that after the initial formation of roll waves on the liquid core surface, an agitated region of high surface area, with attendant high momentum and energy transfers, occurs. This agitated region appears to propagate downstream in a quasi-periodic pattern. Increased inlet liquid flow rates, and high gas annulus flow rates tend to diminish the significance of this agitated region. Observed inverted annular flow (and subsequent downstream flow pattern) hydrodynamic behavior is reported, and comparisions are drawn to data generated by previous experimenters studying post-CHF flow.

NUREG/CR-4278: TRAC-PF1/MOD1 DEVELOPMENT ASSESSMENT. SAHOTA,M.S.; ADDESSIO,F.L. Los Alamos Scientific Laboratory. August 1985. 350pp. 8511010462. LA-10445-MS. 33302:268.

The Transient Reactor Analysis Code (TRAC) is being developed at Los Alamos National Laboratory to provide advanced best-estimate predictions of postulated accidents in light-water reactors. The TRAC-PF1/MOD1 program provides this capability for pressurized water reactors and for many thermal-hydraulic experimental facilities. The code features either a one- or three-dimensional treatment of the pressur vessel and its associated internals; a two-phase, two-fluid, nonequilibrium hydrodynamics model with a noncondensable gas field; flow-regime-dependent constitutive equation treatment; optional reflood-tracking capability for both bottom flood and falling-film quench fronts; and consistent treatment of entire accident sequences from nomal operating conditions through severe trasients. A new numerical algorithm is used in the one-dimensional hydrodynamics that permits this portion of the fluid dynamics to violate the material Courant condition. This technique permits large time steps and, hence, reduced running time for slow transients. This report presents the results of initial developmental assessment calculations performed with TRAC-PF1/MOD1 before its public release. The assessment set consists of six integral effects calculations in the Loss-of-Fluid test and Semiscale facilities. Computer run times required to predict each test also are reported.

NUREG/CR-4280: THE EFFECTS OF SUPERVISOR EXPERIENCE AND ASSISTANCE OF A SHIFT TECHNICAL ADVISOR (STA) ON CREW PERFORMANCE IN CONTROL ROOM SIMULATORS. BEARE,A.N.; DONOVAN,M.D.; LASSITER,D.L.; et al. General Physics Corp. September 1985. 202pp. 8510040318. ORNL/TM-9660. 32861:049.

This report describes the second experiment using a training simulator to evaluate effects of experience level of Senior Reactor Operators (SRO) in the supervisor's role, and presence of a Shift Technical Advisor (STA) on performance of nuclear power plant control room operators/crews. The experiment was conducted in a pressurized water reactor (PWR) plant-referenced simulator. Data was collected on 20 three-man crews of licensed operators, performing four sequences. Performance measures were derived from task analyses of the sequences.

One set of measures focused on task performance; the second set measured control of system parameters. Instructors' ratings and performance scores of trainees were compared to scores of operators/crews, to validate the performance measures. System parameters and control manipulations were recorded by the simulator's computer. Communications and selected verifications were recorded on checklists and videotapes. Questionnaires recorded biographical information and self-reported workloads. No significant differences in overall performance were found attributable to experience of supervisors, nor to presence of a STA. Results were similar to results of an earlier experiment performed with boiling water reactor (BWR) crews. These results are also reported. Reported workloads of supervisors assisted by STA/s were significantly lower than workloads reported by those unassisted.

NUREG/CR-4281: AN EMPIRICAL ANALYSIS OF SELECTED NUCLEAR POWER PLANT MAINTENANCE FACTORS AND PLANT SAFETY. OLSON,J.; OSBORN,R.N.; THURBER,J.A.; et al. Battelle Human Affairs Research Centers. July 1985. 62pp. 8508150065. PNL-5487. 32196:224.

This report contains a statistical analysis of the relationship between selected aspects of nuclear power plant maintenance programs and safety related performance. The report identifies a large number of maintenance resources which can be expected to influence maintenance performance and subsequent plant safety performance. The resources for which data were readily available were related statistically to two sets of performance indicators: maintenance intermediate safety indicators, and final safety performance indicators. The results show that the administrative structure of the plant maintenance program is a significant predictor of performance on both sets of indicators.

NUREG/CR-4283: STUDY OF THE EFFECTS OF ELASTIC UNLOADINGS ON THE JI-R CURVES FROM COMPACT SPECIMENS. SUTTON,G.E.; VASSILAROS,M.G. David W. Taylor Naval Research & Development Center. June 1985. 50pp. 8506260731. 31227:118.

An investigation was performed to evaluate the efforts of elastic unloadings on the J-Integral Resistance Curves of ASTM A106 Class C steel and 3-Ni steel. Compact specimens (1T) were tested using a multi-specimen technique, direct current potential drop technique and the elastic unloading compliance technique with unloading ranging from 10 to 90%. The two former techniques were 0% unloading procedures used to generate the reference J-R curves for comparison to the elastic unloading J-R curves for the two steels. The results of the investigations of these materials indicate that there was no significant difference in the J-R curves that resulted from the elastic unloading compliance technique.

NUREG/CR-4284: NEUTRON EXPOSURE PARAMETERS FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY IRRADIATION SERIES. STALLMANN,F.W.; KAM,F.B.; BALDWIN,C.A. Oak Ridge National Laboratory. August 30, 1985. 39pp. 8509110035. ORNL/TM-9664. 32564:337.

The Nuclear Regulatory Commission's (NRC's) Heavy Section Steel Technology (HSST) Program is concerned with the investigation of cracklike flaws in reactor pressure vessel steels. In the fifth irradiation series, capsules containing a variety of metallurgical test specimens were irradiated to fluences in the range of 1.10(19) to 3.10(19) nuetrons/cm(2) (E > 1.0 MeV). In order to correlate radiation embrittlement to damage fluences, accurate determination of the neutron fluence spectra at the critical location of the test specimen is needed. The part of the neutron spectrum which is responsible for the radiation damage is characterized as "damage exposure parameter." Fluences for energies greater than 1.0 MeV (F > 1.0 MeV) is the most widely used parameter; however, current thinking favors displacements per atom (dpa) in iron as better related to the physical mechanism of radiation damage. Fluences for energies greater than 0.1 MeV (F > 0.1 MeV) are also considered since neutrons in the 0.1 to 1.0 MeV range are likely to contribute to the damage. In order not to prejudice future investigations, all three damage parameters F > 1.0 MeV, F > 0.1 MeV, and dpa will be listed in this report. This NUREG will contain the neutron exposure parameters for the 12 metallurgical specimen capsules which comprise the Fifth HSST Irradiation Series. In order to make available the data in a timely manner and not to delay the analysis of capsules irradiated early in the series, it was decided to make this NUREG a loose-leaf document. The exposure values will be distributed as they become available.

NUREG/CR-4287: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Annual Report,October 1983 - September 1984. SHACK,W.J.; KASSNER,T.F.; MAIYA,P.S.; et al. Argonne National Laboratory. August 1985. 149pp. 8508210430. ANL-85-33. 32338:265.

This progress report summarizes work performed by the Argonne National Laboratory and a subcontractor, E.F. Rybicki, Inc., on environmentally assisted cracking in light water reactors during the twelve months from October 1983 through September 1984.

NUREG/CR-4288: FOCAL MECHANISM ANALYSES FOR VIRGINIA AND EASTERN TENNESSEE EARTHQUAKES (1978-1984). BILLINGER,G.A.; TEAGUE,A.G.; MUNSEY,J.W.; et al. Virginia Polytechnic Institute & State Univ., Blacksburg, VA. June 1985. 91pp. 8508010243. 31928:001.

Focal mechanisms are presented for 11 earthquakes from the Giles County, Virginia, seismic zone and its vicinity and for 12 eathquakes from the Central Virginial seismic zone. These earthquakes ($0 < M < 4$) were monitored by local networks between January 1978 and October 1984. In Giles County, the data base consists of 43 P-wave polarities and 50 SB to P amplitude ratios (SV/P) that yielded six single event focal mechanims (SEFM's) and five composite event focal mechanisms (CFM's). In Central Virginia, 79 P-wave polarities and 51 SV/P ratios are used to determine 11 SEFM's and four CFM's. A computer program FOCMEC was used to determine the focal mechanism solutions. The results for the Giles County seismic zone show mainly strike-slip mechanisms on steeply dipping (73 degrees plus minus 16 degrees) NNE (right lateral motion) and ESE (left lateral motion) trending nodal planes. However, some (4/11) counterclockwise. The P axes in Central Virginia are generally northeast trending for shallow earthquakes (>8 km) and northwest trending for deeper ones (<8 km). In Giles County, where the seismic activity is occurring beneath the Appalachian decollement, faulting and inferred stress orientations are more uniform than in Central Virginia, some 200 km away, where the seismicity is occurring near and above the decollement.

NUREG/CR-4290 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF BABCOCK AND WILCOX PWR PLANTS.Volume 2:Guillotine Break Indirectly Induced By Earthquakes. RAVINDRA,M.K.; CAMPBELL,R.D.; KIPP,T.R.; et al. Lawrence Livermore National Laboratory. July 1985. 146pp. 8508010757. UCRL-53644. 31924:001.

The requirements to design nuclear power plants for the effects of an instantaneous double-ended guillotine break (DEGB) of the reactor coolant loop (RCL) piping have led to excessive design costs, interference with normal plant operation and maintenance, and unnecessary radiation exposure of plant maintenance personnel. This report describes an aspect of the NRC/Lawrence Livermore National Laboratory sponsored research program aimed at demonstrating that the probability of DEGB in RCL Piping of nuclear power plants is acceptably small and the requirements to design for the DEGB effects (e.g., provision of pipe whip restraints) may be removed. This study estimates the probability within the containment of Babcock & Wilcox supplied pressurized water reactor nuclear power plants in the United States. The medium probability of indirect DEGB was estimated to range between 6x10(-11) and 1x10(-7) per year. Using very conservative assumptions, the 90% subjective probability value (confidence) of P(DEGB) was found to be less than 1x10(-5) per year.

NUREG/CR-4291: CONCLUSION AND SUMMARY REPORT ON PHYSICAL BENCHMARKING OF PIPING SYSTEMS. BEZLER,P.; SUBUDHI,M.; SHTEYNGART,S.; et al. Brookhaven National Laboratory. September 1985. 105pp. 8509300512. BNL-NUREG-51897. 32792:322.

Physical benchmark evaluations were used to assess the accuracy and adequacy of the analysis methods and assumptions used in typical piping qualification evaluations. Physical benchmark evaluations have been completed for six systems involving both laboratory and in situ tested piping. In each evaluation elastic finite element methods were used to predict the time history response of a system for which physical test results were available. In the analytical simulations the measured support excitations and the measured damping properties were used as input and the acceleration and displacement response of piping interior points were predicted as output. The linear analysis methods were found to provide reasonable estimates of system response. For a near linear system and using conservative estimates for system damping, a good correlation of response traces and acceptable estimates of response peaks can be expected. Using realistic estimates of uniform system damping, large underestimates of peak response components were observed and deviations of 100% or greater should be expected.

NUREG/CR-4292: A COMPARATIVE ANALYSIS OF CONSTITUTIVE RELATIONS IN TRAC-PFL AND RELAP5/MOD1. ROHATGI,U.S.; JO,J.H.; SLOVIK,G.C. Brookhaven National Laboratory. June 1985. 104pp. 8601070499. BNL-NUREG-51898. 34183:279.

The purpose of this document is to describe the basic thermal hydraulic models and correlations that were used in the TRAC-PF1 (Version 7.0) and RELAP5/MOD1/CYCLE-14 codes. Concerted efforts have also been made to assess the models described in the code manuals and to compare them with their FORTRAN versions in the code. Some discrepancies between the documentation and the code, some errors in the models, and variety of constraints on the models were found and have been reported here. Comments based on BNL experience with TRAC-PF1 and RELAP5/MOD1 assessment have been made. The text contains many FORTRAN variables in order to help the readers who might be interested in modifying these codes. A table comparing the constitutive relationships in these two codes is also presented.

NUREG/CR-4294: LEAK RATE ANALYSIS OF THE WESTINGHOUSE REACTOR COOLANT PUMP. BOARDMAN,T.; JEANMOUGIN,N.; LOFARO,R.; et al. Rockwell International Corp. July 1985. 66pp. 8508020424. 85-ETEC-DRF-171. 31961:006.

An independent analysis was performed to determine seal leakage rates for the Westinghouse Reactor Coolant Pump (RCP) during a postulated station blackout resulting from loss of ac electric power. The analysis confirmed Westinghouse calculations on RCP seals performance for the three conditions investigated: (1) all three seals function, (2) No. 1 seal fails open while Nos. 2 and 3 seals function, and (3) all three seals fail open.

NUREG/CR-4297: EXTREMITY MONITORING:Considerations For Use,Dosimeter Placement,And Evaluation. REECE,W.D.; HARTY,R.; BRACKENBUSH,L.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. December 1985. 110pp. 8601070502. PNL-5509. 34155:250.

Various aspects of extremity dosimetry are presented in this paper to help the licensee decide when to use extremity dosimetry, what method of dosimetry to use, and how to interpret the results of the dosimetry system. The current regulations as they apply to extremity dosimetry and the Nuclear Regulatory Commission's interpretation of these regulations are reviewed. The history of the current Code of Federal Regulations is examined so that the reasoning behind the present regulations may be understood. The recommendations of various radiation advisory groups are also summarized and a critical review of published experimental data applicable to extremity dosimetry is presented. These recommendations and data are used to identify the radiosensitive tissues in the extremities and rank the relative risk to each organ or tissue. The results of laboratory and field measurements of extremity dose to personnel engaged in work typical of different classes of licensees are presented. The licensee classes include fuel fabrication facilities, pressurized water reactors, boiling water reactors, medical radioisotopes operations, and well-logging operations. Finally, recommendations and guidelines for several specific situations are provided to help assist the practicing health physicist with extremity dosimetry.

NUREG/CR-4298: DESIGN AND INSTALLATION OF COMPUTER SYSTEMS TO MEET THE REQUIREMENTS OF 10 CFR 73.55. LEWIS,J.R.; BYERS,K.R.; FLUCKIGER,J.D.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1985. 120pp. 8507250138. PNL-5490. 31794:221.

The Pacific Northwest Laboratory has studied the design and installation of computer-managed systems that can help nuclear power plant licensees to meet the physical security requirements of 10 CFR 73.55 (for access control, alarm monitoring, and alarm recording.) Two objectives were to study the power plant security functions that could be aided by a computer-managed physical security system and to evaluate the safety and security considerations of such a system. A further objective was to develop guidance on system design, selection, and installation. The design guidance includes safety and security requirements, design alternatives, computer security, workspace design and user interface design. Guidance is also provided on writing a system specification for procurement, bid review procedures, and site preparation.

NUREG/CR-4300 V01: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Progress Report, October-March 1985. HUTTON,P.H.; KURTZ,R.J. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1985. 27pp. 8508210022. PNL-5511. 32302:232.

Technical progress in developing continuous acoustic emission (AE) monitoring of nuclear reactor pressure boundaries for flaw detection is discussed in this report. The period covered is October 1, 1984, to April 1, 1985. Topics include final analysis of ZB-1 vessel test data, preparation for continuous AE monitoring of Watts Bar Unit 1 reactor during operation, AE signal pattern recognition development, and development of an ASTM standard for application of continuous AE monitoring to pressure boundaries.

NUREG/CR-4303: HIGH-LEVEL WASTE PRECLOSURE SYSTEMS SAFETY ANALYSIS.Phase 1, Final Report. HARRIS,P.A.; LIGON,D.M.; STAMATELATOS,M.; et al. GA Technologies, Inc./General Atomic Co. September 1985. 330pp. 8509300521. SAND85-7192. 32810:001.

The major effort for this project has been on the gathering, organizing, and assembling of information pertinent to the safety assessment of a nuclear waste repository during preclosure operations. Specific issues addressed in this report are: 1. Detailed analysis of a conceptual basalt repository design in order to identify potential initiating event/accident scenarios capable of causing radiological and/or nonradiological consequences. 2. Evaluation of radiological and nonradiological consequences relevant to a nuclear repository and recommendation of an approach for quantitative evaluation of these consequences. 3. Comparative evaluation of several importance ranking measures that had been used in the nuclear industry in order to select a measure to best meet the needs of the program. 4. Development of event and fault tree models for those initiating events which have passed the preliminary screening process. 5. Compilation of specific data such as initiating event frequencies, component/system failure rates and repair times, personnel injury, and basic information necessary for more detailed radiological consequence evaluations at a later time. 6. Selection of a set of accident scenarios to be quantified in the next study phase to demonstrate the applicability of the proposed methodology that will identify and quantitatively prioritize structures, components, systems, and operations which are important to safety during the preclosure phase of a HLW rep.

NUREG/CR-4304: PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO THE PWR THERMAL-SHOCK ISSUE.Experiment TSE-7. CHEVERTON,R.D.; BALL,D.G.; BOLT,S.E.; et al. Oak Ridge National Laboratory. September 1985. 152pp. 8510040402. GSCA-46. 32859:050.

Thermal-shock experiment TSE-7 was conducted for the purpose of investigating the behavior of surface flaws under pressurized-water-reactor (PWR) overcooling-accident conditions. This experiment was the eighth in a series of thermal-shock experiments conducted for this purpose with large steel cylinders (A508, class-2 chemistry; 991-mm outside diameter x 152-mm wall x 1.2-m length) as a part of the Heavy-Section Steel Technology (HSST) Program. The initial flaw for TSE-7 was a shallow, semielliptical, inner-surface, axially oriented, sharp crack located at midlength of the test cylinder. The thermal shock was applied to the inner surface only, and this was accomplished by effectively dunking the test cylinder, initially at equivalent 93 degrees centigrade, into a large volume of liquid nitrogen. The specific purpose of TSE-7 was to determine whether, in agreement with analysis, a short and shallow surface flaw, in the absence of cladding, would extend on the surface to effectively become a very long flaw as a result of severe thermal-shock loading. During the experiment, there were three major initiation-arrest events. The first event consisted of some radial propagation and very extensive surface extension, with many bifurcations taking place. The second and third events consisted primarily of radial propagation. A fourth initiation event was prevented by warm prestressing. These results were in good agreement with predictions.

NUREG/CR-4305: COMMENTS ON THE LEAK-BEFORE-BREAK CONCEPT FOR NUCLEAR POWER PLANT PIPING SYSTEMS. RODABAUGH,E.C. E.C. Rodabaugh Associates, Inc. * Oak Ridge National Laboratory. September 1985. 58pp. 8510030312. 32854:257.

Leak-before-break entails the concept that, with a high degree of probability, failure of the pressure boundary of piping systems will be signaled by a detectable leak which will provide ample time to shut down and repair that leak. The status of the leak-before-break concept is discussed in this report, including a review of industrial and nuclear power plant experience with respect to leak-before-break, fracture mechanics and potential elimination of postulated pipe breaks in nuclear power plant piping design.

NUREG/CR-4314: BRIEF SURVEY AND COMPARISON OF COMMON CAUSE FAILURE ANALYSIS. WALLER,R.A. Los Alamos Scientific Laboratory. August 1985. 28pp. 8509130416. LA-10474-MS. 32605:035.

This paper presents a brief survey of methods and a list of references for analyzing common cause (mode) failure. Implicit models, explicit modeling techniques, and computer aids are included in the discussion. It is suggested that although current trends are emphasizing development of explicit models, a realistic assessment of data availability will force continued use of implicit or hybrid models in the immediate future.

NUREG/CR-4317 V01: CANADIAN SEISMIC AGREEMENT.Technical Report Covering 1979-1985. HAYMAN,R.B.; BASHAM,P.W.; WETTMILLER,R.J.; et al. Canada, Govt. of. July 1985. 71pp. 8508090583. 32106:001.

This Final Report provides a comprehensive summary of the activities of the Earth Physics Branch (EPB) in eastern Canada, particularly with respect to the Eastern Canadian Telemetered Network (ECTN). The report describes the seismographic system developed by EPB to monitor the regional seismicity. There are detailed descriptions of the logic used to select the various physical components of the system, the hardware and software that constitute the recording system, and the data flow from detection to archive. In addition to the engineering details, there is a discussion of the scientific results from the analysis of the regional seismicity during the reporting period. Particular emphasis has been placed on the Miramichi, New Brunswick earthquakes of January 1982.

NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,January-March 1985. EDLER,S.K. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1985. 27pp. 8509100359. PNL-5516-1. 32528:024.

This document summarizes work performed by Pacific Northwest Laboratory from January 1 through March 31, 1985, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. PNL is operated for the U.S. Department of Energy by Battelle Memorial Institute under Contract DE-AC06-76RLO 1830. Results from an instrumented fuel assembly irradiation program being performed at Halden, Norway, are reported. 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. Fuel assemblies and analytical support are being provided for experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Thermal-hydraulic computer programs are providing best-estimate analyses for a variety of safety issues in light-water reactors. Severe fuel damage tests are being conducted in the NRU Reactor, Chalk River, Canada.

NUREG/CR-4318 V02: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,April-June 1985. EDLER,S.K. Battelle Memorial Institute, Pacific Northwest Laboratories. October 1985. 25pp. 8511210266. PNL-5516-2. 33558:070.

This document summarizes work performed by Pacific Northwest Laboratory (PNL) from April 1 through June 30, 1985, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. PNL is operated for the U.S. Department of Energy by Battelle Memorial Institute under Contract DE-AC06-76RLO 1830. Results from an instrumented fuel assembly irradiation program being performed at Halden, Norway, are reported. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Thermal-hydraulic computer programs are providing best-estimate analyses for a variety of safety issues in light-water reactors. Severe fuel damage tests are being conducted in the National Research Universal (NRU) Reactor, Chalk River, Canada.

NUREG/CR-4321: FULL SCALE MEASUREMENTS OF SMOKE TRANSPORT AND DEPOSITION IN VENTILATION SYSTEM DUCTWORK. MARTIN,R.A.; FENTON,D.L. Los Alamos Scientific Laboratory. July 1985. 39pp. 8511010254. LA-10478-MS. 33303:257.

This study is part of an effort to obtain experimental data in support of the fire accident analysis computer code FIRAC, which was developed at the Los Alamos National Laboratory. FIRAC can predict the transient movement of aerosolized or gaseous material throughout the complex ventilation systems of nuclear fuel cycle facilities. We conducted a preliminary set of full-scale material depletion/modification experiments to help assess the accuracy of the code's aerosol depletion model. Such tests were performed under realistic conditions using real combustion products in full-sized ducts at typical air-flow rates. To produce a combustion aerosol, we burned both polystyrene and polymethyl methacrylate, the most and least smoky fuels typically found in fuel cycle plants, under varied ventilation (oxygen-lean and oxygen-rich) conditions. Aerosol mass deposition, size, and concentration measurements were performed. We found that as much as 25% of polystyrene smoke mass and as little as 2% of the polymethly methacrylate generated at the entrance to a 15.2-m duct is deposited on the duct walls. We also compared our experimental results with theoretical equations currently used in FIRAC.

NUREG/CR-4322 V01: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 1: Enterprise Model. * Touche Ross & Co. November 1985. 400pp. 8601070529. 34190:222.

The NRC has initiated a multi-year program to centralize its information processing in a Corporate Data Network (CDN). The information processing environment will include shared databases, telecommunications, office automation tools, and state-of-the-art software. Touche Ross and Company was contracted to perform a general data requirements analysis for shared databases and to develop a preliminary plan for implementation of the CDN concept. The ENTERPRISE MODEL (Vol. 1) provided the NRC with agency-wide information requirements in the form of data entities and organizational demand patterns as the basis for clustering the entities into logical groups. The DATA DICTIONARY (Vol. 2) provided the NRC with definitions and example attributes and properties for each entity. The DATA MODEL (VOL. 3) defined logical databases and entity relationships within and between databases. The PRELIMINARY STRATEGIC DATA PLAN (Vol. 4) prioritized the development of databases and included a workplan and approach for implementation of the shared database component of the Corporate Data Network.

NUREG/CR-4322 V02: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 2: Data Entity Dictionary. * Touche Ross & Co. November 1985. 100pp. 8601070524. 34198:031.

See NUREG/CR-4322,V01 abstract.

NUREG/CR-4322 V03: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 3: Data Model. * Touche Ross & Co. November 1985. 100pp. 8601070528. 34188:143.

See NUREG/CR-4322,V01 abstract.

NUREG/CR-4322 V04: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 4: Preliminary Strategic Data Plan. * Touche Ross & Co. November 1985. 100pp. 8601070517. 34188:339.

See NUREG/CR-4322,V01 abstract.

NUREG/CR-4325: A PARAMETRIC STUDY OF PWR PRESSURE VESSEL INTEGRITY DURING OVERCOOLING ACCIDENTS,CONSIDERING BOTH 2-D AND 3-D FLAWS. CHEVERTON,R.D.; BALL,D.G. Oak Ridge National Laboratory. September 1985. 40pp. 8510040371. ORNL/TM-9682. 32860:335.

A continuing analysis of the pressurized water reactor pressurized thermal-shock problem indicates that the previously accepted degree of conservatism in the fracture-mechanics model needs to be more closely evaluated and, if excessive, reduced. One feature that was believed to be conservative was the use of two-dimensional as opposed to finite-length flaws. The degree of conservatism could not be adequately investigated because of computational limitations and a lack of knowledge regarding flaw behavior; however, that situation has changed to the extent that some cases involving finite-length flaws can be studied. A flaw of particular interest is one that is located in an axial weld of a plate-type vessel. For those vessels that suffer relatively high radiation damage in the welds, the length of the flaw will be no greater than the length of the weld, and recent calculations indicate that a deep flaw of that length (is equivalent to 2m) is not effectively infinitely long, contrary to previous thinking. The benefit to be derived from consideration of the 2-m flaw and also a semielliptical flaw with a length-to-depth ratio of 6/1 was investigated by analyzing several postulated transients. In doing so the sensitivity of the benefit to a specified maximum of the analysis indicate that for some conditions the benefit in using the 2-m flaw is substantial, but it decreases with increasing pressure, and above a certain pressure there may be no benefit, depending on the duration of the transient and the limit on crack-arrest toughness.

NUREG/CR-4326 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Main Report. BRUSKE,S.J.; DAVIS,C.B.; OGDEN,D.M.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). August 1985. 162pp. 8509100428. EGG-2405. 32530:093.

This report documents the evaluation of the effects of nonsafety grade control system failures on a typical 3-loop Westinghouse pressurized water reactor plant. The methods utilized for this evaluation include a system level failure modes and effects analysis, deterministic computer analysis (utilizing a plant model that includes the nuclear steam supply system, balance of plant systems and control systems), a review of plant occurrences, a probability analysis and a review of applicable Nuclear Regulatory Commission (NRC) criteria pertaining to control systems. This study identified two system failures that could cause transients leading to a steam generator overfill and two system failures that could lead to a reactor coolant cooldown of greater than 100 degrees fahrenheit per hour. It also identified two system failures that could lead to an overpressurization of low temperatures and two steam generator tube rupture events that could be further aggravated by additional system failures. This study concluded that the existing NRC criteria, concerning control systems, adequately addressed the potential problem areas that were identified during this evaluation. Based on the results of this study, it is recommended that the consequences and risk associated with overfill and overcool transients be further investigated. It is also recommended that the probabilities associated with the low temperature overpressurization and the steam generator tube rupture sequences be evaluated by the NRC staff. The results of this study will be factored together with other studies being performed on the effects of control system failures to establish a position for resolution of Unresolved Safety Issue A-47 (Safety Implications of Control Systems).

NUREG/CR-4326 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Appendices. BRUSKE,S.J.; DAVIS,C.B.; OGDEN,D.M.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). October 1985. 233pp. 8511040050. EGG-2405. 33337:327.

This report documents the evaluation of the effects of nonsafety grade control system failures on a typical 3-loop Westinghouse pressurized water reactor plant. The methods utilized for this evaluation include a system level failure modes and effects analysis, deterministic computer analysis (utilizing a plant model that includes the nuclear steam supply system, balance of plant systems and control systems), a review of plant occurrences, a probability analysis and a review of applicable Nuclear Regulatory Commission (NRC) criteria pertaining to control systems. This study identified two system failures that could cause transients leading to a steam generator overfill and two system failures that could lead to a reactor coolant cooldown of greater than 100 degrees fahrenheit per hour. It also identified two system failures that could lead to an overpressurization at low temperatures and two steam generator tube rupture events that could be further aggravated by additional system failures. This study concluded that the existng NRC criteria, concerning control systems, and adequately address the potential problem areas that were identified during this evaluation. Based on the results of this study, it is recommended that the consequences and risk associated with overfill and overcool transients be further investigated. It is also recommended that the probabilities associated with the low temperature overpressurization and the steam generator tube rupture sequences be evaluated by the NRC staff. The results of this study will be factored together with other studies being performed on the effects of control system failures to establish a position for resolution of Unresolved Safety Issue A-47 (Safety Implications of Control Systems).

NUREG/CR-4329: RELIABILITY EVALUATION OF CONTAINMENTS INCLUDING SOIL-STRUCTURE INTERACTION. PIRES,J.; HWANG,H.; REICH,M. Brookhaven National Laboratory. December 1985. 128pp. 8601070439. BNL-NUREG-51906. 34182:290.

Soil-structure interaction effects on the reliability assessment of containment structures are examined. The probability-based method for reliability evaluation of nuclear structures developed at Brookhaven National Laboratory is extended to include soil-structure interaction effects. In this method, reliability of structures is expressed in terms of limit state probabilities. Futhermore, random vibration theory is utilized to calculate limit state probabilities under random seismic loads. Earthquake ground motion is modeled by a segment of a zero-mean, stationary, filtered Gaussian white noise random process, represented by its power spectrum. All possible seismic hazards at a site, represented by a hazard curve, are also included in the analysis. The soil-foundation system is represented by a rigid surface foundation on an elastic halfspace. Random and other uncertainties in the strength properties of the structure, in the stiffness and internal damping of the soil, are also included in the analysis. Finally, a realistic reinforced concrete containment is analyzed to demonstrate the application of the method. For this containment, the soil-structure interaction effects on: (1) limit state probabilities, (2) structural fragility curves, (3) floor response spectra with probabilistic content, and (4) correlation coefficients for total acceleration response at specified structural locations, are examined in detail.

NUREG/CR-4331: SIMPLIFIED SEISMIC PROBABILISTIC RISK ASSESSMENT:Procedures And Limitations. SHIEH,L.C.; JOHNSON,J.J.; WELLS,J.E.; et al. Lawrence Livermore National Laboratory. August 1985. 198pp. 8508220290. UCID-20468. 32345:320.

At the request of the U.S. Nuclear Regulatory Commission, the Lawrence Livermore National Laboratory has developed a simplified seismic probabilistic risk assessment (PRA) methodology. The purpose of this methodology is to reduce the costs while adequately performing seismic probabilistic risk assessments of nuclear power plants. This report summarizes the development of the simplified methodology and explains guidelines for applying the procedures. The development effort is part of the scope of work of the Seismic Safety Margins Research Program (SSMRP).

NUREG/CR-4333: STE. GENEVIEVE FAULT ZONE,MISSOURI AND ILLINOIS. NELSON,W.J.; LUMM,D.K. Illinois, State of. July 1985. 104pp. 8508090574. 32103:255.

The Ste. Genevieve Fault Zone is a major structural feature which strikes NW-SE for about 190 km on the NE flank of the Ozark Dome. There is up to 900 m of vertical displacement on high angle normal and reverse faults in the fault zone. At both ends the Ste. Genevieve Fault Zone dies out into a monocline. Two periods of faulting occurred. The first was in late Middle Devonian time and the second from latest Mississippian through early Pennsylvanian time, with possible minor post-Pennsylvanian movement. No evidence was found to support the hypothesis that the Ste. Genevieve Fault Zone is part of a northwestward extension of the late Precambrian-early Cambrian Reelfoot Rift The magnetic and gravity anomalies cited in support of the "St. Louis arm" of the Reelfoot Rift possibly reflect deep crustal features underlying and older than the volcanic terrain of the St. Francois Mountains (1.2 to 1.5 billion years old). In regard to neotectonics no displacements of Quaternary sediments have been detected, but small earthquakes occur from time to time along the Ste. Genevieve Fault Zone. Many faults in the zone appear capable of slipping under the current stress regime of east-northeast to west-southwest horizontal compression. We conclude that the zone may continue to experience small earth movements, but catastrophic quakes similar to those at New Madrid in 1811-12 are unlikely.

NUREG/CR-4334: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS. BUDNITZ,R.J.; AMICO,P.J.; CORNELL,C.A.; et al. Lawrence Livermore National Laboratory. August 1985. 307pp. 8508260301. UCID-20444. 32369:001.

This report is the second report of the Expert Panel on Quantification of Seismic Margins. The Panel's first report was entitled, "NRC Seismic Design Margins Program Plan." The objective of this report is to discuss progress to date in studying the issue of quantification of seismic margins in nuclear power plants. In particular, it deals with progress towards the establishment of review guidelines that would be useful in studying how much seismic margin exists. The guidelines themselves will be the subject of the next Panel report. The work presented in this report is the result of a detailed study of seismic Probabilistic Risk Assesssments, historical earthquake performance of the nuclear and non-nuclear facilities, and test data, augmented by the individual experience and expertise of the Panel members. The major development discussed in this report is the HCLPF concept, which demonstrates margin by showing that there is a High Confidence of a Low Probability of Failure for a given earthquake size.

NUREG/CR-4335: POTENTIAL BENEFITS OBTAINED BY REQUIRING SAFTEY-GRADE COLD SHUTDOWN SYSTEMS. GALLUP,D.R.; KUNSMAN,D.M.; BOHN,M.P. Sandia National Laboratories. November 1985. 127pp. 8512030665. SAND85-1339. 33734:170.

This work investigates 2 items of concern to pressurized water reactors: the potential benefits that may be obtained by requiring safety-grade cold shutdown systems, and the best suction valve arrangements for residual heat removal systems. The approach taken is to perform a case study of a reference power plant and to extrapolate the results obtained to other PWRs. Our analysis concludes that the core melt frequency at the reference plant can be reduced by an estimated 7.1E-6 per year by requiring safety-grade cold shutdown systems. Most of this reduction can be attributed to making the systems Seismic Category I. For the accident scenarios investigated in this work, adding redundancy to the cold shutdown systems does not significantly reduce the core melt frequency. Further, the ability of PWRs to reach cold shutdown using the auxiliary feedwater system and atmospheric dump valves must be investigated on a plant-by-plant basis. The ADVs at some plants may be sized too small to allow the plant to reach cold shutdown. Finally, we believe that the "best" RHR suction valve arrangement is to have a single suction line without primary system overpressure interlocks on the valves.

NUREG/CR-4339: A REVIEW OF RECENT RESEARCH ON THE SEISMOTECTONICS OF THE SOUTHEASTERN SEABOARD AND AN EVALUATION OF HYPOTHESES ON THE SOURCE OF THE 1886 CHARLESTON,SOUTH CAROLINA EARTHQUAKE. DEWEY,J.W. Interior, Dept. of, Geological Survey. August 1985. 45pp. 8508290532. 32410:197.

In spite of extensive research on the source of the 1886 Charleston, S.C. Earthquake, there is not yet a consensus among earth scientists on the characteristics of the fault that produced the earthquake or on the likelihood of future large earthquakes at other locations of the Southeastern Seaboard. This report reviews the evidence from recent research on three categories of hypothesis: (A) hypotheses on the specific geologic structures that might cause large earthquakes in the Southeastern Seaboard; (B) hypotheses on the seismotectonic zones in which large earthquakes might occur; and (C) hypotheses on temporal variations of seismicity in the Southeastern Seaboard. Hypotheses that are representative of each category are summarized, and evidence for and against each hypothesis is given, if such evidence is available. When data are interpreted in the ways that currently seem to be the most straightforward, the hypotheses that are supported by one kind of evidence are usually opposed by another kind of evidence. Reaching a consensus on the cause of the Charleston earthquake, and on the likelihood of such an earthquake occurring at other locations of the Southeastern Seaboard, will therefore probably require the reconciliation of what currently appear to be contrary pieces of evidence.

NUREG/CR-4340 V01: REACTOR SAFETY RESEARCH SEMI-ANNUAL REPORT.January - June 1985. * Sandia National Laboratories. October 1985. 344pp. 8512270343. SAND85-1606. 34082:117.

Sandia National Laboratories is conducting, under the USNRC's sponsorship, phenomenological research related to the safety of commercial nuclear power reactors. The research includes experiments to simulate the phenomenologu of the accident conditions and the development of analytical models, verified by experiment, which can be used to predict reactor and safety systems performance and behavior under abnormal conditions. The objective of this work is to provide NRC requisite data bases and analytical methods to (1) identify and define safety issues, (2) understand the progression of risk-significant accident sequences, and (3) conduct safety assessments. The collective NRC-sponsored effort at Sandia National Laboratories is directed at enhancing the technology base supporting licensing decisions.

NUREG/CR-4342: UNCERTAINTY AND SENSITIVITY ANALYSIS OF A MODEL FOR MULTICOMPONENT AEROSOL DYNAMICS. HELTON,J.C.; IMAN,R.L.; JOHNSON,J.D.; et al. Sandia National Laboratories. September 1985. 59pp. 8511070252. SAND84-1307. 33380:197.

An uncertainty and sensitivity analysis of the MAEROS model for multicomponent aerosol dynamics is presented. Analysis techniques based on Latin hypercube sampling and regression analysis are used to study the behavior of a two component aerosol in a nuclear power plant containment for a transient accident with loss of AC power (i.e., a TMLB' accident). Conditional on assumed ranges and distributions for selected independent variables (e.g., initial distributions and mass loadings for each component, termperature, pressure, shape factors), estimates are made for distributions of model predictions and for the independent variables which influence these predictions. The analysis indicated that, for the situation under consideration, variables related to agglomeration (e.g., dynamic shape factor, material density, agglomeration shape factor, and turbulence dissipation rate) tended to dominate the observed variability. For comparison, an analysis based on differential techniques is also given. Further, a study of the effects on MAEROS predictions due to the number of particle size classes and the particle size class boundaries is presented. This analysis was performed as part of a project to develop a new system of computer codes (i.e., the MELCOR code system) for use in risk assessments for nuclear power plants.

NUREG/CR-4344: INSTRUCTIONAL SKILLS EVALUATION IN NUCLEAR INDUSTRY TRAINING. MAZOUR,T.J.; BALL,F.M. Analysis & Technology, Inc. November 1985. 136pp. 8512300185. 34094:052.

This report provides information to nuclear power plant training managers and their staffs concerning the job performance requirements of instructional personnel to implement performance-based training programs (also referred to as the Systems Approach to Training.) The information presented in this report is a compilation of information and lessons learned in the nuclear power industry and in other industries using performance-based training programs. The job performance requirements in this report are presented as instructional skills objectives. The process used to develop the instructional skills objectives is described. Each objective includes an Instructional Skills Statement describing the behavior that is expected and an Instructional Skills Standard describing the skills/knowledge that the individual should possess in order to have achieved mastery. The instructional skills objectives are organized according to the essential elements of the Systems Approach to Training and are cross-referenced to three categories of instructional personnel: developers of instruction, instructors, and instructional managers/supervisors. Use of the instructional skills objectives is demonstrated for reviewing instructional staff training and qualification programs, developing criterion-tests, and reviewing the performance and work products of individual staff members.

NUREG/CR-4345: INVESTIGATION OF THE STABILITY OF LWR SPENT FUEL RODS BELOW 250 C. OLSEN,C.S. EG&G Idaho, Inc. (subs. of EG&G, Inc.). September 1985. 153pp. 8511010394. EGG-2409. 33335:071.

A testing program was conducted to investigate the long-term stability of commercial pressurized-water-reactor (PWR) and boiling-water-reactor (BWR) spent fuel rods under a variety of possible dry storage conditions. The objective of this project is to provide the Nuclear Regulatory Commission (NRC) with an experimental bass to evaluate the results of short-term, high-temperature tests and to establish a licensing position for long-term, low-temperature (less than 250 degrees centigrade) spent fuel rod dry storage. A total of nine fuel rods (five BWR and four PWR) were nondestructively examined after interim heating periods, and seven of these fuel rods were destructively examined to determine the degradation of the fuel rods during the long-term, low-temperature dry fuel storage. The results of the non-destructive examinations of all nine fuel rods and the destructive examination of one fuel rod were reported previously. This report presents detailed results of the destructive examinations of six of the remaining fuel rods, with brief descriptions of the examinations of all the test fuel rods as they pertain to the fuel rod behavior during dry spent fuel storage conditions.

NUREG/CR-4346: AEROSOL RELEASE EXPERIMENTS IN THE FUEL AEROSOL SIMULANT TEST FACILITY:UNDERSODIUM EXPERIMENTS. PETRYKOWSKI,J.; LONGEST,A.W.; ROCHELLE,J.M.; et al. Oak Ridge National Laboratory. September 1985. 56pp. 8511070242. ORNL/TM-9479. 33383:186.

The release of UO(2) aerosols from pools of sodium was studied in a series of ten experiments in the Fuel Aerosol Simulant Test (FAST) facility at Oak Ridge National Laboratory. The experiments were designed to provide a mechanistic basis for evalating the radiological source term associated with a postulated, hypothetical core disruptive accident (HCDA) in a liquid metal fast breeder reactor (LMFBR). Aerosol was generated by capacitor discharge vaporization of UO(2) pellets which were submerged in a sodium pool under an argon cover gas. Measurements of the pool and cover gas pressure were used to study the transport of aerosol contained by vapor bubbles within the pool. Cover gas samples were filtered to determine the quantity of aerosol released from the pool. Trace amounts of UO(2) aerosol (<0.3% of the total pellet mass) were detected in the cover gas samples suggesting that the bulk of aerosol was trapped within bubbles confined by the pool. The report contains (1) a description of the experiments, (2) data records for the pool pressure, the cover gas pressure, and the UO(2) aerosol concentration in the cover gas and, (3) an analysis of the experimental findings using simplified models of bubble behavior.

NUREG/CR-4347: EMERGENCY DIESEL GENERATOR OPERATING EXPERIENCE,1981-1983. BATTLE,R.E. Oak Ridge National Laboratory. December 1985. 80pp. 8601070491. ORNL/TM-9739. 34198:155.

The purpose of this report is to update the operating experience of emergency diesel generators in nuclear power plants. Previously, similar data for 1976 through 1980 were reported in NUREG/CR-2989, "Reliability of Emergency AC Power Systems at Nuclear Power Plants." The two data sets are used to show trends of diesel generator performance, and responses by nuclear plant licenses to a nuclear regulatory questionnaire is included for additional data and comparison with the Licensee Event Report (LER) data collected for this report. The LER database was used to collect diesel generator failures, and the databases for diesel generator successes were from nuclear plant licensees' responses to NRC questionnaires. Estimates of diesel generator failure on demand were calculated from the diesel generators test data, from data reported in response to an NRC questionnaire (generic letter 84-15), from diesel generator performance during complete and partial losses of off-site power, and from diesel generator performance for safety injection actuation signals.

NUREG/CR-4350 V01: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 1 - PRA Fundamentals. * Sandia National Laboratories. BREEDING,R.J.; LEAHY,T.J.; et al. Energy, Inc. August 1985. 425pp. 8512270366. SAND85-1495. 34083:101.

The full range of PRA topics are presented, with special emphasis on systems analysis and PRA applications. Systems analysis topics include system modeling such as fault tree and event tree construction, failure rate data, and human reliability. The discussion of PRA applications is centered on past and present PRA-based programs such as WASH-1400 and the Interim Reliability Evaluation Program, as well as on some of the potential future applications of PRA. The relationship of PRA to generic safety issues such as station blackout and Anticipated Transient Without Scram (ATWS) is also discussed. In addition to system modeling the major PRA tasks of accident process analysis, and consequence analysis are presented. An explanation of the results of these activities and the techniques by which these results are derived forms the basis for a discussion of these topics. An additional topic presented in this course is the topic of PRA management, organization, and evaluation. This discussion explains the relationship of sound managment, proper organization, and thorough evaluation to the performance of credible risk assessment.

NUREG/CR-4350 V02: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 2-Probability And Statistics For PRA Applications. IMAN,R.L.; PRAIRIE,R.R. Sandia National Laboratories. September 1985. 285pp. 8510040381. SAND85-1495. 32858:029.

This course is intended to provide the necessary probabilistic and statistical skills to perform a PRA. Fundamental background information is reviewed, but the principal purpose is to address specific techniques used in PRAs and to illustrate them with applications. Specific examples and problems are presented for most of the topics.

NUREG/CR-4350 V03: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 3 - Systems Reliability And Analysis Techniques,Session A -Reliability. LOFGREN,E.V. Science Applications International Corp. (formerly Science Applications, Inc.). * Sandia National Laboratories. August 1985. 210pp. 8511190550. SAND85-1495/3. 33541:225.

This course focuses on the quantitative estimation of reliability at the systems level. Various methods are reviewed, but the structure provided by the fault tree method is used as the basis for system reliability estimates. The principles of fault tree analysis are briefly reviewed. Contributors to system unreliability and unavailability are reviewed, models are given for quantitative evaluation, and the requirements for both generic and plant-specific data are discussed. Also covered are issues of quantifying component faults that relate to the systems context in which the components are embedded. All reliability terms are carefully defined.

NUREG/CR-4350 V04: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 4 - System Reliability And Analysis Techniques,Sessions B/C - Event Trees/Fault Trees. * Sandia National Laboratories. HAASL,D. Institute of Systems Sciences. YOUNG,J. Energy, Inc. October 1985. 143pp. 8510290410. SAND85-1495. 33259:151.

This course employs a combination of lecture material and practical problem solving in order to develop competence and understanding of the principles and techniques of event tree and fault tree analysis. The role of these techniques in the overall context of PRA is described. The emphasis of this course is on the basic, traditional methods of event tree and fault tree analysis.

NUREG/CR-4350 V05: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 5 - Systems Reliability And Analysis Techniques,Session D - Quantification. LOFGREN,E.V. Science Applications International Corp. (formerly Science Applications, Inc.). * Sandia National Laboratories. August 1985. 215pp. 8511190554. SAND85-1495/5. 33533:029.

This course focuses on the probabilistic quantification of accident sequences and the link between accident and consequences. Previous sessions in this series have focused on the quantification of system reliability (Session A) and the development of event trees and fault trees (Session B/C). This course takes the viewpoint that the event tree sequences or combinations of system failures and success are available, and that the Boolean equations for system fault trees have been developed and are available.

NUREG/CR-4350 V06: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 6 - Data Development. LEVERENZ,F.L.; COX,D.C. Battelle Memorial Institute, Columbus Laboratories. * Sandia National Laboratories. August 1985. 165pp. 8512120164. SAND85-1495/6. 33876:163.

This course describes the process used to develop quantitative values for events other than human errors in fault trees, event trees, or other system models. The events included are (1) initiating events, (2) unavailability due to hardware failure, (3) unavailability due to maintenance activity, and (4) unavailability due to test activities. The course material covers the most common reliability models for these events and explains how to evaluate them with uncertainties, determine their parameters, and obtain the raw data for this process. Both Bayesian and classical approaches to the data development process are presented.

NUREG/CR-4350 V07: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 7 - Environmental Transport And Consequence Analysis. RITCHIE,L.T. Sandia National Laboratories. August 1985. 400pp. 8511070092. SAND85-1495/7. 33376:120.

Consequence models have been designed to assess health and economic risks from potential accidents at nuclear power plants. These models have been applied to an ever increasing variety of problems with ever increasing demands to improve modeling capabilities and provide realism. This course discusses the environmental transport of postulated radiological releases and the elements and purpose of accident consequence evaluation.

NUREG/CR-4351: SUMMARY REPORT FOR LOFT ANTICIPATED TRANSIENT EXPERIMENT SERIES L6-8. NALEZNY,C.L.; CHEN,T. EG&G Idaho, Inc. (subs. of EG&G, Inc.). November 1985. 109pp. 8511250106. EGG-2406. 33619:346.

Experiment Series L6-8 was conducted in the Loss-of-Fluid Test (LOFT) facility between August 26 and August 31, 1982. This experiment series simulated six individual transients that have a high probability of occurrence during the lifetime of a commercial pressurized water reactor (PWR). The transients simulated for the experiment consisted of two control rod withdrawals (L6-8B-1 and B-2), three small break recovery methods (L6-8C-1, C-2 and C-3), and one natural circulation cooldown with low decay heat (L6-8D). This report presents the experiment results and compares the experimental data with postexperiment calculations made using the RELAP5/MOD1 computer code. The system response during the two rod withdrawal experiments (L6-8B-1 and L6-8B-2) was successful; during L6-8B-

1, negative reactivity feedback was dominated by moderate feedback, while during L6-8B-2 negative reactivity feedback was dominated by Doppler feedback. The comparison between L6-8C-1 and L6-8C-2 demonstrated that although the pressurizer spray is as effective as the power-operated relief valve (PORV) in depressurizing the primary system, it is also more sensitive to operator control. The results of L6-8C-3 indicate that while pump current (or power), is sensitive to void fraction, or density, in the intact loop, it is not an accurate measure of system inventory. The void fraction in the intact loop was about 50% of the void fraction in the entire primary system. Primary system voiding, which occurred as intended during L6-8D, did not significantly affect natural circulation. The data obtained from these six simulations will be valuable for qualifying computer codes used to calculate anticipated transients in commercial PWRs.

NUREG/CR-4352: SUGGESTED STATE REQUIREMENTS AND CRITERIA FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE REGULATORY PROGRAM. RATLIFF,R.A.; DORNSIFE,B.; AUTRY,V.; et al. Conference of Radiation Control Program Directors, Inc. August 1985. 50pp. 8509060210. 32503:323.

Description of criteria and procedure for a state to follow in the development of a program to regulate a LLW disposal site. This would include identifying those portions of the NRC regulations that should be matters of compatibility, identifying the various expertise and disciplines that will be necessary to effectively regulate a disposal site, identifying the resources necessary for conducting a confirmatory monitoring program, and providing suggestions in other areas which, based on experiences, would result in a more effective regulatory program.

NUREG/CR-4354: A STUDY OF SEISMICITY AND TECTONICS IN NEW ENGLAND.Final Report. EBEL,J.E. Boston College, Chestnut Hill, MA. August 1985. 100pp. 8509100335. 32528:181.

The operation by Weston Observatory of a seismic network in New England from 1974 to 1985 is described, and the results of the seismic monitoring are summarized. The network coverage of Weston Observatory increased from two operating stations in 1974 to 36 stations in 1979 and was stabalized at 30 stations in the early 1980's. The network was used to find the locations and magnitudes of all earthquake activity detected during the study period. Most earthquakes from 1974 to 1985 were found to occur in the same places as those which have been documented historically, although the activity appears to be random both in space and time. Studies of aftershocks and detailed monitoring in selected areas did not show any strong correlations between the earthquake locations and mapped geologic structures. It is concluded that the relationship among earthquakes, tetonic or structural zones and faults exposed on the surface are not well understood. The causes of the earthquake activity in the northeast are not clearly established with the seismic data which was gathered and analyzed.

NUREG/CR-4355 V01: 238 PU(IV) IN MONKEYS.Overview Of Metabolism. DURBIN,P.W.; JEUNG,N.; SCHMIDT,C.T. Lawrence Berkeley Laboratory. September 1985. 96pp. 8510020240. LBL-20022. 32839:157.

Complete balance studies were performed using 21 adult and four adolescent Macaque monkeys (three species, both sexes) to define distribution and retention of (238)Pu(IV) citrate from 2 hr to 1100 d after parenteral injection. Experimental methods are described in detail. Initial distribution (6 adults, 7.5 plus minus 0.5 d) and retention (6 adults, 711 plus minus 310 d) of Pu were, respectively, as follows: skeleton and teeth 28 and 14% ID, liver 60 and 11% ID, other soft tissues 6 and 0.9% ID and excretion 5 and 74% ID. Initial Pu content of ovary and testis was 0.005 and 0.06% ID, respectively, and both declined with T1/2 equivalent to 1 yr. Liver Pu was cleared mainly by excretion to feces, but also by recirculation, with an average T1/2 = 180 d. By 1100 d, most soft tissues lost 50 to 90% of their initial Pu content. The initial Pu concentration on trabecular bone surfaces in red marrow was calculated to be about 4.5 times greater than on compact bone surfaces. About one-half of initial skeletal Pu was eliminated in 1100 days, mainly from cancellous structures in red marrow. Implications for some changes in Internationl Commission on Radiological Protection metabolic and dosimetric models for Pu are noted.

NUREG/CR-4357: THE FEASIBILITY OF DETECTING THE IMPORT OF UNAUTHORIZED RADIOACTIVE MATERIALS INTO THE UNITED STATES. BEE,R.W.; GORDON,J.; KWAN,Q.; et al. Aerospace Corp. September 1985. 235pp. 8510040578. 32862:213.

This report explores the feasibility of establishing and operating a radiation monitoring system at U.S. borders to operate in conjunction with normal U.S. Customs Service inspection procedures to detect the presence of accidental radioactive contamination in imported materials. The study defined potential inadvertent contamination threats of radioactive materials in industrial and commercial usage and explored how such materials might accidentally enter new product manufacturing processes. Radiation monitoring equipment necessary to detect such contamination was examined, and a technical system description and capital, operating, and maintenance costs were developed. Hypothetical scenarios were developed to estimate a range of costs for recall, cleanup, and disposal of contaminted products well after distribution to consumers. Estimates were made of added health risk to consumers associated with exposure to contaminant radiation if the contaminated products go undetected. An economic analysis was performed to provide a common basis for their comparison. Because such aspects as frequency, level of contamination, characteristics of product distribution, and health effects cannot be assessed using absolute bounding conditions, no direct comparisons of courses of action can be made. Rather, the study provides decision makers with various dimensions of the problems, thus providing an improved basis for related decision making.

NUREG/CR-4358: APPLICATIONS OF DENSITY PROFILING TO EQUIPMENT QUALIFICATION ISSUES. GILLEN,K.T.; CLOUGH,R.L.; DHOOGE,N.J. Sandia National Laboratories. September 1985. 38pp. 8510040354. SAND85-1557. 32861:012.

This paper reviews the density profiling technique, a new, inexpensive and versatile analytical method which can yield extremely useful information on heterogeneities in polymers. The technique makes use of a density gradient column to measure the density of a series of successively-cut slices across a sample. Since the density of very thin slices can easily be obtained, density profiles across very small cross-sections (< 1mm) are readily available. A major application of the technique involves oxidation studies of polymers, since oxidation reactions usually lead to substantial increases in polymer density. Diffusion-limited oxidation effects, which lead to heterogeneously oxidized materials, are often present in polymer aging studies in air. Since these effects are responsible for the commonly-observed physical dose-rate effects in radiation aging environments and for non-Arrhenius behavior in thermal aging environments, the availability of simple oxidation profiling techniques is a tremendous aid in validating the aging simulation aspects of equipment qualification procedures. This paper gives examples of the utility of density profiling for studying oxygen diffusion-limited degradation in both radiation and thermal aging environments and in discovering/understanding chemical dose-rate effects in high energy radiation environments.

NUREG/CR-4360 V01: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES. WILLIAMS,W.R. Oak Ridge National Laboratory. September 1985. 135pp. 8512120100. ORNL/ENG/TM-31. 33873:215.

Calculational methods and computer programs for the analysis of source terms for postulated releases of UF(6) are presented. Required thermophysical properties of UF(6), HF, and H(2)O are described in detail. UF(6) reacts with moisture in the ambient environment to form HF and H(2)O. The coexistence of HF and H(2)O significantly alters their pure component properties, and HF vapor polymerizes. A release rate model of UF(6) is presented that considers the transient conditions inside containment and the flashing, multiphase flow of UF(6) along the release pathway. Transient compartment models for simulating UF(6) release inside rooms ventilated by forced- and induced-draft systems are also described. The basic compartment model mass and energy balances are supported by simple heat transfer, ventilation system, and deposition models. A model that can simulate either a closed compartment or a steadt-state ventilation system is also discussed. Listings of all main programs (including two plotting routines) and subroutines are included. Example problems illustrate the analysis of postulated releases using the described programs.

NUREG/CR-4360 V02: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES. WILLIAMS,W.R. Oak Ridge National Laboratory. September 1985. 214pp. 8512120102. ORNL/ENG/TM-31. 33873:350.

Calculational methods and computer programs for the analysis of source terms for postulated releases of UF(6) are presented. Required thermophysical properties of UF(6), HF, and H(2)O are described in detail. UF(6) reacts with moisture in the ambient environment to form HF and H(2)O. The coexistence of HF and H(2)O significantly alters their pure component properties, and HF vapor polymerizes. A release rate model of UF(6) is presented that considers the transient conditions inside containment and the flashing, multiphase flow of UF(6) along the release pathway. Transient compartment models fro simulating UF(6) releases inside rooms ventilated by forced- and induced-draft systems are also described. The basic compartment model mass and energy balances are supported by simple heat transfer, ventilation system, and deposition models. A model that can simulate either a closed compartment or a steady-state ventilation system is also discussed. Listings of all main programs (including two plotting routines) and subroutines are included. Example problems illustrate the analysis of postulated releases using the described programs.

NUREG/CR-4361: STEAM GENERATOR GROUP PROJECT.Annual Report - 1983. CLARK,R.A.; LEWIS,M.; MUSCARA,J. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1985. 125pp. 8511050309. PNL-5017. 33339:181.

The Steam Generator Group Project (SGGP) is an NRC program jointly funded by consortia from France, Italy and Japan, and the Electric Power Research Institute. The SGGP utilizes a steam generator removed from service at a nuclear plant as a vehicle for research on a variety of safety and reliability issues. This report is an annual summary of progress on the program for 1983. Information is presented on the task of removing 969 plugs by drilling them out of the tubes in the steam generator to permit access for eddy current nondestructive testing probes. A description is given of two post-service baseline eddy current inspections, one by Zetec and one by Intercontrole, of about 3000 tubes. Photographic documentation is provided of damage observations on the secondary side of the unit. Corrosion coupon results are reported from the channel head decontaminations described in the 1982 annual report. Plans for further eddy current testing are discussed, including a plan to conduct a round robin series of tests in which up to eight independent, commercial inspection teams would all inspect the same set of selected tubes in the steam generator.

NUREG/CR-4362: STEAM GENERATOR GROUP PROJECT.Annual Report - 1984. CLARK,R.A.; LEWIS,M.; MUSCARA,J. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1985. 109pp. 8511210279. PNL-5417. 33557:321.

This report is a summary of progress in the Surry Steam Generator Group Project for 1984. Information is presented on the analysis of two baseline eddy current inspections of the generator. Round robin series of tests using standard in-service inspection techniques are described along with some preliminary results. Observations are reported of degradation found on tubing specimens removed from the generator, and on support plates characterized in-situ. Residual stresses measured on a tubing specimen are reported. Two steam generator repair demonstrations are described; one for antivibration bar replacement, and one on tube repair methods. Chemical analyses are shown for sludge samples removed from above the tube sheet.

NUREG/CR-4365: DESIGN AND DEVELOPMENT OF A SPECIAL PURPOSE SAFT SYSTEM FOR NONDESTRUCTIVE EVALUATION OF NUCLEAR REACTOR VESSELS AND PIPING COMPONENTS. GANAPATHY,S.; SCHMULT,B.; WU,W.S.; et al. Michigan, Univ. of, Ann Arbor, MI. August 1985. 126pp. 8509100506. 32536:106.

This report describes the design details of a special purpose system for real-time nondestructive evaluation of reactor vessels and piping components. The system consists of several components and the report presents the results of the research aimed at the design of each component and recommendations based on the results. One major component of the NDE system, namely the real-time SAFT processor was designed with sufficient details to enable the fabrication of a prototype by GARD Inc. under a subcontract from The University of Michigan and the report includes their results and conclusions.

NUREG/CR-4367: ORVIRT.PC:A 2-D FINITE-ELEMENT FRACTURE ANALYSIS PROGRAM FOR A MICROCOMPUTER. BRYSON,J.W. Oak Ridge National Laboratory. October 1985. 60pp. 8512270362. ORNL-6208. 34078:271.

ORVIRT.PC (Oak Ridge VIRTual crack extension.Personal Computer) is a 2-D finite element fracture-analysis program for an IBM PC/AT microcomputer. The code is based to a large extent on the techniques used in the ORMGEN-ADINA-ORVIRT fracture-analysis system. ORVIRT.PC is a stand-alone program capable of performing 2-D linear elastic stress and fracture-mechanics analyses. Thermal loadings may be analyzed in addition to mechanical loadings. Crack-face tractions may also be considered. Eight-noded isoparametric elements which combine both performance and ease of modelling are employed in the program. Special crack-tip elements which allow for an inverse square root variation in the near-tip stress and strain fields are used at the crack tip. Detailed user instructions are provided which describe both preparation of input data and program operation. Sample problems are presented which demonstrate good agreement with known solutions.

NUREG/CR-4368: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Semiannual Report,October 1984 - March 1985. KUPPERMAN,D.S.; CLAYTOR,T.N.; MATHIESON,T.; et al. Argonne National Laboratory. October 1985. 38pp. 8511250006. ANL-85-46. 33620:225.

Two pipe-to-endcap weldments with overlays were examined. Results conclude that it is extremely difficult to inspect pipes with overlays because of unpredictable beam distortion due to the overlay and the absence of effective reference pipes. The use of 1-Mhz longitudinal angle-beam probes rather than shear-wave probes may facilitate inspection of such pipes. Four 60-mm-thick cast stainless steel plates with microstructures ranging from equiaxed to primarily columnar grains have been examined with ultrasonic waves. It was found that the longitudinal velocity of sound and the ratio of longitudinal to shear velocity as a function of position can be used to characterize the crystallographic texture. It was also found that the beam skewing that occurs in columnar (but not equiaxed) structures is strong enough so that measurements of probe separation at maximum received signal intensity for 45 degrees shear-wave pitch-catch transducers can be correlated with microstructure. Leaks from a

2-in. ball valve and a flange were studied and compared with leaks from intergranular stress corrosion cracks (IGSCCs) and fatigue cracks. The dependence of acoustic signal on flow rate and frequency for the valve and the flange was comparable to that of fatigue cracks (thermal and mechanical) and different from that of IGSCCs.

NUREG/CR-4373: COMPENDIUM OF COST-EFFECTIVENESS EVALUATIONS OF MODIFICATIONS FOR DOSE REDUCTION AT NUCLEAR POWER PLANTS. BAUM,J.W.; MATTHEWS,G.R. Brookhaven National Laboratory. December 1985. 142pp. 8601070478. BNL-NUREG-51915. 34187:005.

This report summarizes available information on cost effectiveness of engineering modifications of potential value in dose reduction at nuclear power plants. Data was gathered from several U.S. utilities, published literature, equipment and service suppliers, and recent technical meetings. Five simplified econometric models were employed to evaluate data and arrive at a value for cost effectiveness expressed in either (a) dollars/rem; or (b) total dollar savings calculated using a nominal value of $1,000/rem. Models employed were: a basic model with no consideration given to the time value of money; two models in which discounting was used to evaluate costs and savings in terms of present values; and two models in which income taxes and revenue requirements were considered. Results from different models varied by as much as a factor of 10, and were generally lowest for the basic model and highest for the before-tax revenue requirements model. Results for 151 evaluations employing different assumptions concerning number of plants per site and outage impacts were tabulated in order of decreasing cost effectiveness. Twenty-five evaluations were identified as exceptionally cost effective since both costs and dose were saved. Forty evaluations indicated highly cost-effective changes based on costs of less than $1,000/rem saved using results of the present-worth model that included discounting of future dose savings.

NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS. ADAMS,S.R. EG&G Idaho, Inc. (subs. of EG&G, Inc.). October 1985. 258pp. 8512270240. EGG-2415. 34079:355.

A comprehensive evaluation was conducted of the radiation protection practices and programs at prototype LMFBRs with long operational experience. Installations evaluated were the Fast Flux Test Facility (FFTF), Richland, Washington; Experimental Breeder Reactor II (EBR-II) Idaho Falls, Idaho; Prototype Fast Reactor (PFR) Dounreay, Scotland; Phenix, Marcoule, France; and Kompakte Natriumgekuhlte Kernreak Toranlange (KNK II), Karlsruhe, Federal Republic of Germany. The evaluation included external and internal exposure control, respiratory protection procedures, radiation surveillance practices, radioactive waste management, and engineering controls for confining radiation contamination. The theory, design, and operating experience at LMFBRs is described. Aspects of LMFBR health physics different from the LWR experience in the United States are identified. Suggestions are made for modifications to the NRC Standard Review Plan based on the differences.

NUREG/CR-4376: HEAT TRANSFER,CARRYOVER AND FALL BACK IN PWR STEAM GENERATORS DURING TRANSIENTS. LIAO,L.H.; PARLOS,A.; GRIFFITH,P. Massachusetts Institute of Technology, Cambridge, MA. September 1985. 212pp. 8510020307. EPRI NP-4298. 32828:124.

The concern over Pressurized Thermal Shock (PTS), along with many other concerns, indicates the need for accurate knowledge of the steam generator behavior during the blowdown of the steam generator secondary side. To fulfill this need a computer program, SIT-SG (Simulator of Transient in Steam Generator) is developed. This is a one-dimensional best-estimate code with the assumption that the vapor and liquid phases are in thermal equilibrium but not homogeneous. The drift flux model is used to describe the relationship between the vapor and the liquid phase velocity. No momemtum equation is required for SIT-SG because the detailed pressure distribution in the vessel is not important for the blowdown process. Based on the comparisons between the code predictions and the data obtained from the experiments conducted in Battelle-Frankfurt and GE, the best flux model constants for various flow regimes are selected. SIT-SG has been used to predict the carryover, fall back, and heat transfer for the MIT steam generator blowdown experiments. The results are encouraging. It is found that the measured dryout front is much higher than the calculated mixture level. If the effective heat transfer are is determined from the mixture level, the primary-to-secondary heat teransfer will be substantially underpredicted. From the result of the liquid hold up study we would expect to find two mixture levels, one in the bottom of the steam generator and one above the top tube support plate, provided that flooding occurs at all.

NUREG/CR-4377: EVALUATIONS AND UTILIZATIONS OF RISK IMPORTANCES. VESELY,W.E.; DAVIS,T.C. Battelle Memorial Institute, Columbus Laboratories. August 1985. 131pp. 8509100355. BMI-2129. 32528:049.

This report presents approaches for utilizing Probabilistic Risk Analysis (PRA) to determine risk importances. PRAs can be used to identify the importances of risk contributors or proposed changes to designs or operations. The objective of this report is to serve as a handbook and guide in evaluating and applying risk importances. The utilization of both qualitative risk importances and quantitative risk importances is described in this report. Qualitative risk importances are based on the logic models in the PRA, while quantitative risk importances are based on the quantitative results of the PRA. Both types of importances are among the most robust and meaningful information a PRA can provide.

NUREG/CR-4379 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report,Year Four April-June 1985. STAHL,D.; MILLER,N.E. Battelle Memorial Institute, Columbus Laboratories. September 1985. 111pp. 8510020228. 32838:277.

High-level waste glass studies are being concluded and efforts are being directed toward studying spent-fuel performance. The effects of devitrification on glass leach rates are being investigated, and silica dissolution was studied to provide data for the glass dissolution model. Preliminary data support this model. A leach test using organic acids was conducted and leaching trends were observed. Real and simulated spent fuels are being incorporated in integral tests using simulated groundwater in a prototypic repository environment. The reactions of groundwater species with steels are being analyzed to evaluate susceptability to pitting and stress-corrosion cracking. Potential cracking agents are being investigated by slow strain rate experiments. General and pitting corrosion models were further developed, based on known principles of mass transport and radiolytic production. A simplified groundwater-radiolysis model, developed for use with the corrosion models, was compared with other mechanisms for species concentration predictions.

NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES. WRENN,M.E.; SINGH,N.P.; PASCHOA,A.S.; et al. Utah, Univ. of, Salt Lake City, UT. September 1985. 58pp. 8510250529. 33191:286.

The alpha-emitting isotopes of uranium and thorium were determined in the lungs of 14 former uranium miners and in soft tissues and bones of three miners and two millers. These radionuclides were also determined in soft tissues and bones of seven normal controls. The average concentrations in pCi/kg wet weight in 17 former miners' lungs are as follows: U-238, 75; U-234, 80; Th-230, 79. Concentrations of each nuclide ranged from 2 to 325 pCi/kg. The average ratio of U-238/U-234 was 0.92, ranging from 0.64 to 1.06. The mean ratio of Th-230/U-234 indicates that the rate of elimination of uranium and thorium from lungs is the same in former uranium miners. The concentrations of U-234 and U-238 were highest in lung; however, the concentration of Th-230 in bones was either higher than or comparable to its concentration in lung. The concentration ratios of Th-230/U-234 in bone of uranium miners and millers measured in our laboratory have been compared with results predicted by ICRP-30 metabolic models. These results indicate that the ICRP metabolic models for thorium and uranium were only marginally successful in predicting the ratio of Th-230/U-234 in bones, and that effective release rate of uranium from skeleton may be more rapid than predicted by the ICRP model.

NUREG/CR-4383: HIGH PRESSURE INJECTION OF MELT FROM A REACTOR PRESSURE VESSEL - THE DISCHARGE PHASE. PILCH,M.; TARBELL,W.W. Sandia National Laboratories. September 1985. 41pp. 8511070237. SAND85-0012. 33380:262.

Recent probabilistic risk-assessment studies identified potential accident sequences in which reactor vessel failure occurs while the primary system is at elevated pressure. The phenomenology of the discharge phase is reviewed here. We propose an improved model for hole ablation following vessel failure, and we compare the model with experiment data. Gas blowthrough is identified as a mechanism that allows steam to escape through the vessel breach before melt ejection is complete. Gas blowthrough leads to pneumatic atomization of the remaining melt before significant depressurization of the primary system occurs.

NUREG/CR-4385: EFFECTS OF CONTROL SYSTEM FAILURES IN TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A WESTINGHOUSE PRESSURIZED WATER REACTOR. BICKFORD,W.E.; TABATABAI,A.S. Battelle Memorial Institute, Pacific Northwest Laboratories. November 1985. 95pp. 8512050461. PNL-5543. 33769:098.

Pacific Northwest Laboratory (PNL) performed a probablistic risk assessment to develop estimates of core-melt frequency and public risk due to control system failures in a Westinghouse pressurized water reactor. Value/impact analysis of proposed systems modifications to prevent the control system failures were also performed. Four control system failure modes were analyzed: 1) overfill, 2) overcool, 3) overpressure, and 4) steam generator tube rupture. For each mode, two failure sequences were postulated. These analyses were based on the results of failure modes and effects analyses previously performed at Idaho National Engineering Laboratory and conducted in support of the U.S. Nuclear Regulatory Commission's program for Unresolved Safety Issue A-47: Safety Implications of Control Systems.

NUREG/CR-4386: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A BABCOCK AND WILCOX PRESSURIZED WATER REACTOR. BICKFORD,W.E.; TABATABAI,A.S. Battelle Memorial Institute, Pacific Northwest Laboratories. December 1985. 62pp. 851 270329. PNL-5544. 34084:259.

Pacific Northwest Laboratory (PNL) performed probabilistic risk analyses aimed at developing estimates of core-melt frequency and public risk associated with control system failures in a Babcock and Wilcox pressurized water reactor, and value/impact analyses of proposed systems modifications. These analyses were based on the results of failure modes and effects analyses previously performed at the Oak Ridge National Laboratory (ORNL). The control system failure modes that were identified by ORNL and analyzed by PNL fall into three main scenarios: 1) overfill of the steam generators progressing to spillover into the steam lines, 2) ICS hand power failure progressing to steam generator dryout, and 3) ICS automatic power failure progressing to steam generator failure. For each of these modes, failure sequences were postulated. The results of PNL's probabilistic analysis of failure progression to core damage and value/impact analyses of possible resolutions to prevent the occurrence of these failures are presented in this report.

NUREG/CR-4387: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS AND CORE-MELT FREQUENCIES AT A GENERAL ELECTRIC PRESSURIZED WATER REACTOR. BICKFORD,W.E.; TABATABAI,A.S. Battelle Memorial Institute, Pacific Northwest Laboratories. December 1985. 87pp. 8512270082. PNL-5545. 34078:099.

Pacific Northwest Laboratory (PNL) performed probabilistic risk analyses to estimate core-melt frequency and public risk associated with control system failures in a General Electric boiling water reactor. PNL also conducted value/impact analyses of proposed modifications of these control systems to prevent these failures. These analyses were based on failure modes and effects analyses previously performed by the Idaho National Engineering Laboratory (INEL). The control system failure modes identified by INEL and analyzed by PNL fall into three main scenarios: 1) failures that initiate feedwater overfill and also defeat the high level feedwater trip, 2) a failure of the condensate booster pump that results in increased flow to the vessel (overfill), and 3) an inadvertent actuation of the low pressure coolant injection system (LPCI) that also produces an excessive cooldown (overcool). For each of these modes, two failure sequences were postulated. The results of PNL's probabilistic analysis of failure progression to core damage and value/impact analyses of possible resolutions to prevent the occurrence of these failures are presented in this report.

NUREG/CR-4388: AEROSOL BEHAVIOR MODELING (TASK 3) - SUPPORT SERVICES FOR RESEARCH AND EVALUATION OF SEVERE ACCIDENT PHENOMENA AND MITIGATION. JORDAN,H.; KOGAN,V. Battelle Memorial Institute, Columbus Laboratories. October 1985. 78pp. 8511110431. BMI-2130. 33417:016.

This report covers exploratory research done on a number of aerosol topics relevant to nuclear safety during the period 5/1/84-4/30/85. Much of this research required the modification and development of the QUICKM code to accomodate steam condensation/evaporation at wall and particle surfaces in a more mechanistic fashion than previously possible. The principal finding of this effort is that steam condensation on aerosol particles is a very sensitive function of the numerical model and that all present approaches are probably inadequate for its proper assessment. Further work is needed in this area before predictions of aerosol growth by steam condensation can be believed. Another aerosol topic approached with the aid of the QUICKM code was that of the coagglomeration of aerosol particles of different types, as might occur in LWR meltdown sequences when the aerosol generated by core/concrete interaction mingles with the existing radioactive aerosol of the containment. This study showed that a "multiple component" approach, such as that of QUICKM and CONTAIN can yield markedly different predictions to those of a "single component" approach as used in the NAUA code. It also revealed additional sensitivities of the multiple component approach to input data that are generally not well known. Finally, the possible effect on aerosol behavior of the decay of fission products associated with the aerosol particles in containment was investigated. This limited investigation revealed that decay can potentially affect aerosol behavior under some circumstances.

NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP).A Methodology For Estimating Risk Impacts Of Safeguards Measures. HORTON,W.; LOBNER,P.; KARIMIAN,S.; et al. Brookhaven National Laboratory. October 1985. 257pp. 8511190505. BNL-NUREG-51926. 33542:134.

This report presents a method called Measures of Sabotage Risk Employing PRA (MOSREP) which was developed to systematically evaluate the desirability of sabotage vulnerability reduction measures. The method has been specifically designed to be a starting point for resolving Generic Issue A-29. MOSREP has been designed to provide a technical basis for regulatory actions involving design changes, damage control measures, physical protection vandalism, and tampering at nuclear power plants. It is the intent of MOSREP to provide support to the NRC staff in determing effective regulatory strategies and evaluating trade-offs associated with activities which impact both safeguards and safety.

NUREG/CR-4395: CORRELATION OF CV AND KIC/KJC TRANSITION TEMPERATURE INCREASES DUE TO IRRADIATION. HISER,A. Materials Engineering Associates, Inc. November 1985. 94pp. 8512270355. NEA-2086. 34082:024.

Reactor pressure vessel (RPV) surveillance capsules contain Charpy-V (C(v)) specimens, but many do not contain fracture toughness specimens; accordingly, the radiation-induced shift (increase) in the brittle-to-ductile transition region (triangleT) is based upon the triangleT determined from notch ductility (C(v) tests. Since the ASME K(Ic) and K(IR) reference fracture toughness curves are shifted by the triangleT from C(v), assurance that this triangleT does not underestimate triangleT associated with the actual irradiated fracture toughness is required to provide confidence that safety margins do not fall below assumed levels. To assess this behavior, comparisons of triangleT's defined by elastic-plastic fracture toughness and C(v) tests have been made using data from RPV base and weld metals in which irradiations were made under test reactor conditions. As well, comparison of triangleT's at various index levels gives indications of C(v), K(Jc) and K(Bc) curve shape change due to irradiation. Lastly, comparisons between measured triangleT(C(v)) and values using various correlations or models are also made.

NUREG/CR-4396: SIMMER POSTPROCESSOR MANUAL. PARKER,F. Los Alamos Scientific Laboratory. November 1985. 140pp. 8512050431. LA-10549-M. 33770:010.

The T6P postprocessor analyzes SIMMER-II TAPE6 and TAPE36 data. The processor calculates new variables, integrates variables over regions of the mesh, compares values of a variable of a given problem at selected times, compares values of a variable at selected times between problems for parameter studies, calculates derivatives of a variable with respect to time or another variable, traces a variable at a requested location in the mesh over time. The data are presented to the user graphically, using two-dimensional graphs and three-dimensional perspective or contour plots. Interactive graphic techniques may be used with perspective plots.

NUREG/CR-4397: IN-PLANT SOURCE TERM MEASUREMENTS AT PRAIRIE ISLAND NUCLEAR GENERATING STATION. MANDLER,J.W.; STALKER,A.C.; CRONEY,S.T.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). September 1985. 565pp. 8510040592. EGG-2420. 32863:172.

This report presents data obtained at Prairie Island as a part of the In-Plant Source Term Measurement Program in operating light water reactors (LWRs). the work was conducted for the Office of Nuclear Regulatory Research (RES) in support of the Meteorology and Effluent Treatment Branch (METB) of the Office of Nuclear Reactor Regulation (NRR). The primary objective of this program is to provide the Nuclear Regulatory Commission (NRC) with operational data that can be used in evaluation of plant designs for liquid and gaseous radwaste treatment systems. Data presented were obtained at the Prairie Island Nuclear Generating Station, operated by Northern States Power, located near Red Wing, Minnesota. In-plant measurements were conducted during the time period from October 1980 to August 1981. This plant is the fifth in a series of operating LWRs to be studied.

NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS. SCIACCA,F.; NELSON,W.; SIMPKINS,B.; et al. Science & Engineering Associates, Inc. September 1985. 92pp. 8510030131. 32849:332.

This report addresses the differences between the existing and proposed Appendix J and identifies eleven substantive areas where quantifiable impacts will likely result. The analysis indicated that there are four areas of change which tend to dominate all others in terms of cost impacts. The applicable paragraph numbers from Draft E2 of the Appendix J revision and the nature of the change follows: III.A(4) & III.A(6) - Test Pressure & Testing at Reduced Pressure No Longer Allowed; III.A(7)(b)(i) Acceptance Criteria 1.0 La Acceptable "As Found" Leakage; III.A(8)(a) Retesting Following Failure of "As Found" Type A Test - Corrective Action Plan, and III.A(8)(b)(ii) Option to Do More Frequent Type B & C Testing Rather Than More Type A Penalty Tests. The best estimate is that the proposed Appendix J would result in cost savings ranging from about $100 million to $160 million, and increase routine occupational exposure on the order of 10,000 person-rem. These estimates capture the total impact to industry and the NRC over the assumed operating life of all existing and planned future power reactors. All dollar impacts projected to occur in future years have been present worthed at discount rates ranging from 5% to 10%.

NUREG/CR-4399: POSSIBLE OPTIONS FOR REDUCING OCCUPATIONAL DOSE FROM THE TMI-2 BASEMENT. MUNSON,L.F.; HARTY,R. Battelle Memorial Institute, Pacific Northwest Laboratories. November 1985. 125pp. 8512090545. PNL-5557. 33819:135.

The March 28, 1979 accident at Three Mile Island Unit 2 filled the basement to a depth of several feet with highly contaminated water. The water has been drained and various characterization efforts are underway. Dose rates range from approximately 40 to more than 1100 R/hr. Identified sources include a structure of hollow concrete blocks that is estimated to contain between 11,000 and 19,000 curies of cesium 137 and other quantified sources that contain between 570 and 1800 additional curies. Decontamination methods and approaches available for cleanup are discussed.

NUREG/CR-4400: THE IMPACT OF MECHANICAL-AND MAINTENANCE-INDUCED FAILURES OF MAIN REACTOR COOLANT PUMP SEALS ON PLANT SAFETY. AZARM,M.A.; BOCCIO,J.L. Brookhaven National Laboratory. MITRA,S. Impell Corp. December 1985. 103pp. 8601070464. BNL-NUREG-51928. 34186:093. This document presents an investigation of the safety impact resulting from mechanical- and maintenance-induced reactor coolant pump (RCP) seal failures in nuclear power plants. A data survey of the pump seal failures for existing nuclear power plants in the U.S. from several available sources was performed. The annual frequency of pump seal failures in a nuclear power plant was estimated based on the concept of hazard rate and dependency evaluation. The conditional probability of various sizes of leak rates given seal failures was then evaluated. The safety impact of RCP seal failures, in terms of contribution to plant core-melt frequency, were also evaluated for three nuclear power plants. For leak rates below the normal makeup capacity and the impact of plant safety was discussed qualitatively, whereas for leak rates beyond the normal make up capacity, formal PRA methodologies were applied.

NUREG/CR-4401: BEHAVIOR OF CONTROL RODS DURING CORE DEGRADATION: PRESSURIZATION OF SILVER-INDIUM-CADMIUM CONTROL RODS. POWERS,D.A. Sandia National Laboratories. November 1985. 187pp. 8512270269. SAND85-0469. 34081:001.

Activity data for the liquid binary systems Ag-Cd, Ag-In, and In-Cd are correlated in terms of the Wilson equation. These correlations are used to construct a model of the ternary system Ag-In-Cd. Spectroscopic data for the vapor species Ag(g), Ag(2)(g), Ag(3)(g), Ag+(g), In(g), In(2)(g), In+(g), Cd(g), Cd(2)(g), Cd+(g), AgIn(g), and CdIn(g) are reviewed and are used to define thermodynamic functions for these species for temperatures between 298 and 3500 K. Vapor pressures for the liquid phase pure elements, liquid binary alloys, and the liquid ternary alloy are calculated using the Wilson equation model and using the assumption that the condensed phase is an ideal mixture. An azeotrope is predicted for the Ag-In system. Predictions are made of the vaporization of alloys of 80 percent Ag, 15 percent In, and 5 percent Cd used as control materials in some pressurized water reactors.

NUREG/CR-4402 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31,1985. BALL,S.J.; CLEVELAND,J.C.; HARRINGTON,R.M.; et al. Oak Ridge National Laboratory. October 1985. 15pp. 8512270361. ORNL/TM-9798/V1. 34078:329.

Modeling, code development, and analyses of the modular High-Temperature Gas-Cooled Reactor (HTGR) continued with work on the side-by-side design. Fission-product release and transport experiments were completed. A description and assessment report on Oak Ridge National Laboratory HTGR safety codes was issued.

NUREG/CR-4403: SUMMARY OF THE WASTE MANAGEMENT PROGRAMS AT URANIUM RECOVERY FACILITIES AS THEY RELATE TO THE 40 CFR PART 192 STANDARDS. GILLEN,D.; BALDWIN,J.S.; CAMPBELL,A.W.; et al. Oak Ridge National Laboratory. November 1985. 270pp. 8512270087. ORNL/TM-9797. 34076:001.

This study evaluates the degree to which surface impoundments at licensed facilities comply with significant changes in NRC requirements initiated by enactment of EPA's final environmental standards for uranium recovery facilities (40 CFR Part 192). Impoundment liner requirements, groundwater protection standards, groundwater monitoring and corrective action programs, and site closure standards are the most significant regulatory modifications. The compliance status of 30 conventional mills and 31 in-situ mines is determined from a review of Nuclear Regulatory Commission and agreement state docket files through November, 1983. Results of this review show that a majority of conventional uranium mill tailings management systems are proposed expansions to existing impoundments, as well as with respect to some aspects of groundwater monitoring and compliance programs. Furthermore, the status of conventional mill site closure plans is highly uncertain at this time. Although surface impoundments at in-situ uranium mines also are deficient with respect to groundwater monitoring programs, they generally comply with other changes in requirements imposed by 40 CFR Part 192.

NUREG/CR-4406: AN ANALYSIS OF LOW-LEVEL WASTES.Review of Hazardous Waste Regulations And Identification of Radioactive Mixed Wastes. Final Report. BOWERMAN,B.S.; KEMPF,C.R.; MACKENZIE,D.R.; et al. Brookhaven National Laboratory. December 1985. 172pp. 8601070506. BNL-NUREG-51933. 34189:286.

Regulations promulgated by the U.S. Environmental Protection Agency governing the disposal of hazardous wastes were reviewed. A survey was carried out to establish a data base on the nature and composition of low-level radioactive wastes (LLW) in order to determine whether some LLW could also be considered hazardous as defined in 40 CFR Part 261. Completed questionnaires were returned by 97 of the 238 reactor and nonreactor facilities contacted. The waste volumes reported by these respondents corresponded to approximately 29% of all LLW disposed of in 1984. The analysis of the survey results indicated that three broad categories of LLW may be radioactive mixed wastes. They include: waste containing organic liquids, disposed of by all types of generators; wastes containing lead metal, i.e., discarded shielding or lead containers; and wastes containing chromates, i.e., nuclear power plant process wastes where chromates are used as corrosion inhibitors. Certain wastes, specific to particular generators, were identified as potential mixed wastes as well.

NUREG/CR-4414: DIRECT-CONTACT CONDENSATION OF STEAM ON COLD WATER IN STRATIFIED COUNTERCURRENT FLOW. BANKOFF,S.G.; KIM,H.J. Northwestern Univ., Evanston, IL. October 1985. 81pp. 8511120063. 33440:323.

Experimental data are reported on: Local condensation heat transfer coefficients and local interfacial shear stresses for countercurrent stratified flow of steam and cold water at atmospheric pressure in a flat plate geometry at an inclination of 4 degrees, 30 degrees and 33 degrees from horizontal. Data are correlated in terms of the gas and liquid Reynolds numbers.

NUREG/CR-4415: COUNTER-CURRENT STEAM/WATER FLOW ABOVE A PERFORATED PLATE-VERTICAL INJECTION OF WATER. DILBER,I. Northwestern Univ., Evanston, IL. October 1985. 62pp. 8511120087. 33441:283.

Experimental data are presented on steam/water counter-current flow limiting phenomena. Weep points and total dumping points are determined for low and high water injection rates above a perforated plate.

NUREG/CR-4416: STABILITY OF STEAM-WATER COUNTER-CURRENT STRATIFIED FLOW. LEE,S.C. Northwestern Univ., Evanston, IL. October 1985. 242pp. 8511120083. 33441:045.

The stability of countercurrent flow of steam and water in inclined rectangular ducts is investigated. Two flow instabilities which limit the normal condensation process in countercurrent stratifed flow have been identified experimentally: flooding and condensation induced water hammer. Analyses of both conditions are performed on a basis of flow stability and heat transfer consideration.

NUREG/CR-4417: LOCAL PROPERTIES OF COUNTERCURRENT STRATIFIED STEAM-WATER FLOW. KIM,H.J. Northwestern Univ., Evanston, IL. October 1985. 227pp. 8511120029. 33440:001.

A study of condensation in countercurrent stratified flow of steam and subcooled water has been carried out in rectangular channel/flat geometry over a wide range of inclinator angles and aspect ratios. Local condensation rates and pressure gradients were measured from which local heat transfer coefficients and interfacial shear stresses were determined and correlated in terms of Reynolds and Freidl numbers.

NUREG/CR-4422: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN PWR ENVIRONMENTS. CULLEN,W.; GABETTA,G.; HANNINEN,H. Materials Engineering Associates, Inc. December 1985. 116pp. 8601070455. MEA-2078. 34183:063.

The crack-tip micromechanisms and the computational models for environmentally-assisted cracking in pressure vessel and piping steels in high-temperature, low-oxygen (PWR), reactor-grade water are described and evaluated in this report. The micromechanistic models are discussed in some detail, with anodic dissolution and hydrogen assistance being the prime candidates for the successful explanation of the observed phenomena. The anodic dissolution model offers far better quantification of the environmentally-assisted crack growth rats, but tends to overpredict the rates for a large number of conditions. The hydrogen assistance models qualitatively could account for a wide range of effects, but quantification of the model is virtually nonexistent. A variety of calculational models are in various stages of development; all of them are far from use as a predictive tool. Crack-tip strain rate models have received the most attention, and the approach to their use has been to partition the environmentally-assisted growth rates into a mechanically-driven component, with the environmental enhancement superposed. The environment component is then correlated with a calculated crack-tip strain rate.

NUREG/CR-4424: DROPLET SIZES,DYNAMICS AND DEPOSITION IN VERTICAL ANNULAR FLOW. LOPES,J.C.; DUKLER,A.E. Houston, Univ. of, Houston, TX. October 1985. 318pp. 8511110335. 33411:074.

Iodine release from a nuclear power plant during steam generator tube rupture accidents is expected to be strongly dependent on the drop sizes formed as high pressure primary system water is flashed and atomized as it passes through the rupture opening. This study was based on the need for information on drop sizes formed under such conditions. Experiments to measure the fraction of water flashed and the drop sizes formed were performed at typical operating pressures and temperatures with the actual tube diameters and lengths nearly to scale. The mass median drop sizes measured were in the range from about 20 to 60 micrometers for both open-ended and slit rupture geometries. No significant effect on drop size of primary system pressure level was noted over the range from 1100 to 2100 psig.

NUREG/CR-4428: OVERVIEW OF TRAC-BD1/MOD1 ASSESSMENT STUDIES. CHARBONEAU,B.L. EG&G Idaho, Inc. (subs. of EG&G, Inc.). November 1985. 53pp. 8512270264. EGG-2422. 34079:299. This report summarizes a series of computer simulations sponsored by the United States Nuclear Regulatory Commission (USNRC) performed at the Idajo National Engineering Laboratory (INEL) to continue the advancement of boiling water reactor (BWR) safety research. The simulations were performed to evaluate the analysis capabilities of the Transient Reactor Analysis Code BWR version (TRAC-BD1/MOD1) to calculate operational transients, including anticipated transients without scram (ATWS) and loss-of-coolant accidents (LOCAs). The assessment simulations were performed for a broad range of scenarios, to encompass as many different phenomena as possible. Comparisons are made between the measured and calculated data. Conclusions are made with respect to the calculated system pressure response, thermal response, and break flow response, as well as the capabalities to model containment and natural circulation conditions. Recommendations are made with respect to user guidelines.

NUREG/CR-4429: TRAC-BD1/MOD1 USERS GUIDELINE. HANSON,R.G. Idaho National Engineering Laboratory. November 1985. 77pp. 8512120107. EGG-2423. 33873:114.

Code assessment studies and specific code applications have provided insight into the effective use of the TRAC-BWR series of codes. This document reports the experience gained from the studies and serves to assist the user in the effective application of the TRAC-BD1/MOD1 computer code. This document stresses the user's perspective relative to appropriate use of the TRAC-BD1/MOD1 code and is considered an adjunct to other documentation provided with the code.

NUREG/CR-4430: CURRENT METHODOLOGIES FOR ASSESSING THE POTENTIAL FOR EARTHQUARKE-INDUCED LIQUEFACTION IN SOILS. KOESTER,J.P.; FRANKLIN,A.G. Army, Dept. of, Army Engineer Waterways Experiment Station. October 1985. 69pp. 8511120176. 33439:087.

The geotechnical engineering literature reflects continuing evolution of methods for evaluation of liquefaction potential, and several significant advances have been achieved in the past few years; notably in the areas of in situ testing and the use of data from past occurrences of liquefaction, strain-based approaches, the steady-state concept, and non-linear, effective stress analysis. In the light of new knowledge and the reexamination of old data, liquefaction occurrence is no longer believed to be restricted to relatively clean, uniform, loosely-deposited, saturated sands, and a great deal of research emphasis has thus been given, or is proposed, to understanding the dynamic behavior of saturated gravelly soils and fine-grained soils with some plasticity. This report discusses conditions under which the potential for earthquake-induced liquefaction should be evaluated, and describes procedures and criteria that are currently applied to assess the liquefaction potential of soils ranging in gradation from gravels to clays. Emphasis is given to several of the more recent field, laboratory, and theoretical approaches.

NUREG/CR-4432: COMPARISON OF DYNAMIC CHARACTERISTICS OF FUKUSHIMA NUCLEAR POWER PLANT CONTAINMENT BUILDING DETERMINED FROM TESTS AND EARTHQUAKES. SRINIVASAN,M.G.; KOT,C.A.; HSIEH,B.J. Argonne National Laboratory. October 1985. 29pp. 8511190570. ANL-85-67. 33534:004.

Modal parameters determined from response measured in dynamic tests and from analytical models for simulating the tests and two subsequent earthquakes experienced by the containment building of Unit 1 of the Fukushima Power Station complex in Japan are compared for the purpose of evaluating the effectiveness of the dynamic tests in earthquake response prediction. The tests are found to have led to the correct identification of a fundamental frequency. The lack of agreement between test- and earthquake-determined modeshapes and damping is attributable more to the shortcomings of the simulation models than to differences in actual behavior.

NUREG/CR-4435: ORGANIC COMPLEXANT-ENHANCED MOBILITY OF TOXIC ELEMENTS IN LOW-LEVEL WASTES.Annual Report,July 1984 - June 1985. SWANSON,J.L. Battelle Memorial Institute, Pacific Northwest Laboratories. December 1985. 68pp. 8601070474. PNL-4965-8. 34186:299.

This report contains the results of the second year's efforts of a project whose objective is to determine how and to what extent organic complexants affect the mobility of toxic elements in subsurface groundwater at commercial low-level waste disposal sites. The complexants EDTA and picolinate, both of which are used in reactor decontamination operations, were studied most extensively. Hydrous ferric oxide, Fe(2)O(3).H(2)O, and kaolinite clay were the soil components most used. Three toxic elements were studied; Ni, Am, and Cd. Ni and Am have radioactive isotopes that are commonly present in commercial low-level wastes, and Cd is an example of a nonradioactive toxic element that might also be in such wastes. A wide diversity of effects of organic complexants on toxic elements sorption was observed. Some complexes are sorbed by soil components at some pH values, but others are not. Important reactions are slow in some systems but rapid in others. There are two separate reactions in which slow kinetics have been observed in some systems; one is the slow dissociation of a preformed complex and the other is the slow desorption by complexant solutions of a previously sorbed uncomplexed element.

NUREG/CR-4437: EXPLORATORY STUDIES OF ELEMENT INTERACTIONS AND COMPOSITION DEPENDENCIES IN RADIATION SENSITIVITY DEVELOPMENT. HAWTHORNE,J.R. Materials Engineering Associates, Inc. November 1985. 94pp. 8512190250. MEA-2113. 34010:068.

This investigation with laboratory melts of pressure vessel steels (A 533-B or A 302-B base) probes suspect interactions between copper impurities and manganese, molybdenum, chromium and nickel alloying as influencing elevated termperature, radiation sensitivity development. The investigation also qualifies the influence of phosphorus content on radiation sensitivity as a function of copper content and explores suspect contributions of tin and arsenic. Radiation resistance is judged on the basis of Charpy-V (Cv) notch ductility before and after 288 degrees centigrade irradiation to 2.5 x 10(19) n/cm(2), E > 1 MeV. The findings demonstrate clearly that important composition interactions exist in radiation sensitivity development.

NUREG/CR-4439: TRAINING REVIEW CRITERIA AND PROCEDURES. * Analysis & Technology, Inc. June 1985. 125pp. 8512310038. 34104:181.

This report provides a set of training review criteria and procedures which constitute a systematic means of implementing two moniotoring functions identified in the "Commission Policy Statement on Training & Qualifications of Nuclear Power Plant Personnel" (March 20, 1984, 50 FR 11147). These functions are: 1. Continuing evaluation of industrywide training & qualification program effectiveness, and 2. monitoring plant and industry trends and events involving personnel errors. The procedures are organized around the five essential elements of performance - based training articulated in the Policy Statement. The package was designed for use by NRC personnel engaged in the review of performance-based training programs in nuclear power plants. It has been published in a modified version in NUREG/CR format to enable large - scale production and distribution for information purposes.

NUREG/CR-4440: A REVIEW OF EMERGENCY DIESEL GENERATOR PERFORMANCE AT NUCLEAR POWER PLANTS. SUBUDHI,M.; HIGGINS,J.C. Brookhaven National Laboratory. November 1985. 35pp. 8512050369. 33768:355.

An evaluation of standby diesel generator performance at nuclear power plants between 1980 and 1983. All diesel generator vendors except Transamerica Delaval were evaluated. Material reviewed included failure data, inspection reports and previous studies by others. Charts and tables of data including manufacturer versus site location. Conclusion is that diesel generator performance and reliability is reasonably good, when TDI experience is factored out. In addition, total loss of offsite power events are decreasing, thus increased inspection activity at diesel generator manufacturers is not required.

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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NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER.(Final Report)

ABRAHAM,T.
NUREG/CR-2482 V09: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period April 1985-September 1985.

ACKERMANN,G.R.
NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.

ADAMS,K.G.
NUREG/CR-4250: VEHICLE BARRIERS:EMPHASIS ON NATURAL FEATURES.

ADAMS,R.E.
NUREG/CR-3830 V02: AEROSOL RELEASE AND TRANSPORT PROGRAM.Semiannual Progress Report For April 1984-September 1984.
NUREG/CR-4255 V01: AEROSAL RELEASE AND TRANSPORT PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.

ADAMS,S.R.
NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.

ADDESSIO,F.L.
NUREG/CR-4278: TRAC-PF1/MOD1 DEVELOPMENT ASSESSMENT.

ADISOMA,G.S.
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

AHMAD,J.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

AHMED,S.
NUREG/CR-4257: INSPECTION,SURVEILLANCE,AND MONITORING OF ELECTRICAL EQUIPMENT INSIDE CONTAINMENT OF NUCLEAR POWER PLANTS--WITH APPLICATIONS TO ELECTRICAL CABLES.

AKERS,D.W.
NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION.
NUREG/CR-4397: IN-PLANT SOURCE TERM MEASUREMENTS AT PRAIRIE ISLAND NUCLEAR GENERATING STATION.

AKGUN,H.
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

ALAMGIR,M.
NUREG/CR-4128: BWR FULL INTEGRAL SIMULATION TEST (FIST) PHASE II TEST RESULTS AND TRAC-BWR MODEL QUALIFICATION.

ALBA,C.
NUREG/CR-4091: THE EFFECT OF ALTERNATIVE AGING AND ACCIDENT SIMULATIONS ON POLYMER PROPERTIES.

ALBERT,M.F.
NUREG/CR-4081: ABSORPTION OF GASEOUS IODINE BY WATER DROPLETS.

ALPERT,D.J.
NUREG/CR-3537: EXPEDIENT METHODS OF RESPIRATORY PROTECTION:III. SUBMICRON PARTICLE TESTS AND SUMMARY OF QUALITY FACTORS.
NUREG/CR-3657: PRELIMINARY SCREENING OF FUEL CYCLE AND BY-PRODUCT MATERIAL LICENSES FOR EMERGENCY PLANNING.
NUREG/CR-4199: A DEMONSTRATION UNCERTAINTY/SENSITIVITY ANALYSIS USING THE HEALTH AND ECONOMIC CONSEQUENCE MODEL CRAC2.
NUREG/CR-4214: HEALTH EFFECTS MODEL FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Part I:Introduction,Integration & Summary.Part II:Scientific Basis For Health Effects Models.

ALVAREZ,J.L.
NUREG/CR-4033: THE ROLE OF PERSONAL AIR SAMPLING IN RADIATION SAFETY PROGRAMS AND RESULTS OF A LABORATORY EVALUATION OF PERSONAL AIR-SAMPLING EQUIPMENT.

AMES,K.R.
NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS.
NUREG/CR-4220: RELIABILITY ANALYSIS OF CONTAINMENT ISOLATION SYSTEMS.

AMICO,P.J.
NUREG/CR-4334: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS.

ANAVIM,E.
NUREG/CR-4050: A REVIEW OF THE SHOREHAM NUCLEAR POWER STATION PROBABILISTIC RISK ASSESSMENT.Internal Events And Core Damage Frequency.

ANDERSEN,J.G.
NUREG/CR-4127 V01: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 1:Numerical Methods.
NUREG/CR-4127 V02: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 2:Models.
NUREG/CR-4127 V03: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 3:Development Assessment For Plant Application.

ANDERSON,B.S.
NUREG/CR-4178 DRFT: AN EVALUATION OF SELECTED LICENSEE EVENT REPORTS PREPARED PURSUANT TO 10 CFR 50.73.Draft Report.

ANDERSON,J.L.
NUREG/CR-4164: DATA REPORT FOR THE TPFL TEE/CRITICAL FLOW EXPERIMENTS.

ANDERSON,R.F.
NUREG/CR-4094: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN SULFATE LAKE ENVIRONMENTS.
NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.

ANDERSON,W.
NUREG-1095: EVALUATION OF RESPONSES TO IE BULLETIN 82-02.Degradation Of Threaded Fasteners In Reactor Coolant Pressure Boundary Of Pressurized Water-Reactor Plants.

ANDRAE,R.W.
NUREG/CR-4260: TORAC USER'S MANUAL.A Computer Code For Analyzing Tornado-Induced Flow And Material Transport In Nuclear Facilities.

ANDREWS,W.B.
NUREG/CR-2800 S03: GUIDELINES FOR NUCLEAR POWER PLANT SAFETY ISSUE PRIORITIZATION INFORMATION DEVELOPMENT.
NUREG/CR-4153: APPLICATIONS OF FOREIGN PROBABILISTIC SAFETY ASSESSMENT EXPERIENCE TO THE U.S. NUCLEAR REGULATORY PROCESS.

ARAGON,J.J.
NUREG/CR-4146: SIMULATION OF AN EPRI-NEVADA TEST SITE (NTS) HYDROGEN BURN TEST AT THE CENTRAL RECEIVER TEST FACILITY.

ARMANTROUT,G.A.
NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt.
NUREG/CR-4239: ANALYSIS OF THE ABILITY OF CURRENT HEALTH PHYSICS INSTRUMENTS TO PREDICT DOSE IN EXPOSED INDIVIDUALS.

ARMSTRONG,G.
NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984.

ARNOLD,W.D.
NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4114: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS.II.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984.
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.

ATTERIDGE,D.G.
NUREG/CR-3613 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Annual Report For 1984.
NUREG/CR-3613 V03 N1: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Semiannual Report For October 1984 Through March 1985.
NUREG/CR-3911 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Quarterly Report,April-June 1984.

AULT,C.H.
NUREG/CR-4117: FAULTING AND JOINTING IN AND NEAR SURFACE MINES OF SOUTHWESTERN INDIANA.

AUSTIN,P.A.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

AUTRY,V.
NUREG/CR-4352: SUGGESTED STATE REQUIREMENTS AND CRITERIA FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE REGULATORY PROGRAM.

AVCI,H.I.
NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS.
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.

AYERS,A.L.
NUREG/CR-3237: CONTROL OF EXPLOSIVE MIXTURES IN PWR WASTE GAS SYSTEMS.

AZARM,M.A.
NUREG/CR-4229: EVALUATION OF CURRENT METHODOLOGY EMPLOYED IN PROBABILISTIC RISK ASSESSMENT (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.
NUREG/CR-4230: PROBABILITY-BASED EVALUATION OF SELECTED FIRE PROTECTION FEATURES IN NUCLEAR POWER PLANTS.
NUREG/CR-4400: THE IMPACT OF MECHANICAL-AND MAINTENANCE-INDUCED FAILURES OF MAIN REACTOR COOLANT PUMP SEALS ON PLANT SAFETY.

BADALAMENTE,R.
NUREG/CR-3817: DEVELOPMENT,USE AND CONTROL OF MAINTENANCE PROCEDURES IN NUCLEAR POWER PLANTS.Problems And Recommendations.

BAILEY,W.J.
NUREG/CR-3950 V01: FUEL PERFORMANCE ANNUAL REPORT FOR 1983.

BAKER,D.A.
NUREG/CR-2850 V03: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1981.

BALDWIN,C.A.
NUREG/CR-4284: NEUTRON EXPOSURE PARAMETERS FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY IRRADIATION SERIES.

BALDWIN,J.S.
NUREG/CR-4403: SUMMARY OF THE WASTE MANAGEMENT PROGRAMS AT URANIUM RECOVERY FACILITIES AS THEY RELATE TO THE 40 CFR PART 192 STANDARDS.

BALL,D.G.
NUREG/CR-3723: STRESS-INTENSITY-FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4249: PRESSURE VESSEL FRACTURE STUDIES PENETRATING TO THE PWR THERMAL-SHOCK ISSUE:EXPERIMENTS TSE-5,TSE-5A AND TSE-6.
NUREG/CR-4304: PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO THE PWR THERMAL-SHOCK ISSUE.Experiment TSE-7.
NUREG/CR-4325: A PARAMETRIC STUDY OF PWR PRESSURE VESSEL INTEGRITY DURING OVERCOOLING ACCIDENTS,CONSIDERING BOTH 2-D AND 3-D FLAWS.

BALL,F.M.
NUREG/CR-4344: INSTRUCTIONAL SKILLS EVALUATION IN NUCLEAR INDUSTRY TRAINING.

BALL,S.J.
NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1-September 30,1984.
NUREG/CR-3885 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, October 1-December 31,1984.
NUREG/CR-4402 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31,1985.

BANDYOPADHYAY
NUREG/CP-0070: PROCEEDINGS OF THE WORKSHOP ON SEISMIC AND DYNAMIC FRAGILITY OF NUCLEAR POWER PLANT COMPONENTS.

BANKOFF,S.G.
NUREG/CR-4414: DIRECT-CONTACT CONDENSATION OF STEAM ON COLD WATER IN STRATIFIED COUNTERCURRENT FLOW.

BANON,H.
NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3663 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 3:Double Ended Guillotine Break Indirectly Induced By Earthquakes.

BARANOWSKI,P.W.
NUREG-1032 DRFT FC: EVALUATION OF STATION BLACKOUT ACCIDENTS AT NUCLEAR POWER PLANTS.Technical Findings Related To Unresolved Safety Issue A-44.Draft Report For Comment.

BARI,R.A.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.

BARLETTA,R.E.
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.
NUREG/CR-3865: EVALUATION OF THE RADIOACTIVE INVENTORY IN,AND ESTIMATION OF ISOTOPIC RELEASE FROM,THE WASTE IN EIGHT TRENCHES AT THE SHEFFIELD LOW-LEVEL WASTE BURIAL SITE.
NUREG/CR-4069: ANALYSES OF SOILS FROM AN AREA ADJACENT TO THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE AT SHEFFIELD,ILLINOIS.
NUREG/CR-4083: ANALYSES OF SOILS FROM THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITES AT BARNWELL,SC AND RICHLAND,WA.
NUREG/CR-4200: BIODEGRADATION TESTING OF SOLIDIFIED LOW-LEVEL WASTE STREAMS.

BARNER,J.O.
NUREG/CR-3999: ELECTRICALLY HEATED EX-REACTOR PELLET-CLADDING INTERACTION (PCI) SIMULATIONS UTILIZING IRRADIATED ZIRCALOY CLADDING.

BARNES,C.R.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

BARNES,M.G.
NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.

BARNES,V.E.
NUREG/CR-3817: DEVELOPMENT,USE AND CONTROL OF MAINTENANCE PROCEDURES IN NUCLEAR POWER PLANTS.Problems And Recommendations.

BARRETT,R.
NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences.

BARTTER,W.D.
NUREG/CR-3626 V02: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: DESCRIPTION OF MODEL CONTENT,STRUCTURE,AND SENSITIVITY TESTING.
NUREG/CR-4104: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL. Field Evaluation/Validation.

BASHAM,P.W.
NUREG/CR-4317 V01: CANADIAN SEISMIC AGREEMENT.Technical Report Covering 1979-1985.

BASS,B.R.
NUREG/CR-3723: STRESS-INTENSITY-FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.

BATTLE,R.E.
NUREG/CR-3991: FAILURE MODES AND EFFECTS ANALYSIS (FMEA) OF THE ICS/NNI ELECTRIC POWER DISTRIBUTION CIRCUITRY AT THE OCONEE-1 NUCLEAR PLANT.
NUREG/CR-3992: COLLECTION AND EVALUATION OF COMPLETE AND PARTIAL LOSSES OF OFF-SITE POWER AT NUCLEAR POWER PLANTS.
NUREG/CR-4347: EMERGENCY DIESEL GENERATOR OPERATING EXPERIENCE,1981-1983.

BAUGH,J.W.
NUREG/CR-4276: VIBRATION AND WEAR IN STEAM GENERATOR TUBES FOLLOWING CHEMICAL CLEANING - SEMIANNUAL REPORT.

BAUM,J.W.
NUREG/CP-0066: PROCEEDINGS OF AN INTERNATIONAL WORKSHOP ON HISTORIC DOSE EXPERIENCE AND DOSE REDUCTION (ALARA) AT NUCLEAR POWER PLANTS,MAY 29-JUNE 1,1984.
NUREG/CR-3469 V02: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS. Annotated Bibliography Of Selected Readings In Radiation Protection And ALARA.
NUREG/CR-4254: OCCUPATIONAL DOSE REDUCTION AND ALARA AT NUCLEAR POWER PLANTS.Study On High-Dose Jobs,Radwaste Handling,And ALARA Incentives.
NUREG/CR-4373: COMPENDIUM OF COST-EFFECTIVENESS EVALUATIONS OF MODIFICATIONS FOR DOSE REDUCTION AT NUCLEAR POWER PLANTS.

BAUMANN,B.L.
NUREG/CR-4090: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Annual Summary Report - Fiscal Year 1984.

BAXTER,D.E.
NUREG/CR-4262 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR.Main Report.
NUREG/CR-4262 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR. Appendices.

BAYBUTT,P.
NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS.
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.

BEAL,S.
NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

BEARE,A.N.
NUREG/CR-4280: THE EFFECTS OF SUPERVISOR EXPERIENCE AND ASSISTANCE OF A SHIFT TECHNICAL ADVISOR (STA) ON CREW PERFORMANCE IN CONTROL ROOM SIMULATORS.

BECKHAM,R.
NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET-GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE,AND SODIUM CHLORIDE.

BECKMAN,R.J.
NUREG/CR-4217: A STATISTICAL ANALYSIS OF NUCLEAR POWER PLANT VALVE FAILURE-RATE VARIABILITY--SOME PRELIMINARY RESULTS.

BEE,R.W.
NUREG/CR-4357: THE FEASIBILITY OF DETECTING THE IMPORT OF UNAUTHORIZED RADIOACTIVE MATERIALS INTO THE UNITED STATES.

BEEBE,M.R.
NUREG-0020 V09 N04: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of March 31,1985.(Gray Book I)
NUREG-0020 V09 N06: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of May 31,1985.(Gray Book I)
NUREG-0020 V09 N07: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of June 30,1985.(Gray Book I)
NUREG-0020 V09 N09: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of August 31,1985.(Gray Book I)
NUREG-0020 V09 N11: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of October 31,1985.(Gray Book I)

BEEDLOW,P.A.
NUREG/CR-4075: DESIGNING PROTECTIVE COVERS FOR URANIUM MILL TAILINGS PILES. A Review.
NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION.

BEHR,V.L.
NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT.

BELL,J.
NUREG-1127: RADIATION PROTECTION TRAINING AT URANIUM HEXAFLUORIDE AND FUEL FABRICATION PLANTS.

BENEDICT,R.
NUREG-1142: TECHNICAL SPECIFICATIONS FOR RIVER BEND STATION.Docket No. 50-458. (Gulf States Utilities Company)
NUREG-1172: TECHNICAL SPECIFICATIONS FOR RIVER BEND STATION. Docket No. 50-458.(Gulf States Utilities Company)

BENNETT,D.E.
NUREG/CR-3657: PRELIMINARY SCREENING OF FUEL CYCLE AND BY-PRODUCT MATERIAL LICENSES FOR EMERGENCY PLANNING.

BENNETT,J.J.
NUREG/CR-3516: A SURVEY OF THE USES OF RADIOACTIVE MATERIALS IN LOUISIANA'S OFFSHORE WATERS.

BENNETT,R.D.
NUREG/CR-3774 V02: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2A:Technical Requirements For Belowground Vault Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V03: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2B:Technical Requirements For Aboveground Vault Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V04: ALTERNATIVE METHOD FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2C:Technical Requirements For Earth Mounded Concrete Bunker Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V05: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2E:Technical Requirements For Shaft Disposal Of Low Level Radioactive Waste.

BENSON,S.M.
NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt.

BERGERON,K.D.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.

BERGGREN,R.G.
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.
NUREG/CR-4092: ORNL CHARACTERIZATION OF HEAVY-SECTION STEEL TECHNOLOGY PROGRAM PLATES 01,02,AND 03.

BEYER,C.E.
NUREG/CR-4168: GT2F:A COMPUTER CODE FOR ESTIMATING LIGHT WATER REACTOR FUEL ROD FAILURES.

BEZLER,P.
NUREG/CR-1677 V02: PIPING BENCHMARK PROBLEMS,VOLUME II DYNAMIC ANALYSIS INDEPENDENT SUPPORT MOTION RESPONSE SPECTRUM METHOD.
NUREG/CR-4291: CONCLUSION AND SUMMARY REPORT ON PHYSICAL BENCHMARKING OF PIPING SYSTEMS.

BHATTACHARYYA
NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS.Application To Establishing Values Of f1 For Soluble Plutonium.

BICKFORD,W.E.
NUREG/CR-2800 S03: GUIDELINES FOR NUCLEAR POWER PLANT SAFETY ISSUE PRIORITIZATION INFORMATION DEVELOPMENT.
NUREG/CR-4385: EFFECTS OF CONTROL SYSTEM FAILURES IN TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A WESTINGHOUSE PRESSURIZED WATER REACTOR.
NUREG/CR-4386: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A BABCOCK AND WILCOX PRESSURIZED WATER REACTOR.
NUREG/CR-4387: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS AND CORE-MELT FREQUENCIES AT A GENERAL ELECTRIC PRESSURIZED WATER REACTOR.

BIHL,N.K.
NUREG/CR-4397: IN-PLANT SOURCE TERM MEASUREMENTS AT PRAIRIE ISLAND NUCLEAR GENERATING STATION.

BILLINGER,G.A.
NUREG/CR-4288: FOCAL MECHANISM ANALYSES FOR VIRGINIA AND EASTERN TENNESSEE EARTHQUAKES (1978-1984).

BINNALL,E.P.
NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt.

BLACKMAN,H.S.
NUREG/CR-3767: INTERACTIVE SIMULATOR EVALUATION FOR CRT-GENERATED DISPLAYS.
NUREG/CR-4040: OPERATIONAL DECISIONMAKING AND ACTION SELECTION UNDER PSYCHOLOGICAL STRESS IN NUCLEAR POWER PLANTS.
NUREG/CR-4272: RESPONSE TREE EVALUATION:EXPERIMENTAL ASSESSMENT OF AN EXPERT SYSTEM FOR NUCLEAR REACTOR OPERATORS.

BLAND,W.M.
NUREG/CR-4271: RECOMMENDED SAFETY,RELIABILITY,QUALITY ASSURANCE AND MANAGEMENT AEROSPACE TECHNIQUES WITH POSSIBLE APPLICATION BY THE DOE TO THE HIGH LEVEL RADIOACTIVE WASTE REPOSITORY PROGRAM.

BLENCOE,J.G.
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.

BLOND,R.M.
NUREG/CR-4197: SAFETY GOAL SENSITIVITY STUDIES.

BLUHM,D.
NUREG/CR-3952: SEQUOYAH EQUIPMENT HATCH SEAL LEAKAGE.

BOARDMAN,T.
NUREG/CR-4294: LEAK RATE ANALYSIS OF THE WESTINGHOUSE REACTOR COOLANT PUMP.

BOCCIO,J.L.
NUREG/CR-4229: EVALUATION OF CURRENT METHODOLOGY EMPLOYED IN PROBABILISTIC RISK ASSESSMENT (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.
NUREG/CR-4230: PROBABILITY-BASED EVALUATION OF SELECTED FIRE PROTECTION FEATURES IN NUCLEAR POWER PLANTS.
NUREG/CR-4231: EVALUATION OF AVAILABLE DATA FOR PROBABILISTIC RISK ASSESSMENTS (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.
NUREG/CR-4400: THE IMPACT OF MECHANICAL-AND MAINTENANCE-INDUCED FAILURES OF MAIN REACTOR COOLANT PUMP SEALS ON PLANT SAFETY.

BOEGEL,A.J.
NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE.

BOHN,M.P.
NUREG/CR-3558: HANDBOOK OF NUCLEAR POWER PLANT SEISMIC FRAGILITIES. Seismic Safety Margins Research Program.
NUREG/CR-4335: POTENTIAL BENEFITS OBTAINED BY REQUIRING SAFTEY-GRADE COLD SHUTDOWN SYSTEMS.

BOLAND,J.F.
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.

BOLANDER,M.A.
NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR.

BOLD,F.C.
NUREG/CR-3413: OFF-SITE CONSEQUENCES OF RADIOLOGICAL ACCIDENTS:METHODS, COSTS AND SCHEDULES FOR DECONTAMINATION.

BOLT,S.E.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.
NUREG/CR-4249: PRESSURE VESSEL FRACTURE STUDIES PENETRATING TO THE PWR THERMAL-SHOCK ISSUE:EXPERIMENTS TSE-5,TSE-5A AND TSE-6.
NUREG/CR-4304: PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO THE PWR THERMAL-SHOCK ISSUE.Experiment TSE-7.

BONZON,L.L.
NUREG/CR-4095: TEST SERIES 2:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E EXIDE FHC-19 BATTERY CELLS.
NUREG/CR-4096: TEST SERIES 3:SEISMIC FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E C&D LCU-13 BATTERY CELLS.
NUREG/CR-4097: TEST SERIES 4:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED EXIDE EMP-13 BATTERY CELLS.

BOOKER,C.P.
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.

BORKOWSKI,R.J.
NUREG/CR-3831: THE IN-PLANT RELIABILITY DATA BASE FOR NUCLEAR PLANT COMPONENTS.Interim Report - Diesel Generators,Batteries, Chargers And Inverters.

BOVELL,R.
NUREG/CR-3887: HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS.

BOWERMAN,B.S.
NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.
NUREG/CR-4406: AN ANALYSIS OF LOW-LEVEL WASTES:Review of Hazardous Waste Regulations And Identification of Radioactive Mixed Wastes. Final Report.

BOWERS,D.L.
NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT.

BOYACK,B.E.
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.
NUREG/CR-4140: DOMINANT ACCIDENT SEQUENCES IN OCONEE-1 PRESSURIZED WATER REACTOR.

BOZARTH,D.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

BRACKENBUSH,L.
NUREG/CR-4297: EXTREMITY MONITORING:Considerations For Use,Dosimeter Placement,And Evaluation.

BRAILE,L.W.
NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983.

BREEDING,R.J.
NUREG/CR-4350 V01: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 1 - PRA Fundamentals.

BREWSTER,C.
NUREG/CR-2482 V08: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period October 1984 - March 1985.
NUREG/CR-2482 V09: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period April-September 1985.

BRODSKY,A.
NUREG-1127: RADIATION PROTECTION TRAINING AT URANIUM HEXAFLUORIDE AND FUEL FABRICATION PLANTS.
NUREG-1134: RADIATION PROTECTION TRAINING FOR PERSONNEL EMPLOYED IN MEDICAL FACILITIES.

BROEK,D.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

BROOKS,B.G.
NUREG-0713 V05: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS - 1983 ANNUAL REPORT.
NUREG-0714 V04-05: OCCUPATIONAL RADIATION EXPOSURE.Fifteenth And Sixteenth Annual Reports,1982 And 1983.

BROOKSHIRE,R.L.
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.

BROWN,J.
NUREG/CR-4190: CALIFORNIA OFFSHORE SURVEY OF LICENSEES USING RADIOACTIVE MATERIAL.

BROWZIN,B.S.
NUREG/CP-0065: TRANSACTIONS OF THE 8TH INTERNATIONAL CONFERENCE ON STRUCTURE MECHANICS IN REACTOR TECHNOLOGY.Panel Session J-K: Status of Research In Structural And Mechanical Engineering For Nuclear Power Plants.

BRUEMMER,S.M.
NUREG/CR-3613 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Annual Report For 1984.
NUREG/CR-3613 V03 N1: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Semiannual Report For October 1984 Through March 1985.
NUREG/CR-3911 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Quarterly Report,April-June 1984.

BRUSKE,S.J.
NUREG/CR-4262 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR.Main Report.
NUREG/CR-4262 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR. Appendices.
NUREG/CR-4326 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Main Report.
NUREG/CR-4326 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Appendices.

BRUST,F.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

BRYAN,R.H.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.

BRYSON,J.W.
NUREG/CR-3723: STRESS-INTENSITY-FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.
NUREG/CR-4367: ORVIRT.PC:A 2-D FINITE-ELEMENT FRACTURE ANALYSIS PROGRAM FOR A MICROCOMPUTER.

BUCHANAN,M.E.
NUREG/CR-4256: MEASUREMENT OF RESPONSE TIME AND DETECTION OF DEGRADATION IN PRESSURE SENSOR/SENSING LINE SYSTEMS.

BUCKALEW,W.H.
NUREG/CR-4147: THE EFFECT OF ENVIRONMENTAL STRESS ON SYLGARD 70 SILICONE ELASTOMER.

BUDNITZ,R.J.
NUREG/CR-4334: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS.

BURKE,R.P.
NUREG/CR-4197: SAFETY GOAL SENSITIVITY STUDIES.

BURNS,E.L.
NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.

BURTT,J.D.
NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR.
NUREG/CR-4115: INTERNATIONAL STANDARD PROBLEM 13 (LOFT EXPERIMENT L2-5). Final Comparison Report.

BUSCHBACH,T.C.
NUREG/CR-4226: NEW MADRID SEISMOTECTONIC STUDY.Activities During Fiscal Year 1983.

BUSLIK,A.J.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.

BUSTARD,L.D.
NUREG/CR-4091: THE EFFECT OF ALTERNATIVE AGING AND ACCIDENT SIMULATIONS ON POLYMER PROPERTIES.

BUXTON,L.D.
NUREG/CR-3802: RELAP5 ASSESSMENT:QUANTITATIVE KEY PARAMETERS AND RUN TIME STATISTICS.
NUREG/CR-4044: TRAC-PF1 LOCA CALCULATIONS USING FINE-NODE AND COARSE-NODE INPUT MODELS.

BYERS,K.R.
NUREG/CR-4298: DESIGN AND INSTALLATION OF COMPUTER SYSTEMS TO MEET THE REQUIREMENTS OF 10 CFR 73.55.

CADDINGTON,P.
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.

CAGLE,R.J.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.

CAMERON,J.R.
NUREG/CR-4131: INVESTIGATION OF ALTERNATIVE MEANS TO ACCOMPLISH THE GOALS OF BIENNIAL ION CHAMBER CALIBRATION.

CAMP,A.L.
NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT.
NUREG/CR-3913: HECTR VERSION 1.0 USER'S MANUAL.

CAMPBELL,A.W.
NUREG/CR-4403: SUMMARY OF THE WASTE MANAGEMENT PROGRAMS AT URANIUM RECOVERY FACILITIES AS THEY RELATE TO THE 40 CFR PART 192 STANDARDS.

CAMPBELL,R.D.
NUREG/CR-3558: HANDBOOK OF NUCLEAR POWER PLANT SEISMIC FRAGILITIES. Seismic Safety Margins Research Program.
NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3663 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 3:Double Ended Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-4290 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF BABCOCK AND WILCOX PWR PLANTS.Volume 2:Guillotine Break Indirectly Induced By Earthquakes.

CANIANO,R.J.
NUREG-1153: INSPECTION REPORT OF UNAUTHORIZED POSSESSION AND USE OF UNSEALED AMERICIUM-241 AND SUBSEQUENT CONFISCATION. J.C. Haynes Company,Newark,Ohio.

CAREW,J.F.
NUREG/CR-4152: AN INDEPENDENT SAFETY ORGANIZATION.

CARFAGNO,S.P.
NUREG/CR-4257: INSPECTION,SURVEILLANCE,AND MONITORING OF ELECTRICAL EQUIPMENT INSIDE CONTAINMENT OF NUCLEAR POWER PLANTS--WITH APPLICATIONS TO ELECTRICAL CABLES.

CARHART,R.A.
NUREG/CR-4120: MATHEMATICAL MODELING OF ULTIMATE HEAT SINK COOLING PONDS.

CARLIN,F.
NUREG/CR-4091: THE EFFECT OF ALTERNATIVE AGING AND ACCIDENT SIMULATIONS ON POLYMER PROPERTIES.

CASADA,M.L.
NUREG/CR-3922 V01: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Main Report And Appendices A And B.
NUREG/CR-3922 V02: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Appendices C And D.

CASE,F.I.
NUREG/CR-4114: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS.II.

CASHMAN,T.
NUREG/CR-4352: SUGGESTED STATE REQUIREMENTS AND CRITERIA FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE REGULATORY PROGRAM.

CASTO,W.R.
NUREG/CR-3551: SAFETY IMPLICATIONS ASSOCIATED WITH IN-PLANT PRESSURIZED GAS STORAGE AND DISTRIBUTION SYSTEMS IN NUCLEAR POWER PLANTS.

CATE,J.H.
NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.

CATHEY,N.G.
NUREG/CR-3301: CATALOG OF PRA DOMINANT ACCIDENT SEQUENCE INFORMATION.

CAZZOLI,E.G.
NUREG/CR-3498: TWO-DIMENSIONAL MODELING OF INTRA-SUBASSEMBLY HEAT TRANSFER AND BUOYANCY-INDUCED FLOW REDISTRIBUTION IN LMFBRS.

CERBONE,R.J.
NUREG/CR-4152: AN INDEPENDENT SAFETY ORGANIZATION.

CHAN,S.F.
NUREG/CR-4201: THERMAL STABILITY TESTING OF LOW-LEVEL WASTE FORMS.
NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA.

CHANDRASEKARAN
NUREG-0017 R01: CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM PRESSURIZED WATER REACTORS (PWR-GALE CODE).

CHANG,M.T.
NUREG/CR-4221: AN EVALUATION OF STRESS CORROSION CRACK GROWTH IN BWR PIPING SYSTEMS.

CHANG,T.Y.
NUREG-1030 DFT FC: SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING NUCLEAR POWER PLANTS.Unresolved Safety Issue A-46.Draft Report For Comment.

CHAO,B.T.
NUREG/CR-3989: TIME- AND VOLUME-AVERAGED CONSERVATION EQUATIONS FOR MULTIPHASE FLOW.Part One:System Without Internal Solid Structures.

CHARBONEAU,B.L.
NUREG/CR-4428: OVERVIEW OF TRAC-BD1/MOD1 ASSESSMENT STUDIES.

CHARLOT,L.A.
NUREG/CR-3613 V03 N1: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Semiannual Report For October 1984 Through March 1985.

CHEN,J.C.
NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT.
NUREG/CR-4331: SIMPLIFIED SEISMIC PROBABILISTIC RISK ASSESSMENT.Procedures And Limitations.

CHEN,T.
NUREG/CR-4351: SUMMARY REPORT FOR LOFT ANTICIPATED TRANSIENT EXPERIMENT SERIES L6-8.

CHENG,H.S.
NUREG/CR-3943: THE BWR PLAN ANALYZER.

CHENION,J.
NUREG/CR-4091: THE EFFECT OF ALTERNATIVE AGING AND ACCIDENT SIMULATIONS ON POLYMER PROPERTIES.

CHEUNG,Y.K.
NUREG/CR-4127 V02: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 2:Models.
NUREG/CR-4127 V03: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 3:Development Assessment For Plant Application.

CHEVERTON,R.D.
NUREG/CR-3723: STRESS-INTENSITY-FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS.

NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4249: PRESSURE VESSEL FRACTURE STUDIES PENETRATING TO THE PWR THERMAL-SHOCK ISSUE:EXPERIMENTS TSE-5,TSE-5A AND TSE-6.
NUREG/CR-4304: PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO THE PWR THERMAL-SHOCK ISSUE.Experiment TSE-7.
NUREG/CR-4325: A PARAMETRIC STUDY OF PWR PRESSURE VESSEL INTEGRITY DURING OVERCOOLING ACCIDENTS,CONSIDERING BOTH 2-D AND 3-D FLAWS.

CHINN,D.J.
NUREG/CR-3660 V04: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 4:Pipe Failure Induced By Crack Growth In West Coast Plants.

CHO,N.Z.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-3485: PRA REVIEW MANUAL.

CHOCKIE,A.D.
NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE.

CHOPRA,O.K.
NUREG/CR-4204: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS:Annual Report,October 1983 - September 1984.

CHOU,C.K.
NUREG/CR-3660 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 1:Summary Report.
NUREG/CR-3663 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 1:Summary Report.
NUREG/CR-4263: RELIABILITY ANALYSIS OF STIFF VERSUS FLEXIBLE PIPING FINAL PROJECT REPORT.

CHRISTENSEN,D.
NUREG/CR-3145 V03: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION - ANNUAL REPORT.(October 1982 - September 1983, Volume 3).

CHU,K.H.
NUREG/CR-4127 V02: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 2:Models.
NUREG/CR-4127 V03: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 3:Development Assessment For Plant Application.

CHUNG,H.M.
NUREG/CR-3980 V04: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,October-December 1984.
NUREG/CR-4204: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS:Annual Report,October 1983 - September 1984.

CLAIRBORNE,H.C.
NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.

CLARK,R.A.
NUREG/CR-4361: STEAM GENERATOR GROUP PROJECT.Annual Report - 1983.
NUREG/CR-4362: STEAM GENERATOR GROUP PROJECT.Annual Report - 1984.

CLAUSER,M.J.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.

CLAUSS,D.B.
NUREG/CR-4137: PRETEST PREDICTIONS FOR THE RESPONSE OF A 1:8-SCALE STEEL LWR CONTAINMENT BUILDING MODEL TO STATIC OVERPRESSURIZATION.
NUREG/CR-4209: COMPARISON OF ANALYTICAL PREDICTIONS AND EXPERIMENTAL RESULTS FOR A 1:8-SCALE STEEL CONTAINMENT MODEL PRESSURIZED TO FAILURE.

CLAYTOR,T.N.
NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984.
NUREG/CR-4368: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS: Semiannual Report,October 1984 - March 1985.

CLEVELAND,J.C.
NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1-September 30,1984.
NUREG/CR-3885 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, October 1-December 31,1984.
NUREG/CR-4402 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31,1985.

CLIFF,W.C.
NUREG/CR-3659: A MATHEMATICAL MODEL FOR ASSESSING THE UNCERTAINTIES OF INSTRUMENTATION MEASUREMENTS FOR POWER AND FLOW OF PWR REACTORS.

CLINE,J.E.
NUREG/CR-4101: ASSAY OF LONG-LIVED RADIONUCLIDES IN LOW-LEVEL WASTES FROM POWER REACTORS.

CLINE,J.F.
NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION.

CLOUGH,R.L.
NUREG/CR-4008: GENERAL EXTRAPOLATION MODEL FOR AN IMPORTANT CHEMICAL DOSE-RATE EFFECT.
NUREG/CR-4358: APPLICATIONS OF DENSITY PROFILING TO EQUIPMENT QUALIFICATION ISSUES.

COE,L.J.
NUREG/CR-4101: ASSAY OF LONG-LIVED RADIONUCLIDES IN LOW-LEVEL WASTES FROM POWER REACTORS.

COHEN,L.
NUREG-0837 V04 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, July-September 1984.
NUREG-0837 V04 N04: NRC TLD DIRECT RADIATION MONITORING REPORT.Progress Report, October-December 1984.
NUREG-0837 V05 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, January-March 1985.
NUREG-0837 V05 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, April-June 1985.

COHEN,S.
NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

COLBERT,J.J.
NUREG/CR-4067: SUMMARY OF BARRIER DEGRADATION EVENTS AND SMALL ACCIDENTS IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.
NUREG/CR-4068: SUMMARY OF HISTORICAL EXPERIENCE WITH RELEASES OF RADIOACTIVE MATERIALS FROM COMMERCIAL NUCLEAR POWER PLANTS IN THE UNITED STATES.

COLEMAN,D.R.
NUREG/CR-3741 V02: EVALUATION OF POWER REACTOR FUEL ROD ANALYSIS CAPABILITIES. Phase 2 Topical Report.Volume 2:Code Evaluation.

COLLIER,R.P.
NUREG/CR-3937: STEAM GENERATOR TUBE RUPTURE IODINE TRANSPORT MECHANISMS. Task 1:Experimental Studies.

COLLINS,J.L.
NUREG/CR-3930: OBSERVED BEHAVIOR OF CESIUM,IODINE,AND TELLURIUM IN THE ORNL FISSION PRODUCT RELEASE PROGRAM.
NUREG/CR-4037: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-5.
NUREG/CR-4043: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-6.

COLMAR,R.
NUREG-0933 S02: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-0933 S03: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

COMER,M.K.
NUREG/CR-3688 V01: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 1:Main Report.
NUREG/CR-3688 V02: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 2:Appendices.
NUREG/CR-4009: HUMAN RELIABILITY DATA BANK.Evaluation Results.
NUREG/CR-4010: SPECIFICATION OF A HUMAN RELIABILITY DATA BANK FOR CONDUCTING HRA SEGMENTS OF PRAS FOR NUCLEAR POWER PLANTS.

COMPERE,E.L.
NUREG/CR-3551: SAFETY IMPLICATIONS ASSOCIATED WITH IN-PLANT PRESSURIZED GAS STORAGE AND DISTRIBUTION SYSTEMS IN NUCLEAR POWER PLANTS.

CONBERE,W.
NUREG/CR-4061: LEACHATE PLUME MIGRATION DOWNGRADIENT FROM URANIUM TAILINGS DISPOSAL IN MINE STOPES.
NUREG/CR-4192: THE ANALYSIS OF DRAINAGE AND CONSOLIDATION AT TYPICAL URANIUM MILL TAILINGS SITES.

CONDIE,K.G.
NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT.

COOPER,D.W.
NUREG/CR-3537: EXPEDIENT METHODS OF RESPIRATORY PROTECTION:II. SUBMICRON PARTICLE TESTS AND SUMMARY OF QUALITY FACTORS.
NUREG/CR-4214: HEALTH EFFECTS MODEL FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Part I:Introduction,Integration & Summary.Part II:Scientific Basis For Health Effects Models.

CORNELL,C.A.
NUREG/CR-4334: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS.

CORNWELL,B.C.
NUREG/CR-3819: SURVEY OF AGED POWER PLANT FACILITIES.

CORWIN,W.R.
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.

COSTANTINO,C.J.
NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS.

COTHAM,B.M.
NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT.

COUNTS,C.A.
NUREG/CR-2800 S03: GUIDELINES FOR NUCLEAR POWER PLANT SAFETY ISSUE PRIORITIZATION INFORMATION DEVELOPMENT.

COUNTS,C.L.
NUREG/CR-4233: DISTRIBUTION OF CORBICULA FLUMINEA AT NUCLEAR FACILITIES.

COVER,L.E.
NUREG/CR-3558: HANDBOOK OF NUCLEAR POWER PLANT SEISMIC FRAGILITIES. Seismic Safety Margins Research Program.

COWAN,C.E.
NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER.(Final Report)

COX,D.C.
NUREG/CR-4350 V06: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 6 - Data Development.

CRAFT,C.M.
NUREG/CR-3863: ASSESSMENT OF CLASS 1E PRESSURE TRANSMITTER RESPONSE WHEN SUBJECTED TO HARSH ENVIRONMENT SCREENING TESTS.

CROFF,A.G.
NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.

CRONEY,S.T.
NUREG/CR-4181: LEACHABILITY OF RADIONUCLIDES FROM CEMENT SOLIDIFIED WASTE FORMS PRODUCED AT OPERATING NUCLEAR POWER REACTORS.
NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION.
NUREG/CR-4397: IN-PLANT SOURCE TERM MEASUREMENTS AT PRAIRIE ISLAND NUCLEAR GENERATING STATION.

CULLEN,W.
NUREG/CR-4422: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN PWR ENVIRONMENTS.

CULLEN,W.H.
NUREG/CR-3945: FATIGUE CRACK GROWTH RATES OF LOW-CARBON AND STAINLESS PIPING STEELS IN PWR ENVIRONMENT.
NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS.

CUMMINGS,F.M.
NUREG/CR-3609: EVALUATION OF NEUTRON DOSIMETRY TECHNIQUES FOR WELL-LOGGING OPERATIONS.

CURRIE,J.W.
NUREG/CR-3413: OFF-SITE CONSEQUENCES OF RADIOLOGICAL ACCIDENTS:METHODS, COSTS AND SCHEDULES FOR DECONTAMINATION.

CUTSHALL,N.H.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.

DAEMEN,J.J.
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

DAKE,L.S.
NUREG/CR-3915: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL.

DALING,P.M.
NUREG/CR-4070 V02: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Volume 2:Current Status Of Biofouling Surveillance And Control Techniques.
NUREG/CR-4070 V03: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Factors That May Intensify The Safety Consequences Of Biofouling.

DALY,B.J.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

DANDINI,V.J.
NUREG/CR-3954: HECTR ANALYSIS OF EQUIPMENT TEMPERATURE RESPONSES TO SELECTED HYDROGEN BURNS IN AN ICE CONDENSER CONTAINMENT.
NUREG/CR-4146: SIMULATION OF AN EPRI-NEVADA TEST SITE (NTS) HYDROGEN BURN TEST AT THE CENTRAL RECEIVER TEST FACILITY.

DANIELSON,W.F.
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.

DATTA,A.
NUREG-1148: NUCLEAR POWER PLANT FIRE PROTECTION RESEARCH PROGRAM.

DAVIS,C.B.
NUREG/CR-3935: THERMAL-HYDRAULIC ANALYSES OF OVERCOOLING SEQUENCES FOR THE H.B. ROBINSON UNIT 2 PRESSURIZED THERMAL SHOCK STUDY.

NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR.
NUREG/CR-4326 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Main Report.
NUREG/CR-4326 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Appendices.

DAVIS,L.A.
NUREG/CR-3901: DOCUMENTATION AND USER'S GUIDE:GS2 & GS3 - VARIABLY SATURATED FLOW AND MASS TRANSPORT MODELS.

DAVIS,L.T.
NUREG/CR-4258: AN APPROACH TO TEAM SKILLS TRAINING OF NUCLEAR POWER PLANT CONTROL ROOM CREWS.

DAVIS,M.S.
NUREG/CR-2482 V08: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period October 1984 - March 1985.
NUREG/CR-2482 V09: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period April-September 1985.
NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.

DAVIS,R.E.
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.

DAVIS,T.C.
NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.
NUREG/CR-4377: EVALUATIONS AND UTILIZATIONS OF RISK IMPORTANCES.

DAWSON,J.F.
NUREG/CR-3915: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL.

DE JARLAIS,G.
NUREG/CR-4277: INVERTED ANNUAL FLOW EXPERIMENTAL STUDY.

DEAN,R.S.
NUREG/CR-3791: CLOSEOUT OF IE BULLETIN 79-09:FAILURE OF GE TYPE AK-2 CIRCUIT BREAKERS IN SAFETY-RELATED SYSTEMS.
NUREG/CR-4003: CLOSEOUT OF IE BULLETIN 79-04:INCORRECT WEIGHTS FOR SWING CHECK VALVES MANUFACTURED BY VELAN ENGINEERING CORPORATION.
NUREG/CR-4004: CLOSEOUT OF IE BULLETIN 79-25:FAILURES OF WESTINGHOUSE BFD RELAYS IN SAFETY-RELATED SYSTEMS.
NUREG/CR-4005: CLOSEOUT OF IE BULLETIN 80-12:DECAY HEAT REMOVAL SYSTEM OPERABILITY.
NUREG/CR-4006: CLOSEOUT OF IE BULLETIN 81-01:SURVEILLANCE OF MECHANICAL SNUBBERS.

DEARING,J.F.
NUREG/CR-4140: DOMINANT ACCIDENT SEQUENCES IN OCONEE-1 PRESSURIZED WATER REACTOR.

DEBEVEC,C.J.
NUREG-0905: CLOSEOUT OF IE BULLETIN 79-12:SHORT-PERIOD SCRAMS AT BOILING-WATER REACTORS.

DECK,D.L.
NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.

DEEDS,W.E.
NUREG/CR-3949 V01: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM: Semiannual Progress Report For Period Ending June 30,1984.
NUREG/CR-3949 V02: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM: Annual Progress Report For Period Ending December 31,1984.

DEFIELD,J.D.
NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET-GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE,AND SODIUM CHLORIDE.

DEITZ,V.R.
NUREG/CR-3990: CHARCOAL PERFORMANCE UNDER ACCIDENT CONDITIONS IN LIGHT-WATER REACTORS.

DENHAM,D.H.
NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.

DENNEHY,T.G.
NUREG/CR-4365: DESIGN AND DEVELOPMENT OF A SPECIAL PURPOSE SAFT SYSTEM FOR NONDESTRUCTIVE EVALUATION OF NUCLEAR REACTOR VESSELS AND PIPING COMPONENTS.

DENNING,R.S.
NUREG/CR-3937: STEAM GENERATOR TUBE RUPTURE IODINE TRANSPORT MECHANISMS. Task 1:Experimental Studies.

DEUTSCH,W.J.
NUREG/CR-3709: METHODS OF MINIMIZING GROUND-WATER CONTAMINATION FROM IN SITU LEACH URANIUM MINING.Final Report.

DEWALL,K.G.
NUREG/CR-3819: SURVEY OF AGED POWER PLANT FACILITIES.

DEWERD,L.A.
NUREG/CR-4131: INVESTIGATION OF ALTERNATIVE MEANS TO ACCOMPLISH THE GOALS OF BIENNIAL ION CHAMBER CALIBRATION.

DEWEY,J.W.
NUREG/C[illegible]339: A REVIEW OF RECENT RESEARCH ON THE SEISMOTECTONICS OF THE SOUTHEASTERN SEABOARD AND AN EVALUATION OF HYPOTHESES ON THE SOURCE OF THE 1886 CHARLESTON,SOUTH CAROLINA EARTHQUAKE.

DHOOGE,N.J.
NUREG/CR-4358: APPLICATIONS OF DENSITY PROFILING TO EQUIPMENT QUALIFICATION ISSUES.

DIAMENT,H.
NUREG/CR-3442: RADTWO:A COMPUTER CODE FOR SIMULATING FAST-TRANSIENT, TWO-DIMENSIONAL,TWO-LAYER RADIONUCLIDE CONCENTRATION CONDITIONS IN LAKES,RESERVOIRS,RIVERS,ESTUARIES,AND COASTAL REGIONS.

DICK,C.E.
NUREG/CR-4266: STANDARD BETA-PARTICLE AND MONOENERGETIC ELECTRON SOURCES FOR THE CALIBRATION OF BETA-RADIATION PROTECTION INSTRUMENTATION.

DICKSON,C.R.
NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.

DILBER,I.
NUREG/CR-4415: COUNTER-CURRENT STEAM/WATER FLOW ABOVE A PERFORATED PLATE-VERTICAL INJECTION OF WATER.

DINGMAN,S.E.
NUREG/CR-3913: HECTR VERSION 1.0 USER'S MANUAL.

DIONNE,B.J.
NUREG/CP-0066: PROCEEDINGS OF AN INTERNATIONAL WORKSHOP ON HISTORIC DOSE EXPERIENCE AND DOSE REDUCTION (ALARA) AT NUCLEAR POWER PLANTS,MAY 29-JUNE 1,1984.
NUREG/CR-4254: OCCUPATIONAL DOSE REDUCTION AND ALARA AT NUCLEAR POWER PLANTS.Study On High-Dose Jobs,Radwaste Handling,And ALARA Incentives.

DISALVO,R.
NUREG/CR-4177 V01: MANAGEMENT OF SEVERE ACCIDENTS.Perspectives On Managing Severe Accidents In Commercial Nuclear Power Plants.
NUREG/CR-4177 V02: MANAGEMENT OF SEVERE ACCIDENTS.Extending Plant Operating Procedures Into The Severe Accident Regime.

DOBRANICH,D.
NUREG/CR-4044: TRAC-PF1 LOCA CALCULATIONS USING FINE-NODE AND COARSE-NODE INPUT MODELS.

NUREG/CR-4155: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:NORTHWESTERN UNIVERSITY PERFORATED-PLATE CCFL TESTS.
NUREG/CR-4189: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT.Semiscale MOD-2A Feedwater-Line Break (S-SF-3) And Steam-Line Break (S-SF-5) Tests.

DOCTOR,P.
NUREG/CP-0063: PROCEEDINGS OF THE 1984 STATISTICAL SYMPOSIUM ON NATIONAL ENERGY ISSUES.

DODD,C.V.
NUREG/CR-3949 V01: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Semiannual Progress Report For Period Ending June 30,1984.
NUREG/CR-3949 V02: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Annual Progress Report For Period Ending December 31,1984.

DODSON,K.E.
NUREG/CR-3514 V02: THE CHEMICAL BEHAVIOR OF IODINE IN AQUEOUS SOLUTIONS UP TO 150 C.II.Radiation-Redox Conditions.

DODSON,M.E.
NUREG/CR-3906: URANIUM MILL TAILINGS NEUTRALIZATION:CONTAMINANT COMPLEXATION AND TAILINGS LEACHING STUDY.
NUREG/CR-4259: TAILINGS NEUTRALIZATION AND OTHER ALTERNATIVES FOR IMMOBILIZING TOXIC MATERIALS IN TAILINGS.Final Report.

DOERGE,D.H.
NUREG/CR-4090: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Annual Summary Report - Fiscal Year 1984.

DOESBURG,J.M.
NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS.

DOLAN,F.X.
NUREG/CR-3426 V01: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY. Facility And Test Design Report.
NUREG/CR-3426 V02: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY.Data Report.

DONOVAN,M.D.
NUREG/CR-4009: HUMAN RELIABILITY DATA BANK.Evaluation Results.
NUREG/CR-4010: SPECIFICATION OF A HUMAN RELIABILITY DATA BANK FOR CONDUCTING HRA SEGMENTS OF PRAS FOR NUCLEAR POWER PLANTS.
NUREG/CR-4280: THE EFFECTS OF SUPERVISOR EXPERIENCE AND ASSISTANCE OF A SHIFT TECHNICAL ADVISOR (STA) ON CREW PERFORMANCE IN CONTROL ROOM SIMULATORS.

DORNSIFE,B.
NUREG/CR-4352: SUGGESTED STATE REQUIREMENTS AND CRITERIA FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE REGULATORY PROGRAM.

DOUGHERTY,D.R.
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.
NUREG/CR-4062: EXTENDED STORAGE OF LOW-LEVEL RADIOACTIVE WASTES:Potential Problem Areas.
NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA.

DOUGLAS,S.C.
NUREG/CR-3197 V01: REACTION BETWEEN SOME CESIUM-IODINE COMPOUNDS AND THE REACTOR MATERIALS 304 STAINLESS STEEL,INCONEL 600 & SILVER. Volume I:Cesium Hydroxide Reactions.

DOVE,R.C.
NUREG/CR-4274: ANALYSIS AND TESTS ON SMALL-SCALE SHEAR WALLS FY-82 FINAL REPORT.

DRAKE,J.B.
NUREG/CR-3723: STRESS-INTENSITY-FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS.

DRISCOLL,J.W.
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.

DUCE,S.W.
NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION.

DUDA,P.M.
NUREG/CR-2718: STEAM EXPLOSION EXPERIMENTS WITH SINGLE DROPS OF IRON OXIDE MELTED WITH A CO2 LASER.Part II:Parametric Studies.

DUKLER,A.E.
NUREG/CR-4424: DROPLET SIZES,DYNAMICS AND DEPOSITION IN VERTICAL ANNULAR FLOW.

DUNENFELD,M.S.
NUREG/CR-3950 V01: FUEL PERFORMANCE ANNUAL REPORT FOR 1983.

DUNKERLY,S.J.
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.

DUNN,W.E.
NUREG/CR-4120: MATHEMATICAL MODELING OF ULTIMATE HEAT SINK COOLING PONDS.

DUNNING,D.E.
NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE.

DUNWOODY,W.E.
NUREG/CR-4274: ANALYSIS AND TESTS ON SMALL-SCALE SHEAR WALLS FY-82 FINAL REPORT.

DURBIN,P.W.
NUREG/CR-4355 V01: 238 PU(IV) IN MONKEYS.Overview Of Metabolism.

EARY,L.E.
NUREG/CR-3709: METHODS OF MINIMIZING GROUND-WATER CONTAMINATION FROM IN SITU LEACH URANIUM MINING.Final Report.

EASLEY,P.
NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences.

EBEL,J.E.
NUREG/CR-4354: A STUDY OF SEISMICITY AND TECTONICS IN NEW ENGLAND.Final Report.

EBERHARDT,L.L.
NUREG/CR-4268: RATIO METHODS FOR COST-EFFECTIVE FIELD SAMPLING OF COMMERCIAL RADIOACTIVE LOW-LEVEL WASTES.

EDLER,S.K.
NUREG/CR-3810 V03: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,July-September 1984.
NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,October-December 1984.
NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,January-March 1985.
NUREG/CR-4318 V02: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,April-June 1985.

EDMONDS,D.P.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.

EDSON,J.L.
NUREG/CR-4080: DETERMINATION OF THE AVAILABILITY OF CORE EXIT THERMOCOUPLES DURING SEVERE ACCIDENT SITUATIONS.

EHRLICH,M.
NUREG/CR-4266: STANDARD BETA-PARTICLE AND MONOENERGETIC ELECTRON SOURCES FOR THE CALIBRATION OF BETA-RADIATION PROTECTION INSTRUMENTATION.

EIDSON,A.F.
NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984.

EISSENBERG,D.M.
NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.

EL-BASSIONI,A.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-3485: PRA REVIEW MANUAL.

EL-SHINAWY,R.M.
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.

ELDER,H.K.
NUREG/CR-3293 V01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Main Report.
NUREG/CR-3293 V02: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Appendices.

ELLINGWOOD,B.
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.

ELMORE,M.R.
NUREG/CR-4089: EVALUATION OF FIELD-TESTED FUGITIVE DUST CONTROL TECHNIQUES FOR URANIUM MILL TAILINGS PILES.

ELRICK,R.M.
NUREG/CR-3197 V01: REACTION BETWEEN SOME CESIUM-IODINE COMPOUNDS AND THE REACTOR MATERIALS 304 STAINLESS STEEL,INCONEL 600 & SILVER. Volume I:Cesium Hydroxide Reactions.

EMBREY,D.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

EMBREY,D.E.
NUREG/CR-4016 V01: APPLICATION OF SLIM-MAUD:A TEST OF AN INTERACTIVE COMPUTER-BASED METHOD FOR ORGANIZING EXPERT ASSESSMENT OF HUMAN PERFORMANCE AND RELIABILITY.Volume I:Main Report.

EMEIGH,C.W.
NUREG-1065 R01: ACCEPTANCE CRITERIA FOR THE LOW ENRICHED URANIUM REFORM AMENDMENTS.

EMERSON,E.L.
NUREG/CR-3611: RADIOACTIVE MATERIAL (RAM) ACCIDENT/INCIDENT DATA ANALYSIS PROGRAM.

EMRIT,R.
NUREG-0933 S02: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-0933 S03: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

ENDEBROCK,E.G.
NUREG/CR-4274: ANALYSIS AND TESTS ON SMALL-SCALE SHEAR WALLS FY-82 FINAL REPORT.

ENDERLIN,W.I.
NUREG/CR-4276: VIBRATION AND WEAR IN STEAM GENERATOR TUBES FOLLOWING CHEMICAL CLEANING - SEMIANNUAL REPORT.

ENDRES,G.W.
NUREG/CR-3609: EVALUATION OF NEUTRON DOSIMETRY TECHNIQUES FOR WELL-LOGGING OPERATIONS.

ERASLAN,A.H.
NUREG/CR-3442: RADTWO:A COMPUTER CODE FOR SIMULATING FAST-TRANSIENT, TWO-DIMENSIONAL,TWO-LAYER RADIONUCLIDE CONCENTRATION CONDITIONS IN LAKES,RESERVOIRS,RIVERS,ESTUARIES,AND COASTAL REGIONS.

ETTINGER,H.J.
NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET-GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE,AND SODIUM CHLORIDE.

EVANS,D.D.
NUREG-1046: DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN UNSATURATED Z0NE:TECHNICAL CONSIDERATIONS AND RESPONSE TO COMMENTS.
NUREG/CR-4042: A 3-DIMENSIONAL COMPUTER MODEL TO SIMULATE FLUID FLOW AND CONTAINMENT TRANSPORT THROUGH A ROCK FRACTURE SYSTEM.

EVANS,J.S.
NUREG/CR-4214: HEALTH EFFECTS MODEL FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Part I:Introduction,Integration & Summary.Part II:Scientific Basis For Health Effects Models.

FALETTI,D.W.
NUREG/CR-41̈51: INTEGRATION OF EMERGENCY ACTION LEVELS WITH COMBUSTION ENGINEERING EMERGENCY OPERATING PROCEDURES.By Use Of Combustion Engineering Owners Group Emergency Operating Procedure Technical Guidelines.

FANDUS,F.
NUREG/CR-3952: SEQUOYAH EQUIPMENT HATCH SEAL LEAKAGE.

FAYER,M.J.
NUREG/CR-3972: SETTLEMENT OF URANIUM MILL TAILINGS PILES: A COMPARISON OF ANALYSIS TECHNIQUES.
NUREG/CR-4087: MEASUREMENTS OF URANIUM MILL TAILINGS CONSOLIDATION CHARACTERISTICS.
NUREG/CR-4192: THE ANALYSIS OF DRAINAGE AND CONSOLIDATION AT TYPICAL URANIUM MILL TAILINGS SITES.

FEHRINGER,D.J.
NUREG-0946: AN EVALUATION OF RADIONUCLIDE CONCENTRATIONS IN HIGH-LEVEL RADIOACTIVE WASTES.

FENTON,D.L.
NUREG/CR-4264: INVESTIGATION ON HIGH-EFFICIENCY PARTICULATE AIR FILTER PLUGGING BY COMBUSTION AEROSOLS.
NUREG/CR-4321: FULL SCALE MEASUREMENTS OF SMOKE TRANSPORT AND DEPOSITION IN VENTILATION SYSTEM DUCTWORK.

FINDLAY,J.A.
NUREG/CR-4128: BWR FULL INTEGRAL SIMULATION TEST (FIST) PHASE II TEST RESULTS AND TRAC-BWR MODEL QUALIFICATION.

FINEMAN,C.P.
NUREG/CR-4262 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR.Main Report.
NUREG/CR-4262 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR. Appendices.

FINLAYSON,F.C.
NUREG/CR-4133: NUCLEAR POWER SAFETY REPORTING SYSTEM IMPLEMENTATION AND OPERATIONAL SPECIFICATIONS.

FISCHER,L.E.
NUREG/CR-3854: FABRICATION CRITERIA FOR SHIPPING CONTAINERS.

FITZSIMMONS,D.
NUREG/CR-3999: ELECTRICALLY HEATED EX-REACTOR PELLET-CLADDING INTERACTION (PCI) SIMULATIONS UTILIZING IRRADIATED ZIRCALOY CLADDING.

FLANAGAN,G.F.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

FLANIGAN,L.J.
NUREG/CR-3937: STEAM GENERATOR TUBE RUPTURE IODINE TRANSPORT MECHANISMS. Task 1:Experimental Studies.

FLETCHER,C.D.
NUREG/CR-3935: THERMAL-HYDRAULIC ANALYSES OF OVERCOOLING SEQUENCES FOR THE H.B. ROBINSON UNIT 2 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR.

FLUCKIGER,J.D.
NUREG/CR-4298: DESIGN AND INSTALLATION OF COMPUTER SYSTEMS TO MEET THE REQUIREMENTS OF 10 CFR 73.55.

FOLEY,M.G.
NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS.

FOLEY,W.J.
NUREG/CR-3791: CLOSEOUT OF IE BULLETIN 79-09:FAILURE OF GE TYPE AK-2 CIRCUIT BREAKERS IN SAFETY-RELATED SYSTEMS.
NUREG/CR-3794: CLOSEOUT OF IE BULLETIN 80-25:OPERATING PROBLEMS WITH TARGET ROCK SAFETY-RELIEF VALVES AT BWRS.
NUREG/CR-4003: CLOSEOUT OF IE BULLETIN 79-04:INCORRECT WEIGHTS FOR SWING CHECK VALVES MANUFACTURED BY VELAN ENGINEERING CORPORATION.
NUREG/CR-4004: CLOSEOUT OF IE BULLETIN 79-25:FAILURES OF WESTINGHOUSE BFD RELAYS IN SAFETY-RELATED SYSTEMS.
NUREG/CR-4005: CLOSEOUT OF IE BULLETIN 80-12:DECAY HEAT REMOVAL SYSTEM OPERABILITY.
NUREG/CR-4006: CLOSEOUT OF IE BULLETIN 81-01:SURVEILLANCE OF MECHANICAL SNUBBERS.

FRAGOLA,J.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.

FRANKLIN,A.G.
NUREG/CR-4430: CURRENT METHODOLOGIES FOR ASSESSING THE POTENTIAL FOR EARTHQUAKE-INDUCED LIQUEFACTION IN SOILS.

FREEMAN,H.D.
NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION.

FREEMAN-KELLY
NUREG/CR-4172: A USER'S GUIDE FOR MERGE.

FRESCO,A.
NUREG/CR-4228: REVIEW OF THE VOGTLE UNITS 1 AND 2 AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS.

FRUCHTER,J.S.
NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER.(Final Report)

FUENKAJORN,K.
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

FULLER,L.C.
NUREG/CR-3764: BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE.

FURGAL,D.T.
NUREG/CR-3863: ASSESSMENT OF CLASS 1E PRESSURE TRANSMITTER RESPONSE WHEN SUBJECTED TO HARSH ENVIRONMENT SCREENING TESTS.

GABETTA,G.
NUREG/CR-4422: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN PWR ENVIRONMENTS.

GADDY,C.D.
NUREG/CR-3688 V01: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 1:Main Report.
NUREG/CR-3688 V02: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 2:Appendices.
NUREG/CR-4009: HUMAN RELIABILITY DATA BANK.Evaluation Results.
NUREG/CR-4258: AN APPROACH TO TEAM SKILLS TRAINING OF NUCLEAR POWER PLANT CONTROL ROOM CREWS.

GALLAHER,R.B.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.

GALLUCCI,R.H.
NUREG/CR-2800 S03: GUIDELINES FOR NUCLEAR POWER PLANT SAFETY ISSUE PRIORITIZATION INFORMATION DEVELOPMENT.
NUREG/CR-4220: RELIABILITY ANALYSIS OF CONTAINMENT ISOLATION SYSTEMS.

GALLUP,D.R.
NUREG/CR-4335: POTENTIAL BENEFITS OBTAINED BY REQUIRING SAFTEY-GRADE COLD SHUTDOWN SYSTEMS.

GANAPATHY,S.
NUREG/CR-4365: DESIGN AND DEVELOPMENT OF A SPECIAL PURPOSE SAFT SYSTEM FOR NONDESTRUCTIVE EVALUATION OF NUCLEAR REACTOR VESSELS AND PIPING COMPONENTS.

GANTI,C.S.
NUREG/CR-4231: EVALUATION OF AVAILABLE DATA FOR PROBABILISTIC RISK ASSESSMENTS (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.

GAUSE,E.
NUREG/CR-2482 V08: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period October 1984 - March 1985.

GAUSSENS,G.
NUREG/CR-4091: THE EFFECT OF ALTERNATIVE AGING AND ACCIDENT SIMULATIONS ON POLYMER PROPERTIES.

GAVIN,P.
NUREG/CR-4120: MATHEMATICAL MODELING OF ULTIMATE HEAT SINK COOLING PONDS.

GEE,G.W.
NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION.

GENTILI,H.
NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT.

GENTILLON,C.D.
NUREG/CR-3862: DEVELOPMENT OF TRANSIENT INITIATING EVENT FREQUENCIES FOR USE IN PROBABILISTIC RISK ASSESSMENTS.
NUREG/CR-4071: EXPLORATORY TREND AND PATTERN ANALYSIS FOR 1981 LICENSEE EVENT REPORT DATA.

GERDING,T.J.
NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT.

GERGELY,P.
NUREG/CR-4123: SEISMIC FRAGILITY OF REINFORCED CONCRETE STRUCTURES AND COMPONENTS FOR APPLICATION TO NUCLEAR FACILITIES.

GERTMAN,D.I.
NUREG/CR-4040: OPERATIONAL DECISIONMAKING AND ACTION SELECTION UNDER PSYCHOLOGICAL STRESS IN NUCLEAR POWER PLANTS.

GHERSON,P.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

GIESEKE,J.A.
NUREG/CR-3937: STEAM GENERATOR TUBE RUPTURE IODINE TRANSPORT MECHANISMS. Task 1:Experimental Studies.

GILLEN,D.
NUREG/CR-4403: SUMMARY OF THE WASTE MANAGEMENT PROGRAMS AT URANIUM RECOVERY FACILITIES AS THEY RELATE TO THE 40 CFR PART 192 STANDARDS.

GILLEN,K.T.
NUREG/CR-4008: GENERAL EXTRAPOLATION MODEL FOR AN IMPORTANT CHEMICAL DOSE-RATE EFFECT.
NUREG/CR-4358: APPLICATIONS OF DENSITY PROFILING TO EQUIPMENT QUALIFICATION ISSUES.

GILMORE,W.E.
NUREG/CR-3767: INTERACTIVE SIMULATOR EVALUATION FOR CRT-GENERATED DISPLAYS.
NUREG/CR-4227: HUMAN ENGINEERING GUIDELINES FOR THE EVALUATION AND ASSESSMENT OF VIDEO DISPLAY UNITS.

GIRVIN,D.C.
NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER.(Final Report)

GLISSMEYER,J.A.
NUREG/CR-4088: METHODS FOR ESTIMATING RADIOACTIVE AND TOXIC AIRBORNE SOURCE TERMS FOR URANIUM MILLING OPERATIONS.

GODFREY,D.
NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

GOETSCH,S.J.
NUREG/CR-4131: INVESTIGATION OF ALTERNATIVE MEANS TO ACCOMPLISH THE GOALS OF BIENNIAL ION CHAMBER CALIBRATION.

GOLDIN,D.
NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

GOLDMAN,A.S.
NUREG/CR-4107: SEQUENTIAL TEST PROCEDURES FOR DETECTING PROTRACTED MATERIALS LOSSES.

GOODWIN,G.M.
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.

GORDON,J.
NUREG/CR-4357: THE FEASIBILITY OF DETECTING THE IMPORT OF UNAUTHORIZED RADIOACTIVE MATERIALS INTO THE UNITED STATES.

GOTTULA,R.C.
NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT.

GRAHAM,E.D.
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.

GRAY,L.H.
NUREG/CR-4280: THE EFFECTS OF SUPERVISOR EXPERIENCE AND ASSISTANCE OF A SHIFT TECHNICAL ADVISOR (STA) ON CREW PERFORMANCE IN CONTROL ROOM SIMULATORS.

GREEN,N.M.
NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide.

GREENSTREET,W.
NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.

GREER,W.B.
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

GREGORY,W.S.
NUREG/CR-4225: SUMMARY OF EFFICIENCY TESTING OF STANDARD AND HIGH-CAPACITY HIGH-EFFICIENCY PARTICULATE AIR FILTERS SUBJECTED TO SIMULATED TORNADO DEPRESSURIZATION AND EXPLOSIVE SHOCK WAVES.
NUREG/CR-4232: THE RESPONSE OF VENTILATION DAMPERS TO LARGE AIRFLOW PULSES.
NUREG/CR-4260: TORAC USER'S MANUAL.A Computer Code For Analyzing Tornado-Induced Flow And Material Transport In Nuclear Facilities.
NUREG/CR-4264: INVESTIGATION ON HIGH-EFFICIENCY PARTICULATE AIR FILTER PLUGGING BY COMBUSTION AEROSOLS.

GREIMANN,L.
NUREG/CR-3952: SEQUOYAH EQUIPMENT HATCH SEAL LEAKAGE.

GRIESS,J.C.
NUREG/CR-4124: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.

GRIFFITH,P.
NUREG/CR-4376: HEAT TRANSFER,CARRYOVER AND FALL BACK IN PWR STEAM GENERATORS DURING TRANSIENTS.

GRONEMYER,L.
NUREG/CR-4352: SUGGESTED STATE REQUIREMENTS AND CRITERIA FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE REGULATORY PROGRAM.

GUNAJI,M.V.
NUREG/CR-4264: INVESTIGATION ON HIGH-EFFICIENCY PARTICULATE AIR FILTER PLUGGING BY COMBUSTION AEROSOLS.

GUNDERSEN,G.E.
NUREG-1065 R01: ACCEPTANCE CRITERIA FOR THE LOW ENRICHED URANIUM REFORM AMENDMENTS.

GUPPY,J.G.
NUREG/CR-4152: AN INDEPENDENT SAFETY ORGANIZATION.

GUYMON,R.H.
NUREG/CR-3551: SAFETY IMPLICATIONS ASSOCIATED WITH IN-PLANT PRESSURIZED GAS STORAGE AND DISTRIBUTION SYSTEMS IN NUCLEAR POWER PLANTS.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.

GUZOWSKI,R.V.
NUREG/CR-4110: REPOSITORY SITE DATA REPORT FOR UNSATURATED TUFF,YUCCA MOUNTAIN,NEVADA.

HAAS,P.M.
NUREG/CR-3887: HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS.

HAASL,D.
NUREG/CR-4350 V04: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 4 - System Reliability And Analysis Techniques,Sessions B/C - Event Trees/Fault Trees.

HABERMAN,J.H.
NUREG/CR-4218: LOCA SIMULATION IN THE NATIONAL RESEARCH UNIVERSAL REACTOR PROGRAM.Postirradiation Examination Results For The Third Materials Test (MT-3) - Second Campaign.

HACKBARTH,C.J.
NUREG-1046: DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN UNSATURATED ZONE:TECHNICAL CONSIDERATIONS AND RESPONSE TO COMMENTS.

HAGGARD,D.L.
NUREG/CR-3609: EVALUATION OF NEUTRON DOSIMETRY TECHNIQUES FOR WELL-LOGGING OPERATIONS.

HALL,R.E.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.
NUREG/CR-4152: AN INDEPENDENT SAFETY ORGANIZATION.

HALL,W.J.
NUREG/CR-4334: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS.

HAMMOND,R.A.
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.

HANAN,N.
NUREG/CR-3485: PRA REVIEW MANUAL.
NUREG/CR-4050: A REVIEW OF THE SHOREHAM NUCLEAR POWER STATION PROBABILISTIC RISK ASSESSMENT.Internal Events And Core Damage Frequency.

HANEY,L.N.
NUREG/CR-4040: OPERATIONAL DECISIONMAKING AND ACTION SELECTION UNDER PSYCHOLOGICAL STRESS IN NUCLEAR POWER PLANTS.

HANNINEN,H.
NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS.
NUREG/CR-4422: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN PWR ENVIRONMENTS.

HANSON,R.G.
NUREG/CR-4262 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR.Main Report.
NUREG/CR-4262 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR. Appendices.
NUREG/CR-4429: TRAC-BD1/MOD1 USERS GUIDELINE.

HARDY,H.A.
NUREG/CR-2531 R03: INTRODUCTORY USER'S MANUAL FOR THE U.S. NUCLEAR REGULATORY COMMISSION REACTOR SAFETY RESEARCH DATA BANK.

HARDY,J.E.
NUREG/CR-3651: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES.

HARLAN,C.P.
NUREG/CR-3657: PRELIMINARY SCREENING OF FUEL CYCLE AND BY-PRODUCT MATERIAL LICENSES FOR EMERGENCY PLANNING.

HARPER,D.
NUREG/CR-4117: FAULTING AND JOINTING IN AND NEAR SURFACE MINES OF SOUTHWESTERN INDIANA.

HARRER,B.J.
NUREG/CR-3413: OFF-SITE CONSEQUENCES OF RADIOLOGICAL ACCIDENTS:METHODS, COSTS AND SCHEDULES FOR DECONTAMINATION.

HARRINGTON,K.H.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.

HARRINGTON,R.M.
NUREG/CR-3764: BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE.
NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1-September 30,1984.
NUREG/CR-3885 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, October 1-December 31,1984.
NUREG/CR-4402 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31,1985.

HARRIS,P.A.
NUREG/CR-4303: HIGH-LEVEL WASTE PRECLOSURE SYSTEMS SAFETY ANALYSIS.Phase 1, Final Report.

HARRISON,B.D.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.

HARTLEY,J.N.
NUREG/CR-4088: METHODS FOR ESTIMATING RADIOACTIVE AND TOXIC AIRBORNE SOURCE TERMS FOR URANIUM MILLING OPERATIONS.
NUREG/CR-4089: EVALUATION OF FIELD-TESTED FUGITIVE DUST CONTROL TECHNIQUES FOR URANIUM MILL TAILINGS PILES.

HARTY,R.
NUREG/CR-4297: EXTREMITY MONITORING:Considerations For Use,Dosimeter Placement,And Evaluation.
NUREG/CR-4399: POSSIBLE OPTIONS FOR REDUCING OCCUPATIONAL DOSE FROM THE TMI-2 BASEMENT.

HARTZMAN,M.
NUREG/CR-1677 V02: PIPING BENCHMARK PROBLEMS,VOLUME II DYNAMIC ANALYSIS INDEPENDENT SUPPORT MOTION RESPONSE SPECTRUM METHOD.

HASKIN,F.E.
NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT.

HAWTHORNE,J.R.
NUREG/CR-4437: EXPLORATORY STUDIES OF ELEMENT INTERACTIONS AND COMPOSITION DEPENDENCIES IN RADIATION SENSITIVITY DEVELOPMENT.

HAYMAN,R.B.
NUREG/CR-4317 V01: CANADIAN SEISMIC AGREEMENT.Technical Report Covering 1979-1985.

HEABERLIN,S.W.
NUREG/CR-2800 S03: GUIDELINES FOR NUCLEAR POWER PLANT SAFETY ISSUE PRIORITIZATION INFORMATION DEVELOPMENT.

HEATON,H.T.
NUREG/CR-4266: STANDARD BETA-PARTICLE AND MONOENERGETIC ELECTRON SOURCES FOR THE CALIBRATION OF BETA-RADIATION PROTECTION INSTRUMENTATION.

HEBDON,F.J.
NUREG-1022 S02: LICENSEE EVENT REPORT SYSTEM.Evaluation Of First Year Results And Recommendations For Improvements.

HECK,C.L.
NUREG/CR-4127 V01: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 1:Numerical Methods.

HELTON,J.C.
NUREG/CR-3904: A COMPARISON OF UNCERTAINTY AND SENSITIVITY ANALYSIS TECHNIQUES FOR COMPUTER MODELS.
NUREG/CR-4199: A DEMONSTRATION UNCERTAINTY/SENSITIVITY ANALYSIS USING THE HEALTH AND ECONOMIC CONSEQUENCE MODEL CRAC2.
NUREG/CR-4342: UNCERTAINTY AND SENSITIVITY ANALYSIS OF A MODEL FOR MULTICOMPONENT AEROSOL DYNAMICS.

HENAGER,C.H.
NUREG/CR-4070 V03: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Factors That May Intensify The Safety Consequences Of Biofouling.

HENNICK,A.
NUREG/CR-3791: CLOSEOUT OF IE BULLETIN 79-09:FAILURE OF GE TYPE AK-2 CIRCUIT BREAKERS IN SAFETY-RELATED SYSTEMS.
NUREG/CR-3794: CLOSEOUT OF IE BULLETIN 80-25:OPERATING PROBLEMS WITH TARGET ROCK SAFETY-RELIEF VALVES AT BWRS.
NUREG/CR-4003: CLOSEOUT OF IE BULLETIN 79-04:INCORRECT WEIGHTS FOR SWING CHECK VALVES MANUFACTURED BY VELAN ENGINEERING CORPORATION.
NUREG/CR-4004: CLOSEOUT OF IE BULLETIN 79-25:FAILURES OF WESTINGHOUSE BFD RELAYS IN SAFETY-RELATED SYSTEMS.
NUREG/CR-4005: CLOSEOUT OF IE BULLETIN 80-12:DECAY HEAT REMOVAL SYSTEM OPERABILITY.
NUREG/CR-4006: CLOSEOUT OF IE BULLETIN 81-01:SURVEILLANCE OF MECHANICAL SNUBBERS.

HENNINGER,R.J.
NUREG/CR-4140: DOMINANT ACCIDENT SEQUENCES IN OCONEE-1 PRESSURIZED WATER REACTOR.

HENSLEY,W.T.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

HENTE,D.B.
NUREG/CR-4095: TEST SERIES 2:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E EXIDE FHC-19 BATTERY CELLS.
NUREG/CR-4096: TEST SERIES 3:SEISMIC FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E C&D LCU-13 BATTERY CELLS.
NUREG/CR-4097: TEST SERIES 4:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED EXIDE EMP-13 BATTERY CELLS.

HERCZEG,A.L.
NUREG/CR-4094: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN SULFATE LAKE ENVIRONMENTS.
NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.

HERSKOVITZ,M.B.
NUREG/CR-3651: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES.

HESSON,G.M.
NUREG/CR-3659: A MATHEMATICAL MODEL FOR ASSESSING THE UNCERTAINTIES OF INSTRUMENTATION MEASUREMENTS FOR POWER AND FLOW OF PWR REACTORS.

HESTER,O.V.
NUREG/CR-4071: EXPLORATORY TREND AND PATTERN ANALYSIS FOR 1981 LICENSEE EVENT REPORT DATA.

HICKS,B.B.
NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE.

HIGGINS,J.C.
NUREG/CR-4440: A REVIEW OF EMERGENCY DIESEL GENERATOR PERFORMANCE AT NUCLEAR POWER PLANTS.

HILEMAN,J.A.
NUREG/CR-4145: EARTHQUAKE RECURRENCE INTERVALS AT NUCLEAR POWER PLANTS: ANALYSIS AND RANKING.

HILL,O.F.
NUREG/CR-4088: METHODS FOR ESTIMATING RADIOACTIVE AND TOXIC AIRBORNE SOURCE TERMS FOR URANIUM MILLING OPERATIONS.

HINKLE,N.E.
NUREG/CR-4403: SUMMARY OF THE WASTE MANAGEMENT PROGRAMS AT URANIUM RECOVERY FACILITIES AS THEY RELATE TO THE 40 CFR PART 192 STANDARDS.

HINZE,W.J.
NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983.

HISER,A.
NUREG/CR-4395: CORRELATION OF CV AND KIC/KJC TRANSITION TEMPERATURE INCREASES DUE TO IRRADIATION.

HITCHCOCK,J.T.
NUREG/CR-4060: THE DC-1 AND DC-2 DEBRIS COOLABILITY AND MELT DYNAMICS EXPERIMENTS.

HOBBS,R.W.
NUREG/CR-3872: DATA ACQUISITION AND CONTROL OF THE HSST SERIES V IRRADIATION EXPERIMENT AT THE ORR.

HOCHREITER,L.E.
NUREG/CR-4166: ANALYSIS OF FLECHT-SEASET 163-ROD BLOCKED BUNDLE DATA USING COBRA-TF.
NUREG/CR-4167: FLECHT SEASET PROGRAM.Final Report.NRC/EPRI Westinghouse Report Number 16.

HOFFMAN,F.O.
NUREG/CR-3738: ENVIRONMENTAL EFFECTS OF THE URANIUM FUEL CYCLE.A Review Of Data For Technetium.

HOFMAYER,C.H.
NUREG/CP-0070: PROCEEDINGS OF THE WORKSHOP ON SEISMIC AND DYNAMIC FRAGILITY OF NUCLEAR POWER PLANT COMPONENTS.

HOLLAND,R.A.
NUREG-0905: CLOSEOUT OF IE BULLETIN 79-12:SHORT-PERIOD SCRAMS AT BOILING-WATER REACTORS.

HOLMAN,G.S.
NUREG/CR-3660 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 1:Summary Report.
NUREG/CR-3660 V04: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 4:Pipe Failure Induced By Crack Growth In West Coast Plants.
NUREG/CR-3663 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 1:Summary Report.

HOOK,S.E.
NUREG/CR-3516: A SURVEY OF THE USES OF RADIOACTIVE MATERIALS IN LOUISIANA'S OFFSHORE WATERS.

HOPENFELD,J.
NUREG-1108: RADIOACTIVITY TRANSPORT FOLLOWING STEAM GENERATOR TUBE RUPTURE.

HORAN,J.R.
NUREG/CP-0066: PROCEEDINGS OF AN INTERNATIONAL WORKSHOP ON HISTORIC DOSE EXPERIENCE AND DOSE REDUCTION (ALARA) AT NUCLEAR POWER PLANTS,MAY 29-JUNE 1,1984.

HORSCHEL,D.S.
NUREG/CR-3647: DESIGN AND FABRICATION OF A 1/8-SCALE STEEL CONTAINMENT MODEL.

HORTON,W.
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP).A Methodology For Estimating Risk Impacts Of Safeguards Measures.

HOSKER,R.P.
NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE.

HSIEH,B.J.
NUREG/CR-4432: COMPARISON OF DYNAMIC CHARACTERISTICS OF FUKUSHIMA NUCLEAR POWER PLANT CONTAINMENT BUILDING DETERMINED FROM TESTS AND EARTHQUAKES.

HUANG,C.
NUREG/CR-4042: A 3-DIMENSIONAL COMPUTER MODEL TO SIMULATE FLUID FLOW AND CONTAINMENT TRANSPORT THROUGH A ROCK FRACTURE SYSTEM.

HUCHTON,R.L.
NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

HUENEFELD,J.C.
NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE.

HUMPHREYS,P.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

HUMPHREYS,P.C.
NUREG/CR-4016 V01: APPLICATION OF SLIM-MAUD:A TEST OF AN INTERACTIVE COMPUTER-BASED METHOD FOR ORGANIZING EXPERT ASSESSMENT OF HUMAN PERFORMANCE AND RELIABILITY.Volume I:Main Report.

HUNTSMAN,B.L.
NUREG/CR-4033: THE ROLE OF PERSONAL AIR SAMPLING IN RADIATION SAFETY PROGRAMS AND RESULTS OF A LABORATORY EVALUATION OF PERSONAL AIR-SAMPLING EQUIPMENT.

HUTTON,P.H.
NUREG/CR-3825 V03-4: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Quarterly Report, April 1984 - September 1984.Volumes 3 and 4.
NUREG/CR-3915: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL.
NUREG/CR-4300 V01: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Progress Report, October-March 1985.

HWANG,H.
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.
NUREG/CR-4329: RELIABILITY EVALUATION OF CONTAINMENTS INCLUDING SOIL-STRUCTURE INTERACTION.

HWANG,W.S.
NUREG/CR-4128: BWR FULL INTEGRAL SIMULATION TEST (FIST) PHASE II TEST RESULTS AND TRAC-BWR MODEL QUALIFICATION.

ILBERG,D.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-4050: A REVIEW OF THE SHOREHAM NUCLEAR POWER STATION PROBABILISTIC RISK ASSESSMENT.Internal Events And Core Damage Frequency.

IMAN,R.L.
NUREG/CR-3904: A COMPARISON OF UNCERTAINTY AND SENSITIVITY ANALYSIS TECHNIQUES FOR COMPUTER MODELS.
NUREG/CR-4122: A FORTRAN 77 PROGRAM AND USER'S GUIDE FOR THE CALCULATION OF PARTIAL CORRELATION AND STANDARDIZED REGRESSION COEFFICIENTS.
NUREG/CR-4199: A DEMONSTRATION UNCERTAINTY/SENSITIVITY ANALYSIS USING THE HEALTH AND ECONOMIC CONSEQUENCE MODEL CRAC2.
NUREG/CR-4342: UNCERTAINTY AND SENSITIVITY ANALYSIS OF A MODEL FOR MULTICOMPONENT AEROSOL DYNAMICS.
NUREG/CR-4350 V02: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 2-Probability And Statistics For PRA Applications.

IMS,J.R.
NUREG/CR-4133: NUCLEAR POWER SAFETY REPORTING SYSTEM IMPLEMENTATION AND OPERATIONAL SPECIFICATIONS.

IRELAND,J.R.
NUREG/CR-3706: TRAC ANALYSES OF SEVERE OVERCOOLING TRANSIENTS FOR THE OCONEE-1 PWR.

ISAACSON,L.
NUREG/CR-3237: CONTROL OF EXPLOSIVE MIXTURES IN PWR WASTE GAS SYSTEMS.

ISACHSEN,Y.W.
NUREG/CR-3178: STRUCTURAL AND TECTONIC STUDIES IN NEW YORK STATE.Final Report,July 1981 - June 1982.

ISHII,M.
NUREG/CR-4277: INVERTED ANNUAL FLOW EXPERIMENTAL STUDY.

ISKANDER,S.K.
NUREG/CR-4249: PRESSURE VESSEL FRACTURE STUDIES PENETRATING TO THE PWR THERMAL-SHOCK ISSUE:EXPERIMENTS TSE-5,TSE-5A AND TSE-6.
NUREG/CR-4304: PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO THE PWR THERMAL-SHOCK ISSUE.Experiment TSE-7.

IYER,K.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

JACKSON,D.H.
NUREG/CR-4281: AN EMPIRICAL ANALYSIS OF SELECTED NUCLEAR POWER PLANT MAINTENANCE FACTORS AND PLANT SAFETY.

JACKSON,P.O.
NUREG/CR-4057: RADIOLOGICAL ASSESSMENT OF THE TOWN OF EDGEMONT.

JACOBS,G.K.
NUREG/CP-0062: PROCEEDINGS OF THE CONFERENCE ON THE APPLICATION OF GEOCHEMICAL MODELS TO HIGH-LEVEL NUCLEAR WASTE REPOSITORY ASSESSMENT.
NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984.
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.

JAIN,H.
NUREG/CR-2482 V08: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period October 1984 - March 1985.
NUREG/CR-2482 V09: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period April 1985-September 1985.
NUREG/CR-3091 V04: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1983 - March 1984.
NUREG/CR-3091 V05: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period April 1984 - September 1984.

JAMISON,J.D.
NUREG/CR-4151: INTEGRATION OF EMERGENCY ACTION LEVELS WITH COMBUSTION ENGINEERING EMERGENCY OPERATING PROCEDURES.By Use Of Combustion Engineering Owners Group Emergency Operating Procedure Technical Guidelines.

JANG,J.
NUREG-0837 V04 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, July-September 1984.
NUREG-0837 V04 N04: NRC TLD DIRECT RADIATION MONITORING REPORT.Progress Report, October-December 1984.
NUREG-0837 V05 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, January-March 1985.
NUREG-0837 V05 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, April-June 1985.

JANKOWSKI,M.W.
NUREG-0856 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).

JAROSS,R.A.
NUREG/CR-4180: STATE-OF-THE-ART OF SOLID-STATE MOTOR CONTROLLERS.

JARRELL,D.B.
NUREG/CR-4046: DETERMINING CRITICAL FLOW VALVE CHARACTERISTICS USING EXTRAPOLATION TECHNIQUES.

JEANMOUGIN,N.
NUREG/CR-4294: LEAK RATE ANALYSIS OF THE WESTINGHOUSE REACTOR COOLANT PUMP.

JENKINS,J.P.
NUREG/CR-4040: OPERATIONAL DECISIONMAKING AND ACTION SELECTION UNDER PSYCHOLOGICAL STRESS IN NUCLEAR POWER PLANTS.

JENNE,E.A.
NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER.(Final Report)

JEUNG,N.
NUREG/CR-4355 V01: 238 PU(IV) IN MONKEYS.Overview Of Metabolism.

JO,J.
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.

JO,J.H.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4253: REVIEW OF TRAC CALCULATIONS FOR CALVERT CLIFFS PTS STUDY.
NUREG/CR-4292: A COMPARATIVE ANALYSIS OF CONSTITUTIVE RELATIONS IN TRAC-PFL AND RELAP5/MOD1.

JOHNSON,A.C.
NUREG/CR-4288: FOCAL MECHANISM ANALYSES FOR VIRGINIA AND EASTERN TENNESSEE EARTHQUAKES (1978-1984).

JOHNSON,J.D.
NUREG/CR-3657: PRELIMINARY SCREENING OF FUEL CYCLE AND BY-PRODUCT MATERIAL LICENSES FOR EMERGENCY PLANNING.
NUREG/CR-4122: A FORTRAN 77 PROGRAM AND USER'S GUIDE FOR THE CALCULATION OF PARTIAL CORRELATION AND STANDARDIZED REGRESSION COEFFICIENTS.
NUREG/CR-4199: A DEMONSTRATION UNCERTAINTY/SENSITIVITY ANALYSIS USING THE HEALTH AND ECONOMIC CONSEQUENCE MODEL CRAC2.
NUREG/CR-4342: UNCERTAINTY AND SENSITIVITY ANALYSIS OF A MODEL FOR MULTICOMPONENT AEROSOL DYNAMICS.

JOHNSON,J.J.
NUREG/CR-4331: SIMPLIFIED SEISMIC PROBABILISTIC RISK ASSESSMENT.Procedures And Limitations.

JOHNSON,K.I.
NUREG/CR-4070 V02: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Volume 2:Current Status Of Biofouling Surveillance And Control Techniques.
NUREG/CR-4070 V03: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Factors That May Intensify The Safety Consequences Of Biofouling.
NUREG/CR-4267: VESSEL INTEGRITY SIMULATION (VISA) CODE SENSITIVITY STUDY.

JOHNSON,M.P.
NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3922 V01: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Main Report And Appendices A And B.
NUREG/CR-3922 V02: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Appendices C And D.

JONES,J.W.
NUREG/CR-3736: FIELD AND THEORETICAL INVESTIGATIONS OF FRACTURED CRYSTALLINE ROCK NEAR ORACLE,ARIZONA.

JONES,T.N.
NUREG/CR-4092: ORNL CHARACTERIZATION OF HEAVY-SECTION STEEL TECHNOLOGY PROGRAM PLATES 01,02,AND 03.

JORDAN,H.
NUREG/CR-4205: TRAP-MELT2 USER'S MANUAL.
NUREG/CR-4388: AEROSOL BEHAVIOR MODELING (TASK 3) - SUPPORT SERVICES FOR RESEARCH AND EVALUATION OF SEVERE ACCIDENT PHENOMENA AND MITIGATION.

JORGENSEN,C.C.
NUREG/CR-3481 V02: NUCLEAR POWER PLANT PERSONNEL QUALIFICATIONS AND TRAINING: TAPS -- The Task Analysis Profiling System.

JUNG,R.G.
NUREG/CR-4172: A USER'S GUIDE FOR MERGE.

KAGAMI,S.
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.

KAHL,W.K.
NUREG/CR-3831: THE IN-PLANT RELIABILITY DATA BASE FOR NUCLEAR PLANT COMPONENTS.Interim Report - Diesel Generators,Batteries, Chargers And Inverters.

KAHN,B.
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.

KAM,F.B.
NUREG/CR-4031 V02: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutronics Calculations."
NUREG/CR-4031 V03: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutron Exposure Parameters."
NUREG/CR-4284: NEUTRON EXPOSURE PARAMETERS FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY IRRADIATION SERIES.

KAO,C.
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.

KARIMIAN,S.
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP).A Methodology For Estimating Risk Impacts Of Safeguards Measures.

KASHIWA,B.A.
NUREG/CR-4079: ANALYTIC STUDIES PERTAINING TO STEAM GENERATOR TUBE RUPTURE ACCIDENTS.

KASSIR,M.
NUREG/CR-4221: AN EVALUATION OF STRESS CORROSION CRACK GROWTH IN BWR PIPING SYSTEMS.

KASSNER,T.F.
NUREG/CR-4287: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Annual Report,October 1983 - September 1984.

KATO,W.Y.
NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.
NUREG/CR-4152: AN INDEPENDENT SAFETY ORGANIZATION.

KELLER,G.R.
NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983.

KELLY,J.E.
NUREG/CR-2951: THE D9 EXPERIMENT.Heat Removal From Stratified UO2 Debris.
NUREG/CR-4060: THE DC-1 AND DC-2 DEBRIS COOLABILITY AND MELT DYNAMICS EXPERIMENTS.

KELLY,J.M.
NUREG/CR-4166: ANALYSIS OF FLECHT-SEASET 163-ROD BLOCKED BUNDLE DATA USING COBRA-TF.

KELLY,P.
NUREG/CR-4365: DESIGN AND DEVELOPMENT OF A SPECIAL PURPOSE SAFT SYSTEM FOR NONDESTRUCTIVE EVALUATION OF NUCLEAR REACTOR VESSELS AND PIPING COMPONENTS.

KELMERS,A.D.
NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.REPORT FOR OCTOBER-DECEMBER 1984.
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.

KEMPF,C.R.
NUREG/CR-3865: EVALUATION OF THE RADIOACTIVE INVENTORY IN,AND ESTIMATION OF ISOTOPIC RELEASE FROM,THE WASTE IN EIGHT TRENCHES AT THE SHEFFIELD LOW-LEVEL WASTE BURIAL SITE.
NUREG/CR-4406: AN ANALYSIS OF LOW-LEVEL WASTES.Review of Hazardous Waste Regulations And Identification of Radioactive Mixed Wastes. Final Report.

KEMPKA,S.N.
NUREG/CR-4136: SMOKE.A Data Reduction Package For Analysis Of Combustion Experiments.

KEMPPAINEN,M.
NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS.

KENNEDY,R.P.
NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3663 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS,Volume 3:Double Ended Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3805 V02: ENGINEERING CHARACTERIZATION OF GROUND MOTION.Task II: Effects Of Ground Motion Characteristics On Structural Response Considering Localized Structural Nonlinearities And Soil-Structure Interaction Effects.
NUREG/CR-4334: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS.

KERLIN,T.W.
NUREG/CR-4256: MEASUREMENT OF RESPONSE TIME AND DETECTION OF DEGRADATION IN PRESSURE SENSOR/SENSING LINE SYSTEMS.

KERSCHNER,H.F.
NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET-GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE,AND SODIUM CHLORIDE.

KESSLER,J.H.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.

KEYS,W.S.
NUREG/CR-3736: FIELD AND THEORETICAL INVESTIGATIONS OF FRACTURED CRYSTALLINE ROCK NEAR ORACLE,ARIZONA.

KHATIB-RAHBAR
NUREG/CR-3498: TWO-DIMENSIONAL MODELING OF INTRA-SUBASSEMBLY HEAT TRANSFER AND BUOYANCY-INDUCED FLOW REDISTRIBUTION IN LMFBRS.
NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences.

KILLOUGH,G.G.
NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE.

KIM,H.J.
NUREG/CR-4414: DIRECT-CONTACT CONDENSATION OF STEAM ON COLD WATER IN STRATIFIED COUNTERCURRENT FLOW.
NUREG/CR-4417: LOCAL PROPERTIES OF COUNTERCURRENT STRATIFIED STEAM-WATER FLOW.

KIM,S.H.
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.

KIMBALL,C.S.
NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS.

KINCAID,R.H.
NUREG/CR-3805 V02: ENGINEERING CHARACTERIZATION OF GROUND MOTION.Task II: Effects Of Ground Motion Characteristics On Structural Response Considering Localized Structural Nonlinearities And Soil-Structure Interaction Effects.

KINNISON,R.
NUREG/CP-0063: PROCEEDINGS OF THE 1984 STATISTICAL SYMPOSIUM ON NATIONAL ENERGY ISSUES.

KIPP,T.R.
NUREG/CR-4290 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF BABCOCK AND WILCOX PWR PLANTS.Volume 2:Guillotine Break Indirectly Induced By Earthquakes.

KITTMER,C.A.
NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT.

KMETYK,L.N.
NUREG/CR-3802: RELAP5 ASSESSMENT:QUANTITATIVE KEY PARAMETERS AND RUN TIME STATISTICS.
NUREG/CR-3936: RELAP5 ASSESSMENT:CONCLUSIONS AND USER GUIDELINES.

KNEE,H.E.
NUREG/CR-3626 V02: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: DESCRIPTION OF MODEL CONTENT,STRUCTURE,AND SENSITIVITY TESTING.

KNIGHT,T.D.
NUREG/CR-3208: TRAC-PD2 DEVELOPMENTAL ASSESSMENT.
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.
NUREG/CR-3866: TRAC-PD2 INDEPENDENT ASSESSMENT.

KNOPOFF,L.
NUREG/CR-4145: EARTHQUAKE RECURRENCE INTERVALS AT NUCLEAR POWER PLANTS: ANALYSIS AND RANKING.

KOCHER,D.C.
NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE.

KOEHL,E.R.
NUREG/CR-4180: STATE-OF-THE-ART OF SOLID-STATE MOTOR CONTROLLERS.

KOENIG,J.E.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4109: TRAC-PF1 ANALYSES OF POTENTIAL PRESSURIZED-THERMAL SHOCK TRANSIENTS AT CALVERT CLIFFS/UNIT 1.A Combustion Engineering PWR.

KOESTER,J.P.
NUREG/CR-4430: CURRENT METHODOLOGIES FOR ASSESSING THE POTENTIAL FOR EARTHQUARKE-INDUCED LIQUEFACTION IN SOILS.

KOGAN,V.
NUREG/CR-4388: AEROSOL BEHAVIOR MODELING (TASK 3) - SUPPORT SERVICES FOR RESEARCH AND EVALUATION OF SEVERE ACCIDENT PHENOMENA AND MITIGATION.

KOH,B.R.
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.

KOHRT,J.M.
NUREG/CR-4166: ANALYSIS OF FLECHT-SEASET 163-ROD BLOCKED BUNDLE DATA USING COBRA-TF.

KOLLAR,F.
NUREG/CR-4317 V01: CANADIAN SEISMIC AGREEMENT.Technical Report Covering 1979-1985.

KOPSTEIN,F.F.
NUREG/CR-3634: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: USER'S MANUAL.
NUREG/CR-4104: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL. Field Evaluation/Validation.

KOT,C.A.
NUREG/CR-4432: COMPARISON OF DYNAMIC CHARACTERISTICS OF FUKUSHIMA NUCLEAR POWER PLANT CONTAINMENT BUILDING DETERMINED FROM TESTS AND EARTHQUAKES.

KOUSARI,B.
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

KRAMARIC,M.
NUREG-0837 V04 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, July-September 1984.
NUREG-0837 V04 N04: NRC TLD DIRECT RADIATION MONITORING REPORT.Progress Report, October-December 1984.
NUREG-0837 V05 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, January-March 1985.
NUREG-0837 V05 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, April-June 1985.

KRAMER,G.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

KRANTZ,E.A.
NUREG/CR-3301: CATALOG OF PRA DOMINANT ACCIDENT SEQUENCE INFORMATION.

KRASNER,L.M.
NUREG/CR-4231: EVALUATION OF AVAILABLE DATA FOR PROBABILISTIC RISK ASSESSMENTS (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.

KROIS,P.A.
NUREG/CR-3887: HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS.

KU,J.Y.
NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE.

KUHLMAN,M.R.
NUREG/CR-4173: CORSOR USER'S MANUAL.
NUREG/CR-4205: TRAP-MELT2 USER'S MANUAL.

KULAK,R.F.
NUREG/CR-4064: STRUCTURAL RESPONSE OF LARGE PENETRATIONS AND CLOSURES FOR CONTAINMENT VESSELS SUBJECTED TO LOADINGS BEYOND DESIGN BASIS.

KULLBERG,C.M.
NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR.
NUREG/CR-4196: OVERVIEW OF TRAC-BD1 (VERSION 12) ASSESSMENT STUDIES.

KUNSMAN,D.M.
NUREG/CR-4335: POTENTIAL BENEFITS OBTAINED BY REQUIRING SAFTEY-GRADE COLD SHUTDOWN SYSTEMS.

KUPPERMAN,D.S.
NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984.
NUREG/CR-4368: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS: Semiannual Report,October 1984 - March 1985.

KURTH,R.
NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.

KURTZ,R.J.
NUREG/CR-3825 V03-4: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Quarterly Report, April 1984 - September 1984.Volumes 3 and 4.
NUREG/CR-3915: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL.
NUREG/CR-4300 V01: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Progress Report, October-March 1985.

KWAN,Q.
NUREG/CR-4357: THE FEASIBILITY OF DETECTING THE IMPORT OF UNAUTHORIZED RADIOACTIVE MATERIALS INTO THE UNITED STATES.

LAATS,E.T.
NUREG/CR-2531 R03: INTRODUCTORY USER'S MANUAL FOR THE U.S. NUCLEAR REGULATORY COMMISSION REACTOR SAFETY RESEARCH DATA BANK.

LAHEY,R.T.
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.
NUREG/CR-4116: NUFEGQ-NP:A DIGITAL COMPUTER CODE FOR THE LINEAR STABILITY ANALYSIS OF BOILING WATER NUCLEAR REACTORS.

LAI,W.
NUREG/CR-3854: FABRICATION CRITERIA FOR SHIPPING CONTAINERS.

LAMBE,W.M.
NUREG-0970: PROCEDURES FOR MEETING NRC ANTITRUST RESPONSIBILITIES.

LAMONICA,L.B.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

LANDOW,M.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

LANNING,D.D.
NUREG/CR-4168: GT2F:A COMPUTER CODE FOR ESTIMATING LIGHT WATER REACTOR FUEL ROD FAILURES.

LARSEN,R.P.
NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS.Application To Establishing Values Of f1 For Soluble Plutonium.

LARSON,J.R.
NUREG/CR-4041: SYSTEM ANALYSIS HANDBOOK.
NUREG/CR-4041 R01: SYSTEM ANALYSIS HANDBOOK.

LASSITER,D.L.
NUREG/CR-4280: THE EFFECTS OF SUPERVISOR EXPERIENCE AND ASSISTANCE OF A SHIFT TECHNICAL ADVISOR (STA) ON CREW PERFORMANCE IN CONTROL ROOM SIMULATORS.

LAUGHERY,K.R.
NUREG/CR-4206: A SELECT REVIEW OF THE RECENT (1979-1983) BEHAVIORAL RESEARCH LITERATURE ON TRAINING SIMULATORS.

LEAHY,T.J.
NUREG/CR-4350 V01: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 1 - PRA Fundamentals.

LEE,J.Y.
NUREG-0017 R01: CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM PRESSURIZED WATER REACTORS (PWR-GALE CODE).

LEE,S.C.
NUREG/CR-4416: STABILITY OF STEAM-WATER COUNTERCURRENT STRATIFIED FLOW.

LEE,S.Y.
NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.

LEHMICKE,D.J.
NUREG/CR-4173: CORSOR USER'S MANUAL.

LEIGH,C.D.
NUREG/CR-4342: UNCERTAINTY AND SENSITIVITY ANALYSIS OF A MODEL FOR MULTICOMPONENT AEROSOL DYNAMICS.

LEKACH,S.V.
NUREG/CR-3943: THE BWR PLAN ANALYZER.

LEMEUR,M.
NUREG/CR-4091: THE EFFECT OF ALTERNATIVE AGING AND ACCIDENT SIMULATIONS ON POLYMER PROPERTIES.

LEONARD,M.
NUREG/CR-4177 V01: MANAGEMENT OF SEVERE ACCIDENTS.Perspectives On Managing Severe Accidents In Commercial Nuclear Power Plants.
NUREG/CR-4177 V02: MANAGEMENT OF SEVERE ACCIDENTS.Extending Plant Operating Procedures Into The Severe Accident Regime.

LEPEL,E.A.
NUREG/CR-4100: EVALUATION OF INSTRUMENTAL METHODS FOR THE MEASUREMENT OF YELLOWCAKE EMISSIONS.

LEVERENZ,F.L.
NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.
NUREG/CR-4350 V06: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 6 - Data Development.

LEWELLEN,W.S.
NUREG/CR-4157: A SCIENTIFIC CRITIQUE OF AVAILABLE MODELS FOR REAL-TIME SIMULATIONS OF DISPERSION.
NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING.
NUREG/CR-4159: COMPARISON OF THE 1981 INEL DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS.

LEWIS,J.R.
NUREG/CR-4298: DESIGN AND INSTALLATION OF COMPUTER SYSTEMS TO MEET THE REQUIREMENTS OF 10 CFR 73.55.

LEWIS,M.
NUREG/CR-4361: STEAM GENERATOR GROUP PROJECT.Annual Report - 1983.
NUREG/CR-4362: STEAM GENERATOR GROUP PROJECT.Annual Report - 1984.

LEWIS,P.M.
NUREG/CR-4248: RECOMMENDATIONS FOR NRC POLICY ON SHIFT SCHEDULING AND OVERTIME AT NUCLEAR POWER PLANTS.

LEYLAK,J.
NUREG/CR-4120: MATHEMATICAL MODELING OF ULTIMATE HEAT SINK COOLING PONDS.

LI,W.W.
NUREG/CR-4072: THE ESTIMATION OF ATMOSPHERIC DISPERSION AT NUCLEAR POWER PLANTS UTILIZING REAL TIME ANEMOMETER STATISTICS.

LIAO,L.H.
NUREG/CR-4376: HEAT TRANSFER,CARRYOVER AND FALL BACK IN PWR STEAM GENERATORS DURING TRANSIENTS.

LIDIAK,E.G.
NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983.

LIGON,D.M.
NUREG/CR-4303: HIGH-LEVEL WASTE PRECLOSURE SYSTEMS SAFETY ANALYSIS.Phase 1. Final Report.

LIPINSKI,R.J.
NUREG/CR-2951: THE D9 EXPERIMENT.Heat Removal From Stratified UO2 Debris.

LIPPINCOTT,E.P.
NUREG/CR-3746 V02: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.Semiannual Progress Report,April 1984 - September 1984.

LLOYD,R.D.
NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.

LO,T.Y.
NUREG/CR-3660 V04: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 4:Pipe Failure Induced By Crack Growth In West Coast Plants.
NUREG/CR-3663 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 1:Summary Report.

LOBNER,P.
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP).A Methodology For Estimating Risk Impacts Of Safeguards Measures.

LOFARO,R.
NUREG/CR-4294: LEAK RATE ANALYSIS OF THE WESTINGHOUSE REACTOR COOLANT PUMP.

LOFGREN,E.V.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-4350 V03: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 3 - Systems Reliability And Analysis Techniques,Session A -Reliability.
NUREG/CR-4350 V05: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 5 - Systems Reliability And Analysis Techniques,Session D - Quantification.

LONGEST,A.W.
NUREG/CR-4346: AEROSOL RELEASE EXPERIMENTS IN THE FUEL AEROSOL SIMULANT TEST FACILITY:UNDERSODIUM EXPERIMENTS.

LOOMIS,G.G.
NUREG/CR-4073: RESULTS OF THE SEMISCALE MOD-2B STEAM GENERATOR TUBE RUPTURE TEST SERIES.

LOPES,J.C.
NUREG/CR-4424: DROPLET SIZES,DYNAMICS AND DEPOSITION IN VERTICAL ANNULAR FLOW.

LORENZ,R.A.
NUREG/CR-3930: OBSERVED BEHAVIOR OF CESIUM,IODINE,AND TELLURIUM IN THE ORNL FISSION PRODUCT RELEASE PROGRAM.
NUREG/CR-4037: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-5.
NUREG/CR-4043: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-6.

LORET,L.S.
NUREG/CR-4397: IN-PLANT SOURCE TERM MEASUREMENTS AT PRAIRIE ISLAND NUCLEAR GENERATING STATION.

LOSS,F.J.
NUREG/CR-3228 V03: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1984.

LOYOLA,V.M.
NUREG/CR-3361: THE EFFECT OF WATER CHEMISTRY ON THE RATES OF HYDROGEN GENERATION FROM GALVANIZED STEEL CORROSION AT POST-LOCA CONDITIONS.
NUREG/CR-3803: THE EFFECTS OF POST-LOCA CONDITIONS ON A PROTECTIVE COATING (PAINT) FOR THE NUCLEAR POWER INDUSTRY.

LU,S.C.
NUREG/CR-4263: RELIABILITY ANALYSIS OF STIFF VERSUS FLEXIBLE PIPING FINAL PROJECT REPORT.

LUCKAS,W.J.
NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.

LUDEWIG,H.
NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences.

LUDWICK,J.D.
NUREG/CR-4176: EMISSION CONTROL TECHNOLOGY AND QUALITY ASSURANCE NEEDS AT URANIUM MILLING FACILITIES.Includes Supporting Methods For Testing,Operating,And Maintaining Air Pollution Control Devices.

LUMM,D.K.
NUREG/CR-4333: STE. GENEVIEVE FAULT ZONE,MISSOURI AND ILLINOIS.

LYONS,J.A.
NUREG/CR-4317 V01: CANADIAN SEISMIC AGREEMENT.Technical Report Covering 1979-1985.

MACKENZIE,D.R.
NUREG/CR-3865: EVALUATION OF THE RADIOACTIVE INVENTORY IN,AND ESTIMATION OF ISOTOPIC RELEASE FROM,THE WASTE IN EIGHT TRENCHES AT THE SHEFFIELD LOW-LEVEL WASTE BURIAL SITE.
NUREG/CR-4062: EXTENDED STORAGE OF LOW-LEVEL RADIOACTIVE WASTES:Potential Problem Areas.
NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA.
NUREG/CR-4406: AN ANALYSIS OF LOW-LEVEL WASTES:Review of Hazardous Waste Regulations And Identification of Radioactive Mixed Wastes. Final Report.

MACKOWIAK,D.P.
NUREG/CR-3862: DEVELOPMENT OF TRANSIENT INITIATING EVENT FREQUENCIES FOR USE IN PROBABILISTIC RISK ASSESSMENTS.

MADDEN,E.G.
NUREG/CR-4104: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL. Field Evaluation/Validation.

MAERKER,R.E.
NUREG/CR-4039: GAMMA-RAY CHARACTERIZATION OF THE TWO-YEAR IRRADIATION EXPERIMENT PERFORMED AT THE POOLSIDE FACILITY.

MAIYA,P.S.
NUREG/CR-4287: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Annual Report,October 1983 - September 1984.

MALLEN,A.N.
NUREG/CR-3943: THE BWR PLAN ANALYZER.

MANAHAN,M.
NUREG/CR-4177 V01: MANAGEMENT OF SEVERE ACCIDENTS.Perspectives On Managing Severe Accidents In Commercial Nuclear Power Plants.

MANDLER,J.W.
NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION.
NUREG/CR-4397: IN-PLANT SOURCE TERM MEASUREMENTS AT PRAIRIE ISLAND NUCLEAR GENERATING STATION.

MANN,N.R.
NUREG/CR-4145: EARTHQUAKE RECURRENCE INTERVALS AT NUCLEAR POWER PLANTS: ANALYSIS AND RANKING.

MANNING,J.J.
NUREG/CR-3887: HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS.

MARCH-LEUBA,J.
NUREG/CR-4256: MEASUREMENT OF RESPONSE TIME AND DETECTION OF DEGRADATION IN PRESSURE SENSOR/SENSING LINE SYSTEMS.

MARSCHALL,C.W.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

MARTIN,D.M.
NUREG/CR-4198: FRACTURE IN GLASS/HIGH LEVEL WASTE CANISTERS.

MARTIN,R.A.
NUREG/CR-4260: TORAC USER'S MANUAL.A Computer Code For Analyzing Tornado-Induced Flow And Material Transport In Nuclear Facilities.
NUREG/CR-4264: INVESTIGATION ON HIGH-EFFICIENCY PARTICULATE AIR FILTER PLUGGING BY COMBUSTION AEROSOLS.
NUREG/CR-4321: FULL SCALE MEASUREMENTS OF SMOKE TRANSPORT AND DEPOSITION IN VENTILATION SYSTEM DUCTWORK.

MARTIN,R.E.
NUREG-1149: TECHNICAL SPECIFICATIONS FOR LIMERICK GENERATING STATION, UNIT 1.Docket No. 50-352. (Philadelphia Electric Company)

MARTIN,W.J.
NUREG/CR-3709: METHODS OF MINIMIZING GROUND-WATER CONTAMINATION FROM IN SITU LEACH URANIUM MINING.Final Report.

MARTINI,D.K.
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.

MARTZ,H.F.
NUREG/CR-4217: A STATISTICAL ANALYSIS OF NUCLEAR POWER PLANT VALVE FAILURE-RATE VARIABILITY--SOME PRELIMINARY RESULTS.

MAST,P.K.
NUREG/CR-3757: TRAN B-2:THE EFFECT OF LOW STEEL CONTENT ON FUEL PENETRATION IN A NON-MELTING CYLINDRICAL FLOW CHANNEL.
NUREG/CR-3944: TRAN B-3:EXPERIMENTAL INVESTIGATION OF FUEL CRUST STABILITY ON MELTING SURFACES OF AN ANNULAR FLOW CHANNEL.

MATHEWS,P.
NUREG-0933 S02: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

MATHIESON,T.
NUREG/CR-4368: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS: Semiannual Report,October 1984 - March 1985.

MATHIEU,G.G.
NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.

MATTHEWS,G.R.
NUREG/CR-4373: COMPENDIUM OF COST-EFFECTIVENESS EVALUATIONS OF MODIFICATIONS FOR DOSE REDUCTION AT NUCLEAR POWER PLANTS.

MAXEY,W.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

MAYO,C.W.
NUREG/CR-3991: FAILURE MODES AND EFFECTS ANALYSIS (FMEA) OF THE ICS/NNI ELECTRIC POWER DISTRIBUTION CIRCUITRY AT THE OCONEE-1 NUCLEAR PLANT.

MAYS,G.T.
NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.

MAZOUR,T.J.
NUREG/CR-4344: INSTRUCTIONAL SKILLS EVALUATION IN NUCLEAR INDUSTRY TRAINING.

MCARTHUR,D.A.
NUREG/CR-3757: TRAN B-2:THE EFFECT OF LOW STEEL CONTENT ON FUEL PENETRATION IN A NON-MELTING CYLINDRICAL FLOW CHANNEL.
NUREG/CR-3944: TRAN B-3:EXPERIMENTAL INVESTIGATION OF FUEL CRUST STABILITY ON MELTING SURFACES OF AN ANNULAR FLOW CHANNEL.

MCBRIDE,A.F.
NUREG/CR-3991: FAILURE MODES AND EFFECTS ANALYSIS (FMEA) OF THE ICS/NNI ELECTRIC POWER DISTRIBUTION CIRCUITRY AT THE OCONEE-1 NUCLEAR PLANT.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

MCBRIDE,K.C.
NUREG/CR-4298: DESIGN AND INSTALLATION OF COMPUTER SYSTEMS TO MEET THE REQUIREMENTS OF 10 CFR 73.55.

MCCANN,M.
NUREG/CR-2815 V02 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 8-12.

MCCANN,M.W.
NUREG/CR-3485: PRA REVIEW MANUAL.

MCCARDELL,R.K.
NUREG/CR-3948: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY.

MCCLUNG,R.W.
NUREG/CR-3949 V01: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Semiannual Progress Report For Period Ending June 30,1984.
NUREG/CR-3949 V02: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Annual Progress Report For Period Ending December 31,1984.

MCCLURE,J.D.
NUREG/CR-3611: RADIOACTIVE MATERIAL (RAM) ACCIDENT/INCIDENT DATA ANALYSIS PROGRAM.

MCCONNELL,J.W.
NUREG/CR-4150: EPICOR-II RESIN DEGRADATION RESULTS FROM FIRST RESIN SAMPLES OF PF-8 AND PF-20.

MCCONNELL,R.J.
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.

MCCORMICK,R.D.
NUREG/CR-3948: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY.

MCCRAY,J.G.
NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984.

MCCULLAGH,C.M.
NUREG/CR-3237: CONTROL OF EXPLOSIVE MIXTURES IN PWR WASTE GAS SYSTEMS.

MCCULLOCH,R.W.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.

MCCULLOCH,W.H.
NUREG/CR-3954: HECTR ANALYSIS OF EQUIPMENT TEMPERATURE RESPONSES TO SELECTED HYDROGEN BURNS IN AN ICE CONDENSER CONTAINMENT.

MCDONALD,S.
NUREG-0714 V04-05: OCCUPATIONAL RADIATION EXPOSURE.Fifteenth And Sixteenth Annual Reports,1982 And 1983.

MCELROY,N.L.
NUREG-1134: RADIATION PROTECTION TRAINING FOR PERSONNEL EMPLOYED IN MEDICAL FACILITIES.

MCELROY,W.N.
NUREG/CR-3319: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium.
NUREG/CR-3746 V02: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.Semiannual Progress Report,April 1984 - September 1984.
NUREG/CR-3746 V03: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.1984 Annual Report,October 1,1983 - September 30, 1984.

MCEWEN,J.E.
NUREG/CR-4093: SAFETY/SAFEGUARDS INTERACTIONS DURING SAFETY-RELATED EMERGENCIES AT NUCLEAR POWER REACTOR FACILITIES.

MCGOWAN,J.J.
NUREG/CR-3978: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL PLATE AND WELDS FOR THE FOURTH HSST IRRADIATION SERIES.
NUREG/CR-4086: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL WELDS FOR THE THIRD HSST IRRADIATION SERIES.

MCGUIRE,M.V.
NUREG/CR-4139: THE MAILED SURVEY:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL.

MCGUIRE,R.K.
NUREG/CR-4145: EARTHQUAKE RECURRENCE INTERVALS AT NUCLEAR POWER PLANTS: ANALYSIS AND RANKING.

MCGUIRE,S.A.
NUREG-1140 DRFT FC: A REGULATORY ANALYSIS ON EMERGENCY PREPAREDNESS FOR FUEL CYCLE AND OTHER RADIOACTIVE MATERIAL LICENSEES.Draft Report For Comment.

MCKEON,T.J.
NUREG/CR-3972: SETTLEMENT OF URANIUM MILL TAILINGS PILES: A COMPARISON OF ANALYSIS TECHNIQUES.
NUREG/CR-4061: LEACHATE PLUME MIGRATION DOWNGRADIENT FROM URANIUM TAILINGS DISPOSAL IN MINE STOPES.

MEIER,J.K.
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.

MEISTER,H.
NUREG/CR-4055: THE D10 EXPERIMENT:COOLABILITY OF UO2 DEBRIS IN SODIUM WITH DOWNWARD HEAT REMOVAL.

MELBER,B.D.
NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.

MENGS,W.J.
NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT.

MENSING,R.W.
NUREG/CR-3660 V04: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 4:Pipe Failure Induced By Crack Growth In West Coast Plants.

MERKLE,J.G.
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.

MERONEY,R.N.
NUREG/CR-4072: THE ESTIMATION OF ATMOSPHERIC DISPERSION AT NUCLEAR POWER PLANTS UTILIZING REAL TIME ANEMOMETER STATISTICS.

MESSIER,M.E.
NUREG-0970: PROCEDURES FOR MEETING NRC ANTITRUST RESPONSIBILITIES.

METCALF,R.
NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT.

METZGER,V.
NUREG/CR-3208: TRAC-PD2 DEVELOPMENTAL ASSESSMENT.

MEYER,R.E.
NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4114: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS.II.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984.
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.

MEYER,R.O.
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).
NUREG/CR-4173: CORSOR USER'S MANUAL.

MILIAN,L.
NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.

MILLER,B.E.
NUREG/CR-3914: PUMP AND VALVE QUALIFICATION REVIEW GUIDE.

MILLER,C.A.
NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS.

MILLER,C.F.
NUREG/CR-4178 DRFT: AN EVALUATION OF SELECTED LICENSEE EVENT REPORTS PREPARED PURSUANT TO 10 CFR 50.73.Draft Report.

MILLER,D.D.
NUREG/CR-4170: AN ULTRA-HIGH SPEED RESIDUE PROCESSOR FOR SAFT INSPECTION SYSTEM IMAGE ENHANCEMENT.

MILLER,L.F.
NUREG/CR-3872: DATA ACQUISITION AND CONTROL OF THE HSST SERIES V IRRADIATION EXPERIMENT AT THE ORR.
NUREG/CR-4256: MEASUREMENT OF RESPONSE TIME AND DETECTION OF DEGRADATION IN PRESSURE SENSOR/SENSING LINE SYSTEMS.

MILLER,N.E.
NUREG/CR-3900 V02: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,July-September 1984.
NUREG/CR-3900 V03: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,October-December 1984.
NUREG/CR-3900 V04: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Annual Report,April 1984 - April 1985.
NUREG/CR-4379 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report,Year Four April-June 1985.

MILLER,R.L.
NUREG/CR-4090: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Annual Summary Report - Fiscal Year 1984.

MILLER,W.O.
NUREG/CR-3774 V04: ALTERNATIVE METHOD FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2C:Technical Requirements For Earth Mounded Concrete Bunker Disposal Of Low Level Radioactive Waste.

MILLS,W.R.
NUREG/CR-3791: CLOSEOUT OF IE BULLETIN 79-09:FAILURE OF GE TYPE AK-2 CIRCUIT BREAKERS IN SAFETY-RELATED SYSTEMS.

MILSTEAD,W.
NUREG-0933 S02: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-0933 S03: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

MINARICK,J.W.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

MINNERS,W.
NUREG-0933 S02: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-0933 S03: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

MITCHELL,D.H.
NUREG/CR-4023: FIELD PERFORMANCE ASSESSMENT OF SYNTHETIC LINERS FOR URANIUM TAILINGS POND.A Status Report.

MITCHELL,G.W.
NUREG/CR-2951: THE D9 EXPERIMENT.Heat Removal From Stratified UO2 Debris.
NUREG/CR-4055: THE D10 EXPERIMENT:COOLABILITY OF UO2 DEBRIS IN SODIUM WITH DOWNWARD HEAT REMOVAL.

MITCHELL,J.A.
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).

MITRA,S.
NUREG/CR-4400: THE IMPACT OF MECHANICAL-AND MAINTENANCE-INDUCED FAILURES OF MAIN REACTOR COOLANT PUMP SEALS ON PLANT SAFETY.

MIZNER,A.A.
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.

MJOLSNESS,R.C.
NUREG/CR-4079: ANALYTIC STUDIES PERTAINING TO STEAM GENERATOR TUBE RUPTURE ACCIDENTS.

MOAYERI,N.
NUREG/CR-4365: DESIGN AND DEVELOPMENT OF A SPECIAL PURPOSE SAFT SYSTEM FOR NONDESTRUCTIVE EVALUATION OF NUCLEAR REACTOR VESSELS AND PIPING COMPONENTS.

MOELLER,D.W.
NUREG/CR-4214: HEALTH EFFECTS MODEL FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Part I:Introduction,Integration & Summary.Part II:Scientific Basis For Health Effects Models.

MOELLER,M.P.
NUREG/CR-4160: HISTORICAL SUMMARY OF OCCUPATIONAL RADIATION EXPOSURE EXPERIENCE IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.

MOHR,C.M.
NUREG/CR-3633 V01 S1: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.

MOHR,P.B.
NUREG/CR-4035: A HIGHWAY ACCIDENT INVOLVING RADIOPHARMACEUTICALS NEAR BROOKHAVEN,MISSISSIPPI ON DECEMBER 3,1983.

MONROE,R.E.
NUREG/CR-3019: RECOMMENDED WELDED CRITERIA FOR USE IN THE FABRICATION OF SHIPPING CONTAINERS FOR RADIOACTIVE MATERIALS.

MONT,M.E.
NUREG/CR-4035: A HIGHWAY ACCIDENT INVOLVING RADIOPHARMACEUTICALS NEAR BROOKHAVEN,MISSISSIPPI ON DECEMBER 3,1983.

MOORE,E.B.
NUREG/CR-1755 ADD01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING NUCLEAR REACTORS AT MULTIPLE-REACTOR STATIONS.Effects On Decommissioning Of Interim Inability To Dispose Of Wastes Offsite.

MOORE,T.O.
NUREG/CR-3953: THE USE OF MAG-1 SPECTACLES WITH POSITIVE-AND NEGATIVE-PRESSURE RESPIRATORS.

MORETTI,E.S.
NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS.Application To Establishing Values Of f1 For Soluble Plutonium.

MORGENSTERN,M.
NUREG/CR-3817: DEVELOPMENT,USE AND CONTROL OF MAINTENANCE PROCEDURES IN NUCLEAR POWER PLANTS.Problems And Recommendations.
NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE.

MORISSEAU,D.S.
NUREG/CR-4139: THE MAILED SURVEY:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL.

MORRIS,B.M.
NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.

MORRISON,H.W.
NUREG/CR-3837: MULTIPLE-SEQUENTIAL FAILURE MODEL.Evaluation Of And Procedures For Human Error Dependency.

MOTES,B.G.
NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

MOUL,D.A.
NUREG/CR-4093: SAFETY/SAFEGUARDS INTERACTIONS DURING SAFETY-RELATED EMERGENCIES AT NUCLEAR POWER REACTOR FACILITIES.

MUHLHEIM,M.D.
NUREG/CR-3922 V01: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Main Report And Appendices A And B.
NUREG/CR-3922 V02: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Appendices C And D.

MULCAHEY,T.P.
NUREG/CR-4180: STATE-OF-THE-ART OF SOLID-STATE MOTOR CONTROLLERS.

MUNRO,P.S.
NUREG/CR-4317 V01: CANADIAN SEISMIC AGREEMENT.Technical Report Covering 1979-1985.

MUNSEY,J.W.
NUREG/CR-4288: FOCAL MECHANISM ANALYSES FOR VIRGINIA AND EASTERN TENNESSEE EARTHQUAKES (1978-1984).

MUNSON,L.F.
NUREG/CR-4399: POSSIBLE OPTIONS FOR REDUCING OCCUPATIONAL DOSE FROM THE TMI-2 BASEMENT.

MUNSON,L.H.
NUREG/CR-4160: HISTORICAL SUMMARY OF OCCUPATIONAL RADIATION EXPOSURE EXPERIENCE IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.

MURATA,K.K.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.

MURPHY,E.
NUREG-0844 DRFT FC: NRC INTEGRATED PROGRAM FOR RESOLUTION OF UNRESOLVED SAFETY ISSUES A-3,A-4 AND A-5 REGARDING STEAM GENERATOR TUBE INTEGRITY.Draft Report For Comment.

MURPHY,G.A.
NUREG/CR-3922 V01: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Main Report And Appendices A And B.
NUREG/CR-3922 V02: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Appendices C And D.
NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.

MUSCARA,J.
NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.
NUREG-1155 V04: RESEARCH PROGRAM PLAN.Non-Destructive Examination.
NUREG/CR-4361: STEAM GENERATOR GROUP PROJECT.Annual Report - 1983.
NUREG/CR-4362: STEAM GENERATOR GROUP PROJECT.Annual Report - 1984.

MYERS,D.A.
NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS.

NAKAGAKI,M.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

NALEZNY,C.L.
NUREG/CR-3005: SUMMARY OF THE NUCLEAR REGULATORY COMMISSION'S LOFT PROGRAM RESEARCH FINDINGS.
NUREG/CR-4351: SUMMARY REPORT FOR LOFT ANTICIPATED TRANSIENT EXPERIMENT SERIES L6-8.

NANSTAD,R.K.
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.
NUREG/CR-4249: PRESSURE VESSEL FRACTURE STUDIES PENETRATING TO THE PWR THERMAL-SHOCK ISSUE:EXPERIMENTS TSE-5,TSE-5A AND TSE-6.
NUREG/CR-4304: PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO THE PWR THERMAL-SHOCK ISSUE.Experiment TSE-7.

NASSERSHARIF,B.
NUREG/CR-4140: DOMINANT ACCIDENT SEQUENCES IN OCONEE-1 PRESSURIZED WATER REACTOR.

NELSON,L.S.
NUREG/CR-2718: STEAM EXPLOSION EXPERIMENTS WITH SINGLE DROPS OF IRON OXIDE MELTED WITH A CO2 LASER.Part II:Parametric Studies.

NELSON,R.A.
NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT.

NELSON,R.W.
NUREG/CR-4061: LEACHATE PLUME MIGRATION DOWNGRADIENT FROM URANIUM TAILINGS DISPOSAL IN MINE STOPES.

NELSON,W.
NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

NELSON,W.J.
NUREG/CR-4333: STE. GENEVIEVE FAULT ZONE,MISSOURI AND ILLINOIS.

NELSON,W.R.
NUREG/CR-4272: RESPONSE TREE EVALUATION:EXPERIMENTAL ASSESSMENT OF AN EXPERT SYSTEM FOR NUCLEAR REACTOR OPERATORS.

NETI,S.
NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT.

NEUMAN,S.P.
NUREG/CR-3736: FIELD AND THEORETICAL INVESTIGATIONS OF FRACTURED CRYSTALLINE ROCK NEAR ORACLE,ARIZONA.

NEWTON,R.D.
NUREG/CR-4133: NUCLEAR POWER SAFETY REPORTING SYSTEM IMPLEMENTATION AND OPERATIONAL SPECIFICATIONS.

NICHOLS,F.A.
NUREG/CR-4287: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Annual Report,October 1983 - September 1984.

NICHOLSON,P.R.
NUREG-0970: PROCEDURES FOR MEETING NRC ANTITRUST RESPONSIBILITIES.

NICHOLSON,T.J.
NUREG-1046: DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN UNSATURATED ZONE:TECHNICAL CONSIDERATIONS AND RESPONSE TO COMMENTS.

NICOLOSI,L.
NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.

NORWOOD,K.S.
NUREG/CR-3930: OBSERVED BEHAVIOR OF CESIUM,IODINE,AND TELLURIUM IN THE ORNL FISSION PRODUCT RELEASE PROGRAM.
NUREG/CR-4037: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-5.
NUREG/CR-4043: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-6.
NUREG/CR-4105: AN ASSESSMENT OF THERMAL GRADIENT TUBE RESULTS FROM THE HI SERIES OF FISSION PRODUCT RELEASE TESTS.

NOURBAKHSH,H.P.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

NOVICK,V.J.
NUREG/CR-4033: THE ROLE OF PERSONAL AIR SAMPLING IN RADIATION SAFETY PROGRAMS AND RESULTS OF A LABORATORY EVALUATION OF PERSONAL AIR-SAMPLING EQUIPMENT.

NOWATZKI,E.A.
NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984.

NOYCE,J.R.
NUREG/CR-4101: ASSAY OF LONG-LIVED RADIONUCLIDES IN LOW-LEVEL WASTES FROM POWER REACTORS.

NUKARI,N.H.
NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.

O'BRIEN,J.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-3485: PRA REVIEW MANUAL.
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP):A Methodology For Estimating Risk Impacts Of Safeguards Measures.

O'BRIEN,J.N.
NUREG/CR-3519: HUMAN ERROR PROBABILITY ESTIMATION USING LICENSEE EVENT REPORTS.
NUREG/CR-3837: MULTIPLE-SEQUENTIAL FAILURE MODEL.Evaluation Of And Procedures For Human Error Dependency.
NUREG/CR-4103: USES OF HUMAN RELIABILITY ANALYSIS PROBABILISTIC RISK ASSESSMENT RESULTS TO RESOLVE PERSONNEL PERFORMANCE ISSUES THAT COULD AFFECT SAFETY.

OBERLANDER,P.L.
NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUNDWATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions.
NUREG/CR-4251 V02: MITIGATIVE TECHNIQUES FOR GROUNDWATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 2:Case Study Analysis Of Hydrologic Characterization And Mitigative Schemes.

OGDEN,D.M.
NUREG/CR-3935: THERMAL-HYDRAULIC ANALYSES OF OVERCOOLING SEQUENCES FOR THE H.B. ROBINSON UNIT 2 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR.
NUREG/CR-4326 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Main Report.
NUREG/CR-4326 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Appendices.

OLDHAM,R.D.
NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS.Application To Establishing Values Of f1 For Soluble Plutonium.

OLSEN,C.S.
NUREG/CR-4056: PARTICULATE AND GAS RELEASE FROM LIGHT-WATER-REACTOR (LWR) FUEL RODS STORED IN INERT AND DRY AIR ATMOSPHERES.
NUREG/CR-4074: THE PERFORMANCE OF DEFECTED SPENT LWR FUEL RODS IN INERT GAS AND DRY AIR STORAGE ATMOSPHERES.
NUREG/CR-4084: DRY SPENT FUEL STORAGE TEST PLAN FOR DESTRUCTIVE FUEL ROD EXAMINATIONS.
NUREG/CR-4345: INVESTIGATION OF THE STABILITY OF LWR SPENT FUEL RODS BELOW 250 C.

OLSON,J.
NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE.
NUREG/CR-4125 V01: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 1:Guidelines For Utility Organization And Administration Plan.
NUREG/CR-4125 V02: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 2:Workbook For Assessment Of Organization And Management.
NUREG/CR-4281: AN EMPIRICAL ANALYSIS OF SELECTED NUCLEAR POWER PLANT MAINTENANCE FACTORS AND PLANT SAFETY.

OPITZ,B.E.
NUREG/CR-3906: URANIUM MILL TAILINGS NEUTRALIZATION:CONTAMINANT COMPLEXATION AND TAILINGS LEACHING STUDY.
NUREG/CR-4259: TAILINGS NEUTRALIZATION AND OTHER ALTERNATIVES FOR IMMOBILIZING TOXIC MATERIALS IN TAILINGS.Final Report.

OSBORN,R.N.
NUREG/CR-4125 V01: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 1:Guidelines For Utility Organization And Administration Plan.
NUREG/CR-4125 V02: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 2:Workbook For Assessment Of Organization And Management.
NUREG/CR-4281: AN EMPIRICAL ANALYSIS OF SELECTED NUCLEAR POWER PLANT MAINTENANCE FACTORS AND PLANT SAFETY.

OSBORNE,M.F.
NUREG/CR-3930: OBSERVED BEHAVIOR OF CESIUM,IODINE,AND TELLURIUM IN THE ORNL FISSION PRODUCT RELEASE PROGRAM.
NUREG/CR-4037: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-5.
NUREG/CR-4043: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-6.

OSTMEYER,R.H.
NUREG/CR-4185: AN ASSESSMENT OF DOSIMETRY DATA FOR ACCIDENTAL RADIONUCLIDE RELEASES FROM NUCLEAR REACTORS.

OSTMEYER,R.M.
NUREG/CR-4169: AN APPROACH TO TREATING RADIONUCLIDE DECAY HEATING FOR USE IN THE MELCOR CODE SYSTEM.

OTTINGER,C.A.
NUREG/CR-2951: THE D9 EXPERIMENT.Heat Removal From Stratified UO2 Debris.
NUREG/CR-4055: THE D10 EXPERIMENT:COOLABILITY OF UO2 DEBRIS IN SODIUM WITH DOWNWARD HEAT REMOVAL.

OUELLETTE,A.L.
NUREG/CR-3197 V01: REACTION BETWEEN SOME CESIUM-IODINE COMPOUNDS AND THE REACTOR MATERIALS 304 STAINLESS STEEL,INCONEL 600 & SILVER. Volume I:Cesium Hydroxide Reactions.

OWCA,W.A.
NUREG/CR-4164: DATA REPORT FOR THE TPFL TEE/CRITICAL FLOW EXPERIMENTS.

OWCZARSKI,P.C.
NUREG/CR-3317: TECHNICAL BASES AND USER'S MANUAL FOR THE PROTOTYPE OF SPARC - A SUPPRESSION POOL AEROSOL REMOVAL CODE.
NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.

OWINGS,T.D.
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.

PAGE,R.E.
NUREG/CR-3613 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Annual Report For 1984.
NUREG/CR-3613 V03 N1: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Semiannual Report For October 1984 Through March 1985.
NUREG/CR-3911 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Quarterly Report,April-June 1984.

PAIK,C.Y.
NUREG/CR-4166: ANALYSIS OF FLECHT-SEASET 163-ROD BLOCKED BUNDLE DATA USING COBRA-TF.

PALAZZO,R.J.
NUREG/CR-3516: A SURVEY OF THE USES OF RADIOACTIVE MATERIALS IN LOUISIANA'S OFFSHORE WATERS.

PAPASPYROPOULOS
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

PAPAZOGLOU,I.A.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-3485: PRA REVIEW MANUAL.
NUREG/CR-4228: REVIEW OF THE VOGTLE UNITS 1 AND 2 AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS.

PAPPAS,R.A.
NUREG/CR-3915: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL.

PARK,J.Y.
NUREG/CR-4287: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Annual Report,October 1983 - September 1984.

PARKER,F.
NUREG/CR-4396: SIMMER POSTPROCESSOR MANUAL.

PARKER,G.B.
NUREG/CR-4075: DESIGNING PROTECTIVE COVERS FOR URANIUM MILL TAILINGS PILES. A Review.

PARKER,S.F.
NUREG/CR-4159: COMPARISON OF THE 1981 INEL DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS.

PARLOS,A.
NUREG/CR-4376: HEAT TRANSFER,CARRYOVER AND FALL BACK IN PWR STEAM GENERATORS DURING TRANSIENTS.

PASCHOA,A.S.
NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.

PASEDAG,W.F.
NUREG-0856 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).

PASUPATHI,V.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

PEABODY,C.A.
NUREG-0856 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).

PELOQUIN,R.A.
NUREG/CR-2850 V03: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1981.

PELTO,P.J.
NUREG/CR-4220: RELIABILITY ANALYSIS OF CONTAINMENT ISOLATION SYSTEMS.

PENG,S.J.
NUREG/CR-4116: NUFEGQ-NP:A DIGITAL COMPUTER CODE FOR THE LINEAR STABILITY ANALYSIS OF BOILING WATER NUCLEAR REACTORS.

PESCATORE,C.
NUREG/CR-2482 V08: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period October 1984 - March 1985.

PETERSON,A.C.
NUREG/CR-3772: RELAP5 ASSESSMENT:SEMISCALE SMALL BREAK TESTS S-UT-1,S-UT-2, S-UT-6,S-UT-7 AND S-UT-8.
NUREG/CR-3919: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:NEPTUNUS PRESSURIZER TEST Y05.

PETRYKOWSKI,J.
NUREG/CR-4346: AEROSOL RELEASE EXPERIMENTS IN THE FUEL AEROSOL SIMULANT TEST FACILITY:UNDERSODIUM EXPERIMENTS.

PHILIPPACOPOULO
NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS.

PHILLIPS,L.D.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

PICIULO,P.L.
NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.
NUREG/CR-4069: ANALYSES OF SOILS FROM AN AREA ADJACENT TO THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE AT SHEFFIELD,ILLINOIS.
NUREG/CR-4083: ANALYSES OF SOILS FROM THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITES AT BARNWELL,SC AND RICHLAND,WA.
NUREG/CR-4200: BIODEGRADATION TESTING OF SOLIDIFIED LOW-LEVEL WASTE STREAMS.
NUREG/CR-4201: THERMAL STABILITY TESTING OF LOW-LEVEL WASTE FORMS.
NUREG/CR-4406: AN ANALYSIS OF LOW-LEVEL WASTES:Review of Hazardous Waste Regulations And Identification of Radioactive Mixed Wastes. Final Report.

PILCH,M.
NUREG/CR-4383: HIGH PRESSURE INJECTION OF MELT FROM A REACTOR PRESSURE VESSEL - THE DISCHARGE PHASE.

PILGRIM,M.K.
NUREG/CR-4093: SAFETY/SAFEGUARDS INTERACTIONS DURING SAFETY-RELATED EMERGENCIES AT NUCLEAR POWER REACTOR FACILITIES.

PIN,F.G.
NUREG/CR-4403: SUMMARY OF THE WASTE MANAGEMENT PROGRAMS AT URANIUM RECOVERY FACILITIES AS THEY RELATE TO THE 40 CFR PART 192 STANDARDS.

PIRES,J.
NUREG/CR-4329: RELIABILITY EVALUATION OF CONTAINMENTS INCLUDING SOIL-STRUCTURE INTERACTION.

PITTMAN,J.
NUREG-0933 S02: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-0933 S03: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

PLOGER,S.A.
NUREG/CR-3948: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY.

PODOWSKI,M.Z.
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.
NUREG/CR-4116: NUFEGQ-NP:A DIGITAL COMPUTER CODE FOR THE LINEAR STABILITY ANALYSIS OF BOILING WATER NUCLEAR REACTORS.

POLICASTRO,A.J.
NUREG/CR-4120: MATHEMATICAL MODELING OF ULTIMATE HEAT SINK COOLING PONDS.

POLKY,J.N.
NUREG/CR-4170: AN ULTRA-HIGH SPEED RESIDUE PROCESSOR FOR SAFT INSPECTION SYSTEM IMAGE ENHANCEMENT.

POLLACK,H.N.
NUREG/CR-31[illegible]5 V03: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION - ANNUAL REPORT.(October 1982 - September 1983, Volume 3).

POLOSKI,J.P.
NUREG/CR-3301: CATALOG OF PRA DOMINANT ACCIDENT SEQUENCE INFORMATION.

POORE,W.P.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.

POPELAR,C.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

POSTMA,A.K.
NUREG/CR-3317: TECHNICAL BASES AND USER'S MANUAL FOR THE PROTOTYPE OF SPARC - A SUPPRESSION POOL AEROSOL REMOVAL CODE.

POWERS,D.A.
NUREG/CR-4401: BEHAVIOR OF CONTROL RODS DURING CORE DEGRADATION: PRESSURIZATION OF SILVER-INDIUM-CADMIUM CONTROL RODS.

POWERS,T.B.
NUREG/CR-2800 S03: GUIDELINES FOR NUCLEAR POWER PLANT SAFETY ISSUE PRIORITIZATION INFORMATION DEVELOPMENT.

PRAIRIE,R.R.
NUREG/CR-4350 V02: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 2-Probability And Statistics For PRA Applications.

PRATT,W.
NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences.

PREVOST,J.
NUREG/CR-4294: LEAK RATE ANALYSIS OF THE WESTINGHOUSE REACTOR COOLANT PUMP.

PRICE,D.S.
NUREG/CR-3516: A SURVEY OF THE USES OF RADIOACTIVE MATERIALS IN LOUISIANA'S OFFSHORE WATERS.

PRICE,J.C.
NUREG/CR-4131: INVESTIGATION OF ALTERNATIVE MEANS TO ACCOMPLISH THE GOALS OF BIENNIAL ION CHAMBER CALIBRATION.

PRICE,J.M.
NUREG/CR-3537: EXPEDIENT METHODS OF RESPIRATORY PROTECTION.III. SUBMICRON PARTICLE TESTS AND SUMMARY OF QUALITY FACTORS.

PRINE,D.W.
NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984.
NUREG/CR-4368: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRs: Semiannual Report,October 1984 - March 1985.

PRUITT,J.S.
NUREG/CR-4266: STANDARD BETA-PARTICLE AND MONOENERGETIC ELECTRON SOURCES FOR THE CALIBRATION OF BETA-RADIATION PROTECTION INSTRUMENTATION.

PU,J.
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.

PUGH,C.E.
NUREG/CR-3744 V02: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR APRIL-SEPTEMBER 1984.
NUREG/CR-4219 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.

RADFORD,L.R.
NUREG/CR-3817: DEVELOPMENT,USE AND CONTROL OF MAINTENANCE PROCEDURES IN NUCLEAR POWER PLANTS.Problems And Recommendations.

RAGAN,G.E.
NUREG/CR-4256: MEASUREMENT OF RESPONSE TIME AND DETECTION OF DEGRADATION IN PRESSURE SENSOR/SENSING LINE SYSTEMS.

RAGHURAM,S.
NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS.
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.

RAMIREZ,A.L.
NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt.

RANDOLPH,P.D.
NUREG/CR-3237: CONTROL OF EXPLOSIVE MIXTURES IN PWR WASTE GAS SYSTEMS.

RANKIN,W.L.
NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS.

RANSOM,C.B.
NUREG/CR-4262 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR.Main Report.
NUREG/CR-4262 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR. Appendices.
NUREG/CR-4326 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Main Report.
NUREG/CR-4326 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Appendices.

RAO,K.S.
NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE.

RATHBUN,L.A.
NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.

RATLIFF,R.A.
NUREG/CR-4352: SUGGESTED STATE REQUIREMENTS AND CRITERIA FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE REGULATORY PROGRAM.

RATZEL,A.C.
NUREG/CR-4136: SMOKE.A Data Reduction Package For Analysis Of Combustion Experiments.
NUREG/CR-4138: DATA ANALYSES FOR NEVADA TEST SITE (NTS) PREMIXED COMBUSTION TESTS.

RAVINDRA,M.K.
NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3663 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 3:Double Ended Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-4290 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF BABCOCK AND WILCOX PWR PLANTS.Volume 2:Guillotine Break Indirectly Induced By Earthquakes.

REECE,W.D.
NUREG/CR-4297: EXTREMITY MONITORING:Considerations For Use,Dosimeter Placement,And Evaluation.

REED,A.W.
NUREG/CR-4136: SMOKE.A Data Reduction Package For Analysis Of Combustion Experiments.

REED,J.W.
NUREG/CR-2815 V02 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 8-12.
NUREG/CR-3485: PRA REVIEW MANUAL.
NUREG/CR-4334: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS.

REED,K.A.
NUREG/CR-3953: THE USE OF MAG-1 SPECTACLES WITH POSITIVE- AND NEGATIVE-PRESSURE RESPIRATORS.

REESE,R.T.
NUREG/CR-3647: DESIGN AND FABRICATION OF A 1/8-SCALE STEEL CONTAINMENT MODEL.

REICH,M.
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.
NUREG/CR-4149: ULTIMATE PRESSURE CAPACITY OF REINFORCED AND PRESTRESSED CONCRETE CONTAINMENT.
NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS.
NUREG/CR-4221: AN EVALUATION OF STRESS CORROSION CRACK GROWTH IN BWR PIPING SYSTEMS.
NUREG/CR-4329: RELIABILITY EVALUATION OF CONTAINMENTS INCLUDING SOIL-STRUCTURE INTERACTION.

REMEC,I.
NUREG/CR-4031 V02: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutronics Calculations."
NUREG/CR-4031 V03: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutron Exposure Parameters."

REST,J.
NUREG/CR-3980 V02: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,April-June 1984.
NUREG/CR-3980 V03: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,July-September 1984.
NUREG/CR-3980 V04: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,October-December 1984.

REXROTH,P.E.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.

RHODES,D.B.
NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT.

RIANI,L.
NUREG-0933 S02: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-0933 S03: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

RICH,B.L.
NUREG/CR-4033: THE ROLE OF PERSONAL AIR SAMPLING IN RADIATION SAFETY PROGRAMS AND RESULTS OF A LABORATORY EVALUATION OF PERSONAL AIR-SAMPLING EQUIPMENT.

RICHARDSON,E.
NUREG-0714 V04-05: OCCUPATIONAL RADIATION EXPOSURE.Fi enth And Sixteenth Annual Reports,1982 And 1983.

RIDEOUT,T.B.
NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS.

RIGGS,R.
NUREG-..33 S02: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-0933 S03: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-1128: TRIAL EVALUATIONS IN COMPARISON WITH THE 1983 SAFETY GOALS.

RIORDAN,B.
NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

RITCHIE,L.T.
NUREG/CR-4350 V07: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 7 - Environmental Transport And Consequence Analysis.

RITTER,P.D.
NUREG/CR-4033: THE ROLE OF PERSONAL AIR SAMPLING IN RADIATION SAFETY PROGRAMS AND RESULTS OF A LABORATORY EVALUATION OF PERSONAL AIR-SAMPLING EQUIPMENT.

ROBERSON,P.L.
NUREG/CR-4297: EXTREMITY MONITORING:Considerations For Use,Dosimeter Placement,And Evaluation.

ROBERTSON,D.E.
NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER.(Final Report)

ROBINSON,G.C.
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.

ROCHELLE,J.M.
NUREG/CR-4346: AEROSOL RELEASE EXPERIMENTS IN THE FUEL AEROSOL SIMULANT TEST FACILITY:UNDERSODIUM EXPERIMENTS.

RODABAUGH,E.C.
NUREG/CR-4305: COMMENTS ON THE LEAK-BEFORE-BREAK CONCEPT FOR NUCLEAR POWER PLANT PIPING SYSTEMS.

ROHATGI,U.S.
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-4253: REVIEW OF TRAC CALCULATIONS FOR CALVERT CLIFFS PTS STUDY.
NUREG/CR-4292: A COMPARATIVE ANALYSIS OF CONSTITUTIVE RELATIONS IN TRAC-PFL AND RELAP5/MOD1.

ROLLER,S.F.
NUREG/CR-3721 V01: PRESSURE MEASUREMENTS IN A HYDROGEN COMBUSTION ENVIRONMENT. Hydrogen-Air Combustion Test Series 1 And 2 In The FITS Tank.

ROONEY,J.J.
NUREG/CR-3922 V01: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Main Report And Appendices A And B.
NUREG/CR-3922 V02: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Appendices C And D.

ROSA,E.A.
NUREG/CR-4016 V01: APPLICATION OF SLIM-MAUD:A TEST OF AN INTERACTIVE COMPUTER-BASED METHOD FOR ORGANIZING EXPERT ASSESSMENT OF HUMAN PERFORMANCE AND RELIABILITY.Volume I:Main Report.

ROSCOE,B.J.
NUREG/CR-4250: VEHICLE BARRIERS:EMPHASIS ON NATURAL FEATURES.

ROSE,J.A.
NUREG/CR-3819: SURVEY OF AGED POWER PLANT FACILITIES.

ROSS,P.A.
NUREG-0020 V09 N04: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of March 31,1985.(Gray Book I)
NUREG-0020 V09 N06: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of May 31,1985.(Gray Book I)
NUREG-0020 V09 N07: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of June 30,1985.(Gray Book I)
NUREG-0020 V09 N09: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of August 31,1985.(Gray Book I)
NUREG-0020 V09 N11: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of October 31,1985.(Gray Book I)

RUGER,C.
NUREG/CR-2815 V02 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 8-12.
NUREG/CR-4229: EVALUATION OF CURRENT METHODOLOGY EMPLOYED IN PROBABILISTIC RISK ASSESSMENT (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.

RUNKLE,G.E.
NUREG/CR-3657: PRELIMINARY SCREENING OF FUEL CYCLE AND BY-PRODUCT MATERIAL LICENSES FOR EMERGENCY PLANNING.
NUREG/CR-4185: AN ASSESSMENT OF DOSIMETRY DATA FOR ACCIDENTAL RADIONUCLIDE RELEASES FROM NUCLEAR REACTORS.

RUTHER,W.E.
NUREG/CR-4287: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Annual Report,October 1983 - September 1984.

RYDER,C.P.
NUREG-0856 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment)

SAARI,L.M.
NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.

SACCOMANNO,G.
NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.

SAGENDORF,J.F.
NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.

SAHA,P.
NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-4252: INDEPENDENT ASSESSMENT OF TRAC-PD2/MOD1 CODE WITH BCL ECC BYPASS TESTS.

SAHOTA,M.S.
NUREG/CR-4278: TRAC-PF1/MOD1 DEVELOPMENT ASSESSMENT.

SAILOR,V.L.
NUREG/CR-4067: SUMMARY OF BARRIER DEGRADATION EVENTS AND SMALL ACCIDENTS IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.
NUREG/CR-4068: SUMMARY OF HISTORICAL EXPERIENCE WITH RELEASES OF RADIOACTIVE MATERIALS FROM COMMERCIAL NUCLEAR POWER PLANTS IN THE UNITED STATES.

SALAZAR,E.A.
NUREG/CR-3863: ASSESSMENT OF CLASS 1E PRESSURE TRANSMITTER RESPONSE WHEN SUBJECTED TO HARSH ENVIRONMENT SCREENING TESTS.

SALLACH,R.A.
NUREG/CR-3197 V01: REACTION BETWEEN SOME CESIUM-IODINE COMPOUNDS AND THE REACTOR MATERIALS 304 STAINLESS STEEL,INCONEL 600 & SILVER. Volume I:Cesium Hydroxide Reactions.

SAMANTA,P.K.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.
NUREG/CR-3837: MULTIPLE-SEQUENTIAL FAILURE MODEL.Evaluation Of And Procedures For Human Error Dependency.
NUREG/CR-4231: EVALUATION OF AVAILABLE DATA FOR PROBABILISTIC RISK ASSESSMENTS (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.

SANDERS,R.D.
NUREG/CR-4150: EPICOR-II RESIN DEGRADATION RESULTS FROM FIRST RESIN SAMPLES OF PF-8 AND PF-20.

SAWYER,W.D.
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

SCHELLING,F.J.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.

SCHMIDT,C.T.
NUREG/CR-4355 V01: 238 PU(IV) IN MONKEYS.Overview Of Metabolism.

SCHMULT,B.
NUREG/CR-4365: DESIGN AND DEVELOPMENT OF A SPECIAL PURPOSE SAFT SYSTEM FOR NONDESTRUCTIVE EVALUATION OF NUCLEAR REACTOR VESSELS AND PIPING COMPONENTS.

SCHNITZLER,B.G.
NUREG/CR-4203: A CALCULATIONAL METHOD FOR DETERMINING BIOLOGICAL DOSE RATES FROM IRRADIATED RESEARCH REACTOR FUEL.

SCHRECK,R.I.
NUREG/CR-3317: TECHNICAL BASES AND USER'S MANUAL FOR THE PROTOTYPE OF SPARC - A SUPPRESSION POOL AEROSOL REMOVAL CODE.
NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.

SCHWARTZ,M.W.
NUREG/CR-4035: A HIGHWAY ACCIDENT INVOLVING RADIOPHARMACEUTICALS NEAR BROOKHAVEN,MISSISSIPPI ON DECEMBER 3,1983.

SCHWARTZ,R.B.
NUREG/CR-4266: STANDARD BETA-PARTICLE AND MONOENERGETIC ELECTRON SOURCES FOR THE CALIBRATION OF BETA-RADIATION PROTECTION INSTRUMENTATION.

SCHWEIZER,R.L.
NUREG/CR-4093: SAFETY/SAFEGUARDS INTERACTIONS DURING SAFETY-RELATED EMERGENCIES AT NUCLEAR POWER REACTOR FACILITIES.

SCIACCA,F.
NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

SCIACCA,F.W.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.

SCOTT,P.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

SEARS,R.G.
NUREG/CR-3019: RECOMMENDED WELDED CRITERIA FOR USE IN THE FABRICATION OF SHIPPING CONTAINERS FOR RADIOACTIVE MATERIALS.

SEAVER,D.A.
NUREG/CR-3688 V01: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 1:Main Report.
NUREG/CR-3688 V02: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 2:Appendices.

SEELEY,F.G.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.REPORT FOR OCTOBER-DECEMBER 1984.
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.

SEGE,G.
NUREG-0933 S02: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-0933 S03: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-1128: TRIAL EVALUATIONS IN COMPARISON WITH THE 1983 SAFETY GOALS.

SEGOL,G.
NUREG/CR-3901: DOCUMENTATION AND USER'S GUIDE:GS2 & GS3 - VARIABLY SATURATED FLOW AND MASS TRANSPORT MODELS.

SEITZ,M.G.
NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT.

SELBY,D.L.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

SENGLAUB,M.E.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.

SERKIZ,A.W.
NUREG-0869 R01: USI A-43 REGULATORY ANALYSIS.
NUREG-0897 R01: CONTAINMENT EMERGENCY SUMP PERFORMANCE.(Technical Findings Related To Unresolved Safety Issues).

SERNE,R.J.
NUREG/CR-3709: METHODS OF MINIMIZING GROUND-WATER CONTAMINATION FROM IN SITU LEACH URANIUM MINING.Final Report.
NUREG/CR-3906: URANIUM MILL TAILINGS NEUTRALIZATION:CONTAMINANT COMPLEXATION AND TAILINGS LEACHING STUDY.
NUREG/CR-4259: TAILINGS NEUTRALIZATION AND OTHER ALTERNATIVES FOR IMMOBILIZING TOXIC MATERIALS IN TAILINGS.Final Report.

SERPAN,C.Z.
NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.

SERVER,W.L.
NUREG/CR-4212: IN-PLACE THERMAL ANNEALING OF NUCLEAR REACTOR PRESSURE VESSELS.

SHA,W.T.
NUREG/CR-3989: TIME- AND VOLUME-AVERAGED CONSERVATION EQUATIONS FOR MULTIPHASE FLOW.Part One:System Without Internal Solid Structures.

SHACK,W.J.
NUREG/CR-3998 V02: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,April-June 1984.
NUREG/CR-3998 V03: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,October-December 1984.
NUREG/CR-4287: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Annual Report,October 1983 - September 1984.

SHACKELFORD,M.
NUREG/CR-3855: CHARACTERIZATION OF NUCLEAR REACTOR CONTAINMENT PENETRATION - FINAL REPORT.

SHAFAGHI,A.
NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.

SHAFER,J.M.
NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions.
NUREG/CR-4251 V02: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 2:Case Study Analysis Of Hydrologic Characterization And Mitigative Schemes.

SHAH,V.N.
NUREG/CR-3948: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY.

SHAPIRO,B.J.
NUREG/CR-3301: CATALOG OF PRA DOMINANT ACCIDENT SEQUENCE INFORMATION.

SHARMA,S.
NUREG/CR-4149: ULTIMATE PRESSURE CAPACITY OF REINFORCED AND PRESTRESSED CONCRETE CONTAINMENT.
NUREG/CR-4221: AN EVALUATION OF STRESS CORROSION CRACK GROWTH IN BWR PIPING SYSTEMS.

SHAUG,J.C.
NUREG/CR-4127 V02: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 2:Models.
NUREG/CR-4127 V03: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 3:Development Assessment For Plant Application.

SHEA,C.E.
NUREG/CR-4069: ANALYSES OF SOILS FROM AN AREA ADJACENT TO THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE AT SHEFFIELD,ILLINOIS.
NUREG/CR-4083: ANALYSES OF SOILS FROM THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITES AT BARNWELL,SC AND RICHLAND,WA.
NUREG/CR-4200: BIODEGRADATION TESTING OF SOLIDIFIED LOW-LEVEL WASTE STREAMS.

SHENG,Y.P.
NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING.

SHEPHERD,J.E.
NUREG/CR-3638: HYDROGEN-STEAM JET-FLAME FACILITY AND EXPERIMENTS.
NUREG/CR-4136: SMOKE:A Data Reduction Package For Analysis Of Combustion Experiments.

SHERWOOD,D.R.
NUREG/CR-4259: TAILINGS NEUTRALIZATION AND OTHER ALTERNATIVES FOR IMMOBILIZING TOXIC MATERIALS IN TAILINGS.Final Report.

SHIEH,L.C.
NUREG/CR-4331: SIMPLIFIED SEISMIC PROBABILISTIC RISK ASSESSMENT.Procedures And Limitations.

SHINOZUKA,M.
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.
NUREG/CR-4334: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS.

SHIRE,P.R.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.

SHIU,K.
NUREG/CR-4050: A REVIEW OF THE SHOREHAM NUCLEAR POWER STATION PROBABILISTIC RISK ASSESSMENT.Internal Events And Core Damage Frequency.

SHIU,K.K.
NUREG/CR-2815 V02 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 8-12.
NUREG/CR-3485: PRA REVIEW MANUAL.

SHOR,R.W.
NUREG/CR-3738: ENVIRONMENTAL EFFECTS OF THE URANIUM FUEL CYCLE.A Review Of Data For Technetium.

SHORT,S.A.
NUREG/CR-3805 V02: ENGINEERING CHARACTERIZATION OF GROUND MOTION.Task II: Effects Of Ground Motion Characteristics On Structural Response Considering Localized Structural Nonlinearities And Soil-Structure Interaction Effects.

SHORTENCARIER
NUREG/CR-4122: A FORTRAN 77 PROGRAM AND USER'S GUIDE FOR THE CALCULATION OF PARTIAL CORRELATION AND STANDARDIZED REGRESSION COEFFICIENTS.

SHTEYNGART,S.
NUREG/CR-4291: CONCLUSION AND SUMMARY REPORT ON PHYSICAL BENCHMARKING OF PIPING SYSTEMS.

SHUMWAY,R.W.
NUREG/CR-3633 V01 S1: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.
NUREG/CR-3633 V04: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.Volume 4: Developmental Assessment.

SIEGEL,A.I.
NUREG/CR-3626 V02: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: DESCRIPTION OF MODEL CONTENT,STRUCTURE,AND SENSITIVITY TESTING.
NUREG/CR-4104: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL. Field Evaluation/Validation.

SIEGEL,M.D.
NUREG/CR-4110: REPOSITORY SITE DATA REPORT FOR UNSATURATED TUFF,YUCCA MOUNTAIN,NEVADA.

SILBERBERG,M.
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).

SILVER,E.G.
NUREG/CR-3430 V02: NUCLEAR POWER PLANT OPERATING EXPERIENCE - 1982.Annual Report.

SIMMONS,M.A.
NUREG/CR-4268: RATIO METHODS FOR COST-EFFECTIVE FIELD SAMPLING OF COMMERCIAL RADIOACTIVE LOW-LEVEL WASTES.

SIMONEN,E.P.
NUREG/CR-4287: VESSEL INTEGRITY SIMULATION (VISA) CODE SENSITIVITY STUDY.

SIMONEN,F.A.
NUREG/CR-4267: VESSEL INTEGRITY SIMULATION (VISA) CODE SENSITIVITY STUDY.

SIMPKINS,B.
NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

SIMPSON,E.S.
NUREG/CR-3736: FIELD AND THEORETICAL INVESTIGATIONS OF FRACTURED CRYSTALLINE ROCK NEAR ORACLE,ARIZONA.

SIMPSON,F.B.
NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION.

SIMPSON,H.J.
NUREG/CR-4094: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN SULFATE LAKE ENVIRONMENTS.
NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.

SINGER,G.L.
NUREG/CR-3633 V01 S1: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.

SINGH,N.P.
NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.

SISKIND,B.
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.
NUREG/CR-4062: EXTENDED STORAGE OF LOW-LEVEL RADIOACTIVE WASTES:Potential Problem Areas.
NUREG/CR-4406: AN ANALYSIS OF LOW-LEVEL WASTES:Review of Hazardous Waste Regulations And Identification of Radioactive Mixed Wastes. Final Report.

SKAGGS,R.L.
NUREG/CR-3752: EFFECTS OF HYDROLOGIC VARIABLES ON ROCK RIPRAP DESIGN FOR URANIUM TAILINGS IMPOUNDMENTS.
NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions.
NUREG/CR-4251 V02: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 2:Case Study Analysis Of Hydrologic Characterization And Mitigative Schemes.

SKORPIK,J.R.
NUREG/CR-3915: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL.

SLOVIK,G.C.
NUREG/CR-4252: INDEPENDENT ASSESSMENT OF TRAC-PD2/MOD1 CODE WITH BCL ECC BYPASS TESTS.
NUREG/CR-4292: A COMPARATIVE ANALYSIS OF CONSTITUTIVE RELATIONS IN TRAC-PFL AND RELAP5/MOD1.

SMALLEY,J.F.
NUREG/CR-3865: EVALUATION OF THE RADIOACTIVE INVENTORY IN,AND ESTIMATION OF ISOTOPIC RELEASE FROM,THE WASTE IN EIGHT TRENCHES AT THE SHEFFIELD LOW-LEVEL WASTE BURIAL SITE.

SMITH,B.W.
NUREG/CR-4108: DEVELOPMENT OF MC&A ALARM RESOLUTION PROCEDURES.

SMITH,C.R.
NUREG/CR-4117: FAULTING AND JOINTING IN AND NEAR SURFACE MINES OF SOUTHWESTERN INDIANA.

SMITH,F.J.
NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.

SMITH,J.H.
NUREG/CR-3949 V01: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Semiannual Progress Report For Period Ending June 30,1984.
NUREG/CR-3949 V02: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Annual Progress Report For Period Ending December 31,1984.

SMITH,K.L.
NUREG/CR-3862: DEVELOPMENT OF TRANSIENT INITIATING EVENT FREQUENCIES FOR USE IN PROBABILISTIC RISK ASSESSMENTS.

SMITH,P.D.
NUREG/CR-4331: SIMPLIFIED SEISMIC PROBABILISTIC RISK ASSESSMENT.Procedures And Limitations.

SMITH,P.R.
NUREG/CR-4225: SUMMARY OF EFFICIENCY TESTING OF STANDARD AND HIGH-CAPACITY HIGH-EFFICIENCY PARTICULATE AIR FILTERS SUBJECTED TO SIMULATED TORNADO DEPRESSURIZATION AND EXPLOSIVE SHOCK WAVES.
NUREG/CR-4232: THE RESPONSE OF VENTILATION DAMPERS TO LARGE AIRFLOW PULSES.

SMITH,R.C.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4109: TRAC-PF1 ANALYSES OF POTENTIAL PRESSURIZED-THERMAL SHOCK TRANSIENTS AT CALVERT CLIFFS/UNIT 1.A Combustion Engineering PWR.

SOARES,C.G.
NUREG/CR-4266: STANDARD BETA-PARTICLE AND MONOENERGETIC ELECTRON SOURCES FOR THE CALIBRATION OF BETA-RADIATION PROTECTION INSTRUMENTATION.

SOMMERS,P.E.
NUREG/CR-4125 V01: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 1:Guideline For Utility Organization And Administration Plan.
NUREG/CR-4125 V02: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 2:Workbook For Assessment Of Organization And Management.
NUREG/CR-4281: AN EMPIRICAL ANALYSIS OF SELECTED NUCLEAR POWER PLANT MAINTENANCE FACTORS AND PLANT SAFETY.

SOO,P.
NUREG/CR-2482 V06: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program.October 1983 - March 1984.
NUREG/CR-2482 V07: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program April 1984-September 1984.
NUREG/CR-2482 V08: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period October 1984 - March 1985.
NUREG/CR-2482 V09: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period April-September 1985.
NUREG/CR-3091 V04: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1983 - March 1984.
NUREG/CR-3091 V05: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period April 1984 - September 1984.
NUREG/CR-3091 V06: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1984 - March 1985.

SOO,S.L.
NUREG/CR-3989: TIME- AND VOLUME-AVERAGED CONSERVATION EQUATIONS FOR MULTIPHASE FLOW.Part One:System Without Internal Solid Structures.

SOONG,A.L.
NUREG-1127: RADIATION PROTECTION TRAINING AT URANIUM HEXAFLUORIDE AND FUEL FABRICATION PLANTS.

SPALETTO,M.I.
NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS.Application To Establishing Values Of f1 For Soluble Plutonium.

SPANNER,G.E.
NUREG/CR-4023: FIELD PERFORMANCE ASSESSMENT OF SYNTHETIC LINERS FOR URANIUM TAILINGS POND.A Status Report.

SPETTELL,C.M.
NUREG/CR-4016 V01: APPLICATION OF SLIM-MAUD:A TEST OF AN INTERACTIVE COMPUTER-BASED METHOD FOR ORGANIZING EXPERT ASSESSMENT OF HUMAN PERFORMANCE AND RELIABILITY.Volume I.Main Report.
NUREG/CR-4103: USES OF HUMAN RELIABILITY ANALYSIS PROBABILISTIC RISK ASSESSMENT RESULTS TO RESOLVE PERSONNEL PERFORMANCE ISSUES THAT COULD AFFECT SAFETY.

SPRIGGS,G.D.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4109: TRAC-PF1 ANALYSES OF POTENTIAL PRESSURIZED-THERMAL SHOCK TRANSIENTS AT CALVERT CLIFFS/UNIT 1.A Combustion Engineering PWR.

SRINIVASAN,M.G.
NUREG/CR-4432: COMPARISON OF DYNAMIC CHARACTERISTICS OF FUKUSHIMA NUCLEAR POWER PLANT CONTAINMENT BUILDING DETERMINED FROM TESTS AND EARTHQUAKES.

STAHL,D.
NUREG/CR-3900 V02: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,July-September 1984.
NUREG/CR-3900 V03: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,October-December 1984.
NUREG/CR-3900 V04: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Annual Report,April 1984 - April 1985.
NUREG/CR-4379 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report,Year Four April-June 1985.

STALKER,A.C.
NUREG/CR-4397: IN-PLANT SOURCE TERM MEASUREMENTS AT PRAIRIE ISLAND NUCLEAR GENERATING STATION.

STALLMANN,F.W.
NUREG/CR-4031 V03: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutron Exposure Parameters."
NUREG/CR-4284: NEUTRON EXPOSURE PARAMETERS FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY IRRADIATION SERIES.

STAMATELATOS,M.
NUREG/CR-4303: HIGH-LEVEL WASTE PRECLOSURE SYSTEMS SAFETY ANALYSIS.Phase 1, Final Report.

START,G.E.
NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.

STAUB,W.P.
NUREG/CR-4403: SUMMARY OF THE WASTE MANAGEMENT PROGRAMS AT URANIUM RECOVERY FACILITIES AS THEY RELATE TO THE 40 CFR PART 192 STANDARDS.

STEELE,R.
NUREG/CR-3819: SURVEY OF AGED POWER PLANT FACILITIES.
NUREG/CR-4141: CONTAINMENT PURGE AND VENT VALVE TEST PROGRAM FINAL REPORT.

STELZMAN,W.J.
NUREG/CR-4092: ORNL CHARACTERIZATION OF HEAVY-SECTION STEEL TECHNOLOGY PROGRAM PLATES 01,02,AND 03.

STERNER,P.
NUREG-1095: EVALUATION OF RESPONSES TO IE BULLETIN 82-02.Degradation Of Threaded Fasteners In Reactor Coolant Pressure Boundary Of Pressurized Water-Reactor Plants.

STETZENBACH,K.
NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984.

STEVENS,D.L.
NUREG/CR-3659: A MATHEMATICAL MODEL FOR ASSESSING THE UNCERTAINTIES OF INSTRUMENTATION MEASUREMENTS FOR POWER AND FLOW OF PWR REACTORS.

STILLWELL,W.G.
NUREG/CR-3688 V01: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 1:Main Report.
NUREG/CR-3688 V02: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 2:Appendices.

STITT,B.D.
NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR.
NUREG/CR-4326 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Main Report.
NUREG/CR-4326 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Appendices.

STOETZEL,G.A.
NUREG/CR-4160: HISTORICAL SUMMARY OF OCCUPATIONAL RADIATION EXPOSURE EXPERIENCE IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.

STROMBERG,H.M.
NUREG/CR-4262 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR.Main Report.
NUREG/CR-4262 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR. Appendices.
NUREG/CR-4326 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Main Report.
NUREG/CR-4326 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Appendices.

STROSNIDER,J.
NUREG-1155 V03: RESEARCH PROGRAM PLAN.Piping.

SUBRAMANIAN,C.
NUREG/CR-4119: INTEGRITY OF CONTAINMENT PENETRATIONS UNDER SEVERE ACCIDENT CONDITIONS FY84 ANNUAL REPORT.

SUBUDHI,M.
NUREG/CR-1677 V02: PIPING BENCHMARK PROBLEMS,VOLUME II DYNAMIC ANALYSIS INDEPENDENT SUPPORT MOTION RESPONSE SPECTRUM METHOD.
NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.
NUREG/CR-4291: CONCLUSION AND SUMMARY REPORT ON PHYSICAL BENCHMARKING OF PIPING SYSTEMS.
NUREG/CR-4440: A REVIEW OF EMERGENCY DIESEL GENERATOR PERFORMANCE AT NUCLEAR POWER PLANTS.

SUES,R.H.
NUREG/CR-4290 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF BABCOCK AND WILCOX PWR PLANTS.Volume 2:Guillotine Break Indirectly Induced By Earthquakes.

SULLIVAN,T.
NUREG/CR-2482 V08: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period October 1984 - March 1985.
NUREG/CR-2482 V09: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period April 1985-September 1985.

SULLIVAN,W.H.
NUREG/CR-3301: CATALOG OF PRA DOMINANT ACCIDENT SEQUENCE INFORMATION.

SUNDARUM,R.K.
NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT.

SUTCLIFFE,C.R.
NUREG/CR-4045: LITERATURE REVIEW ON AEROSOL-SAMPLING DEVICES FOR RESPIRATORY FIELD STUDIES.

SUTHERLAND,W.A.
NUREG/CR-4128: BWR FULL INTEGRAL SIMULATION TEST (FIST) PHASE II TEST RESULTS AND TRAC-BWR MODEL QUALIFICATION.

SUTTON,G.E.
NUREG/CR-4283: STUDY OF THE EFFECTS OF ELASTIC UNLOADINGS ON THE JI-R CURVES FROM COMPACT SPECIMENS.

SWAIN,R.L.
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.

SWANSON,J.L.
NUREG/CR-4435: ORGANIC COMPLEXANT-ENHANCED MOBILITY OF TOXIC ELEMENTS IN LOW-LEVEL WASTES.Annual Report,July 1984 - June 1985.

SWYLER,K.J.
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.

SYKES,R.I.
NUREG/CR-4157: A SCIENTIFIC CRITIQUE OF AVAILABLE MODELS FOR REAL-TIME SIMULATIONS OF DISPERSION.
NUREG/CR-4159: COMPARISON OF THE 1981 INEL DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS.

SZAWLEWICZ,S.A.
NUREG/CP-0058 V01: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CP-0058 V02: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CP-0058 V04: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CP-0058 V05: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CP-0058 V06: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.

TABATABAI,A.S.
NUREG/CR-4385: EFFECTS OF CONTROL SYSTEM FAILURES IN TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A WESTINGHOUSE PRESSURIZED WATER REACTOR.
NUREG/CR-4386: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A BABCOCK AND WILCOX PRESSURIZED WATER REACTOR.
NUREG/CR-4387: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS AND CORE-MELT FREQUENCIES AT A GENERAL ELECTRIC PRESSURIZED WATER REACTOR.

TALEYARKHAN,R.
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.

TANG,P.K.
NUREG/CR-4260: TORAC USER'S MANUAL.A Computer Code For Analyzing Tornado-Induced Flow And Material Transport In Nuclear Facilities.

TARBELL,W.W.
NUREG/CR-4383: HIGH PRESSURE INJECTION OF MELT FROM A REACTOR PRESSURE VESSEL - THE DISCHARGE PHASE.

TASAO,L.
NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt.

TAWIL,J.J.
NUREG/CR-3413: OFF-SITE CONSEQUENCES OF RADIOLOGICAL ACCIDENTS:METHODS, COSTS AND SCHEDULES FOR DECONTAMINATION.

TAYLOR,D.D.
NUREG/CR-3633 V01 S1: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.

TAYLOR,J.
NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.

TAYLOR,J.H.
NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.
NUREG/CR-4152: AN INDEPENDENT SAFETY ORGANIZATION.

TAYLOR,J.T.
NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

TEAGUE,A.G.
NUREG/CR-4288: FOCAL MECHANISM ANALYSES FOR VIRGINIA AND EASTERN TENNESSEE EARTHQUAKES (1978-1984).

TEICHMANN,T.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-2815 V02 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 8-12.
NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.
NUREG/CR-3485: PRA REVIEW MANUAL.

THATCHER,D.
NUREG-0933 S02: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

THEOFANOUS,T.G.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

THIE,J.A.
NUREG/CR-4256: MEASUREMENT OF RESPONSE TIME AND DETECTION OF DEGRADATION IN PRESSURE SENSOR/SENSING LINE SYSTEMS.

THOMA,J.O.
NUREG-1110: COMPARISON OF LICENSING ACTIVITIES FOR OPERATING PLANTS DESIGNED BY BABCOCK & WILCOX.

THOMAS,J.M.
NUREG/CR-4268: RATIO METHODS FOR COST-EFFECTIVE FIELD SAMPLING OF COMMERCIAL RADIOACTIVE LOW-LEVEL WASTES.

THOMAS,V.W.
NUREG/CR-4057: RADIOLOGICAL ASSESSMENT OF THE TOWN OF EDGEMONT.
NUREG/CR-4100: EVALUATION OF INSTRUMENTAL METHODS FOR THE MEASUREMENT OF YELLOWCAKE EMISSIONS.

THOMPSON,S.L.
NUREG/CR-3802: RELAP5 ASSESSMENT:QUANTITATIVE KEY PARAMETERS AND RUN TIME STATISTICS.
NUREG/CR-3820 V03: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM.Quarterly Report,July-September 1984.

THOMPSON,V.N.
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.

THOMS,K.R.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.

THORNGREN,L.G.
NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.

THURBER,J.A.
NUREG/CR-4125 V01: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 1:Guidelines For Utility Organization And Administration Plan.

NUREG/CR-4125 V02: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 2:Workbook For Assessment Of Organization And Management.
NUREG/CR-4281: AN EMPIRICAL ANALYSIS OF SELECTED NUCLEAR POWER PLANT MAINTENANCE FACTORS AND PLANT SAFETY.

TIEN,P.L.
NUREG/CR-4110: REPOSITORY SITE DATA REPORT FOR UNSATURATED TUFF,YUCCA MOUNTAIN,NEVADA.

TILL,J.E.
NUREG/CR-3738: ENVIRONMENTAL EFFECTS OF THE URANIUM FUEL CYCLE.A Review Of Data For Technetium.

TKACHYK,J.W.
NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

TOALSTON,A.L.
NUREG-0970: PROCEDURES FOR MEETING NRC ANTITRUST RESPONSIBILITIES.

TOBIAS,M.L.
NUREG/CR-3830 V02: AEROSOL RELEASE AND TRANSPORT PROGRAM.Semiannual Progress Report For April 1984-September 1984.
NUREG/CR-4255 V01: AEROSAL RELEASE AND TRANSPORT PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.

TORRONEN,K.
NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS.

TOSTE,A.P.
NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER.(Final Report)

TOTH,L.M.
NUREG/CR-3514 V02: THE CHEMICAL BEHAVIOR OF IODINE IN AQUEOUS SOLUTIONS UP TO 150 C.II.Radiation-Redox Conditions.

TRAVIS,J.R.
NUREG/CR-3930: OBSERVED BEHAVIOR OF CESIUM,IODINE,AND TELLURIUM IN THE ORNL FISSION PRODUCT RELEASE PROGRAM.
NUREG/CR-4020: HMS:A COMPUTER PROGRAM FOR TRANSIENT,THREE-DIMENSIONAL MIXING GASES.
NUREG/CR-4037: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-5.
NUREG/CR-4043: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-6.

TREBILCOCK,W.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.

TRIER,R.M.
NUREG/CR-4094: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN SULFATE LAKE ENVIRONMENTS.
NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.

TRIGGS,T.J.
NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS.

TSANG,F.Y.
NUREG/CR-3237: CONTROL OF EXPLOSIVE MIXTURES IN PWR WASTE GAS SYSTEMS.

TURGEON,K.S.
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.

TURNER,J.H.
NUREG/CR-3922 V01: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Main Report And Appendices A And B.
NUREG/CR-3922 V02: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Appendices C And D.

TURNEY,J.R.
NUREG/CR-4258: AN APPROACH TO TEAM SKILLS TRAINING OF NUCLEAR POWER PLANT CONTROL ROOM CREWS.

UNIONE,A.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-2815 V02 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 8-12.

UPDEGRAFF,C.D.
NUREG/CR-4110: REPOSITORY SITE DATA REPORT FOR UNSATURATED TUFF,YUCCA MOUNTAIN,NEVADA.

URIBE,R.
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.

VADEN,J.
NUREG/CR-4352: SUGGESTED STATE REQUIREMENTS AND CRITERIA FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE REGULATORY PROGRAM.

VAGINS,M.
NUREG-1155 V01: RESEARCH PROGRAM PLAN.Reactor Vessels.
NUREG-1155 V03: RESEARCH PROGRAM PLAN.Piping.

VALENTINE,B.M.
NUREG/CR-4178 DRFT: AN EVALUATION OF SELECTED LICENSEE EVENT REPORTS PREPARED PURSUANT TO 10 CFR 50.73.Draft Report.

VALENZUELA,J.A.
NUREG/CR-3426 V01: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY. Facility And Test Design Report.
NUREG/CR-3426 V02: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY.Data Report.

VANDEGRIFT,G.F.
NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT.

VANDER MOLEN,H.
NUREG-0933 S02: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

VANDERMOLEN,H.
NUREG-0933 S03: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

VARMA,A.K.
NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING.

VASLOW,F.
NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA.

VASSILAROS,M.G.
NUREG/CR-4283: STUDY OF THE EFFECTS OF ELASTIC UNLOADINGS ON THE JI-R CURVES FROM COMPACT SPECIMENS.

VEAKIS,E.
NUREG/CR-2482 V08: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period October 1984 - March 1985.
NUREG/CR-3091 V04: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1983 - March 1984.
NUREG/CR-3091 V05: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period April 1984 - September 1984.

VESELY,W.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.

VESELY,W.E.
NUREG/CR-4377: EVALUATIONS AND UTILIZATIONS OF RISK IMPORTANCES.

VICKROY,S.C.
NUREG/CR-4298: DESIGN AND INSTALLATION OF COMPUTER SYSTEMS TO MEET THE REQUIREMENTS OF 10 CFR 73.55.

VIERZBA,E.
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VINCENT,B.G.
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VORA,J.P.
NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.

VOSKA,K.J.
NUREG/CR-3519: HUMAN ERROR PROBABILITY ESTIMATION USING LICENSEE EVENT REPORTS.

WAHI,K.K.
NUREG/CR-4110: REPOSITORY SITE DATA REPORT FOR UNSATURATED TUFF,YUCCA MOUNTAIN,NEVADA.

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WALLO,A.
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WALSH,M.E.
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WALTERS,W.H.
NUREG/CR-3752: EFFECTS OF HYDROLOGIC VARIABLES ON ROCK RIPRAP DESIGN FOR URANIUM TAILINGS IMPOUNDMENTS.
NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION.

WANG,Y.K.
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NUREG/CR-4291: CONCLUSION AND SUMMARY REPORT ON PHYSICAL BENCHMARKING OF PIPING SYSTEMS.

WARD,R.C.
NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE.

WARRINER,J.B.
NUREG/CR-3774 V02: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2A:Technical Requirements For Belowground Vault Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V03: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2B:Technical Requirements For Aboveground Vault Disposal Of Low Level Radioactive Waste.

WASCOM,R.L.
NUREG/CR-3516: A SURVEY OF THE USES OF RADIOACTIVE MATERIALS IN LOUISIANA'S OFFSHORE WATERS.

WASTAG,M.
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WATERMAN,M.E.
NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR.
NUREG/CR-4195: OVERVIEW OF TRAC-PD2 ASSESSMENT CALCULATIONS.
NUREG/CR-4326 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Main Report.
NUREG/CR-4326 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Appendices.

WATKINS,J.C.
NUREG/CR-4141: CONTAINMENT PURGE AND VENT VALVE TEST PROGRAM FINAL REPORT.

WEAKLEY,S.A.
NUREG/CR-2800 S03: GUIDELINES FOR NUCLEAR POWER PLANT SAFETY ISSUE PRIORITIZATION INFORMATION DEVELOPMENT.

WEBER,C.F.
NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1-September 30,1984.
NUREG/CR-3885 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, October 1-December 31,1984.
NUREG/CR-4402 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31,1985.

WEBSTER,C.S.
NUREG/CR-3930: OBSERVED BEHAVIOR OF CESIUM,IODINE,AND TELLURIUM IN THE ORNL FISSION PRODUCT RELEASE PROGRAM.
NUREG/CR-4037: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-5.
NUREG/CR-4043: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-6.

WEILANDICS,C.
NUREG/CR-3469 V02: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS. Annotated Bibliography Of Selected Readings In Radiation Protection And ALARA.

WEINSTOCK,E.V.
NUREG/CR-4152: AN INDEPENDENT SAFETY ORGANIZATION.

WEISS,A.J.
NUREG/CP-0059 V01: PROCEEDINGS OF THE MITI-NRC SEISMIC INFORMATION EXCHANGE MEETING,VOLUME I.
NUREG/CP-0071: TRANSACTIONS OF THE THIRTEENTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984.
NUREG/CR-2331 V04 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July 1 -September 30,1984.
NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.

WELC,J.
NUREG/CR-3145 V03: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION - ANNUAL REPORT.(October 1982 - September 1983, Volume 3).

WELLS,J.E.
NUREG/CR-4331: SIMPLIFIED SEISMIC PROBABILISTIC RISK ASSESSMENT.Procedures And Limitations.

WENSEL,R.G.
NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT.

WERES,O.
NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt.

WESCOTT,R.G.
NUREG-1165: ESRP 7.1.1 "ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RELEASES OF RADIOACTIVE MATERIALS TO GROUNDWATER."

WESLEY,D.A.
NUREG/CR-3558: HANDBOOK OF NUCLEAR POWER PLANT SEISMIC FRAGILITIES. Seismic Safety Margins Research Program.

WESTER,M.J.
NUREG/CR-3913: HECTR VERSION 1.0 USER'S MANUAL.

WETTMILLER,R.J.
NUREG/CR-4317 V01: CANADIAN SEISMIC AGREEMENT.Technical Report Covering 1979-1985.

WHATLEY,S.K.
NUREG/CP-0062: PROCEEDINGS OF THE CONFERENCE ON THE APPLICATION OF GEOCHEMICAL MODELS TO HIGH-LEVEL NUCLEAR WASTE REPOSITORY ASSESSMENT.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984.
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.

WHEELER,W.A.
NUREG/CR-3817: DEVELOPMENT,USE AND CONTROL OF MAINTENANCE PROCEDURES IN NUCLEAR POWER PLANTS.Problems And Recommendations.

WHITE,A.S.
NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.

WHITE,J.D.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

WHITE,J.R.
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.

WHITMAN,G.D.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.

WIDRIG,R.D.
NUREG/CR-4125 V01: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 1:Guidelines For Utility Organization And Administration Plan.
NUREG/CR-4125 V02: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 2:Workbook For Assessment Of Organization And Management.

WILKOWSKI,G.M.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

WILLIAMS,D.C.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.

WILLIAMS,L.
NUREG/CR-4031 V02: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutronics Calculations."

WILLIAMS,W.R.
NUREG/CR-4360 V01: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.
NUREG/CR-4360 V02: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.

WILLIFORD,R.E.
NUREG/CR-4168: GT2F:A COMPUTER CODE FOR ESTIMATING LIGHT WATER REACTOR FUEL ROD FAILURES.

WILLIS,C.A.
NUREG-0017 R01: CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM PRESSURIZED WATER REACTORS (PWR-GALE CODE).

WILSON,J.H.
NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1-September 30,1984.
NUREG/CR-3885 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, October 1-December 31,1984.
NUREG/CR-4402 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31,1985.

WILSON,M.D.
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.

WINEGARDNER,W.
NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.

WITHEE,C.J.
NUREG-1065 R01: ACCEPTANCE CRITERIA FOR THE LOW ENRICHED URANIUM REFORM AMENDMENTS.

WOLF,J.J.
NUREG/CR-3626 V02: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: DESCRIPTION OF MODEL CONTENT,STRUCTURE,AND SENSITIVITY TESTING.
NUREG/CR-3634: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: USER'S MANUAL.
NUREG/CR-4104: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL. Field Evaluation/Validation.

WOLLENBERG,H.A.
NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt.

WOMELSDUFF,J.E.
NUREG/CR-3361: THE EFFECT OF WATER CHEMISTRY ON THE RATES OF HYDROGEN GENERATION FROM GALVANIZED STEEL CORROSION AT POST-LOCA CONDITIONS.
NUREG/CR-3803: THE EFFECTS OF POST-LOCA CONDITIONS ON A PROTECTIVE COATING (PAINT) FOR THE NUCLEAR POWER INDUSTRY.

WONG,C.N.
NUREG/CR-4044: TRAC-PF1 LOCA CALCULATIONS USING FINE-NODE AND COARSE-NODE INPUT MODELS.

WONG,K.S.
NUREG/CR-4190: CALIFORNIA OFFSHORE SURVEY OF LICENSEES USING RADIOACTIVE MATERIAL.

WOO,H.H.
NUREG/CR-3019: RECOMMENDED WELDED CRITERIA FOR USE IN THE FABRICATION OF SHIPPING CONTAINERS FOR RADIOACTIVE MATERIALS.

WOOD,R.S.
NUREG-1131: FINANCIAL ANALYSIS OF POTENTIAL RETROSPECTIVE PREMIUM ASSESSMENTS UNDER THE PRICE-ANDERSON SYSTEM.

WORRELL,R.B.
NUREG/CR-4213: SETS REFERENCE MANUAL.

WREATHALL,J.
NUREG/CR-4177 V01: MANAGEMENT OF SEVERE ACCIDENTS.Perspectives On Managing Severe Accidents In Commercial Nuclear Power Plants.
NUREG/CR-4177 V02: MANAGEMENT OF SEVERE ACCIDENTS.Extending Plant Operating Procedures Into The Severe Accident Regime.

WRENN,M.E.
NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.

WRIGHT,A.L.
NUREG/CR-4346: AEROSOL RELEASE EXPERIMENTS IN THE FUEL AEROSOL SIMULANT TEST FACILITY:UNDERSODIUM EXPERIMENTS.

WRIGHT,K.W.
NUREG/CR-4101: ASSAY OF LONG-LIVED RADIONUCLIDES IN LOW-LEVEL WASTES FROM POWER REACTORS.

WRIGHT,M.A.
NUREG/CR-4117: FAULTING AND JOINTING IN AND NEAR SURFACE MINES OF SOUTHWESTERN INDIANA.

WU,Y.S.
NUREG/CR-4365: DESIGN AND DEVELOPMENT OF A SPECIAL PURPOSE SAFT SYSTEM FOR NONDESTRUCTIVE EVALUATION OF NUCLEAR REACTOR VESSELS AND PIPING COMPONENTS.

WULFF,W.
NUREG/CR-3943: THE BWR PLAN ANALYZER.

WYANT,F.J.
NUREG/CR-4147: THE EFFECT OF ENVIRONMENTAL STRESS ON SYLGARD 70 SILICONE ELASTOMER.

YAZDANDOOST,A.
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

YEE,C.S.
NUREG/CR-3537: EXPEDIENT METHODS OF RESPIRATORY PROTECTION:III. SUBMICRON PARTICLE TESTS AND SUMMARY OF QUALITY FACTORS.

YOUNG,J.
NUREG/CR-4350 V01: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 1 - PRA Fundamentals.
NUREG/CR-4350 V04: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 4 - System Reliability And Analysis Techniques,Sessions B/C - Event Trees/Fault Trees.

YOUNG,J.A.
NUREG/CR-4057: RADIOLOGICAL ASSESSMENT OF THE TOWN OF EDGEMONT.
NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.

YOUNG,T.E.
NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION.
NUREG/CR-4397: IN-PLANT SOURCE TERM MEASUREMENTS AT PRAIRIE ISLAND NUCLEAR GENERATING STATION.

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NUREG/CR-2815 V02 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 8-12.

YOUNGBLOOD,R.
NUREG/CR-4228: REVIEW OF THE VOGTLE UNITS 1 AND 2 AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS.

YOUNGBLOOD,R.W.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-3485: PRA REVIEW MANUAL.

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NUREG-1046: DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN UNSATURATED ZONE:TECHNICAL CONSIDERATIONS AND RESPONSE TO COMMENTS.

A302-B Steel
NUREG/CR-4437: EXPLORATORY STUDIES OF ELEMENT INTERACTIONS AND COMPOSITION DEPENDENCIES IN RADIATION SENSITIVITY DEVELOPMENT.

A533-B Steel
NUREG/CR-4437: EXPLORATORY STUDIES OF ELEMENT INTERACTIONS AND COMPOSITION DEPENDENCIES IN RADIATION SENSITIVITY DEVELOPMENT.

AC Power
NUREG-1032 DRFT FC: EVALUATION OF STATION BLACKOUT ACCIDENTS AT NUCLEAR POWER PLANTS.Technical Findings Related To Unresolved Safety Issue A-44.Draft Report For Comment.

ACRS
NUREG-1125 V01: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 1,Part 1:ACRS Reports On Project Reviews (A-F).
NUREG-1125 V02: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 2,Part 1:ACRS Reports On Project Reviews (G-P).
NUREG-1125 V03: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 3,Part 1:ACRS Reports On Project Reviews (Q-Z).
NUREG-1125 V04: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 4,Part 2:ACRS Reports On Generic Subjects (Accident Analysis - Generic Items).
NUREG-1125 V05: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 5,Part 2:ACRS Reports On Generic Subjects (HTGR - Regulatory Guides).
NUREG-1125 V06: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 6,Part 2:ACRS Reports On Generic Subjects (RPA - Appendix C).

ALARA
NUREG/CP-0066: PROCEEDINGS OF AN INTERNATIONAL WORKSHOP ON HISTORIC DOSE EXPERIENCE AND DOSE REDUCTION (ALARA) AT NUCLEAR POWER PLANTS,MAY 29-JUNE 1,1984.
NUREG/CR-3293 V01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Main Report.
NUREG/CR-3293 V02: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Appendices.
NUREG/CR-3469 V02: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS. Annotated Bibliography Of Selected Readings In Radiation Protection And ALARA.
NUREG/CR-4090: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Annual Summary Report - Fiscal Year 1984.
NUREG/CR-4254: OCCUPATIONAL DOSE REDUCTION AND ALARA AT NUCLEAR POWER PLANTS.Study On High-Dose Jobs,Radwaste Handling,And ALARA Incentives.
NUREG/CR-4373: COMPENDIUM OF COST-EFFECTIVENESS EVALUATIONS OF MODIFICATIONS FOR DOSE REDUCTION AT NUCLEAR POWER PLANTS.

ASME Code
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NUREG/CR-3647: DESIGN AND FABRICATION OF A 1/8-SCALE STEEL CONTAINMENT MODEL.

ATWS
NUREG/CR-3633 V01 S1: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.
NUREG/CR-3704: BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE.
NUREG/CR-4046: DETERMINING CRITICAL FLOW VALVE CHARACTERISTICS USING EXTRAPOLATION TECHNIQUES.

Abnormal Occurrence
NUREG-0090 V07 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1984.
NUREG-0090 V07 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1984.
NUREG-0090 V08 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.Janurary-March 1985.
NUREG-0090 V08 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1985.

Aboveground Vault
NUREG/CR-3774 V03: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2B:Technical Requirements For Aboveground Vault Disposal Of Low Level Radioactive Waste.

Abstract
NUREG-0304 V09 N04: REGULATORY AND TECHNICAL REPORTS.Annual Compilation For 1984.
NUREG-0304 V10 N02: REGULATORY AND TECHNICAL REPORTS.Compilation For Second Quarter 1985,April-June.

Access Control
NUREG/CR-4298: DESIGN AND INSTALLATION OF COMPUTER SYSTEMS TO MEET THE REQUIREMENTS OF 10 CFR 73.55.
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP).A Methodology For Estimating Risk Impacts Of Safeguards Measures.

Accident
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).
NUREG-0981 R01: NRC/FEMA OPERATIONAL RESPONSE PROCEDURES FOR RESPONSE TO A COMMERCIAL NUCLEAR REACTOR ACCIDENT.
NUREG-1037 DRFT FC: CONTAINMENT PERFORMANCE WORKING GROUP REPORT.Draft Report For Comment.
NUREG-1116: A REVIEW OF THE CURRENT UNDERSTANDING OF THE POTENTIAL FOR CONTAINMENT FAILURE FROM IN-VESSEL STEAM EXPLOSIONS.
NUREG/CR-3293 V01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Main Report.
NUREG/CR-3293 V02: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Appendices.
NUREG/CR-3301: CATALOG OF PRA DOMINANT ACCIDENT SEQUENCE INFORMATION.
NUREG/CR-3317: TECHNICAL BASES AND USER'S MANUAL FOR THE PROTOTYPE OF SPARC - A SUPPRESSION POOL AEROSOL REMOVAL CODE.
NUREG/CR-3413: OFF-SITE CONSEQUENCES OF RADIOLOGICAL ACCIDENTS.METHODS, COSTS AND SCHEDULES FOR DECONTAMINATION.
NUREG/CR-3485: PRA REVIEW MANUAL.
NUREG/CR-3611: RADIOACTIVE MATERIAL (RAM) ACCIDENT/INCIDENT DATA ANALYSIS PROGRAM.
NUREG/CR-3816 V02: REACTOR SAFETY RESEARCH.Quarterly Report,April-June 1984.

NUREG/CR-3816 V03: REACTOR SAFETY RESEARCH.Quarterly Report,July-September 1984.
NUREG/CR-3855: CHARACTERIZATION OF NUCLEAR REACTOR CONTAINMENT PENETRATION - FINAL REPORT.
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.
NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT.
NUREG/CR-3943: THE BWR PLAN ANALYZER.
NUREG/CR-3954: HECTR ANALYSIS OF EQUIPMENT TEMPERATURE RESPONSES TO SELECTED HYDROGEN BURNS IN AN ICE CONDENSER CONTAINMENT.
NUREG/CR-3990: CHARCOAL PERFORMANCE UNDER ACCIDENT CONDITIONS IN LIGHT-WATER REACTORS.
NUREG/CR-4035: A HIGHWAY ACCIDENT INVOLVING RADIOPHARMACEUTICALS NEAR BROOKHAVEN,MISSISSIPPI ON DECEMBER 3,1983.
NUREG/CR-4055: THE D10 EXPERIMENT:COOLABILITY OF UO2 DEBRIS IN SODIUM WITH DOWNWARD HEAT REMOVAL.
NUREG/CR-4067: SUMMARY OF BARRIER DEGRADATION EVENTS AND SMALL ACCIDENTS IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.
NUREG/CR-4080: DETERMINATION OF THE AVAILABILITY OF CORE EXIT THERMOCOUPLES DURING SEVERE ACCIDENT SITUATIONS.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.
NUREG/CR-4091: THE EFFECT OF ALTERNATIVE AGING AND ACCIDENT SIMULATIONS ON POLYMER PROPERTIES.
NUREG/CR-4119: INTEGRITY OF CONTAINMENT PENETRATIONS UNDER SEVERE ACCIDENT CONDITIONS FY84 ANNUAL REPORT.
NUREG/CR-4146: SIMULATION OF AN EPRI-NEVADA TEST SITE (NTS) HYDROGEN BURN TEST AT THE CENTRAL RECEIVER TEST FACILITY.
NUREG/CR-4169: AN APPROACH TO TREATING RADIONUCLIDE DECAY HEATING FOR USE IN THE MELCOR CODE SYSTEM.
NUREG/CR-4172: A USER'S GUIDE FOR MERGE.
NUREG/CR-4185: AN ASSESSMENT OF DOSIMETRY DATA FOR ACCIDENTAL RADIONUCLIDE RELEASES FROM NUCLEAR REACTORS.
NUREG/CR-4197: SAFETY GOAL SENSITIVITY STUDIES.
NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS.
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.
NUREG/CR-4214: HEALTH EFFECTS MODEL FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Part I:Introduction,Integration & Summary.Part II:Scientific Basis For Health Effects Models.
NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUNDWATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions.
NUREG/CR-4251 V02: MITIGATIVE TECHNIQUES FOR GROUNDWATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 2:Case Study Analysis Of Hydrologic Characterization And Mitigative Schemes.
NUREG/CR-4262 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR.Main Report.
NUREG/CR-4262 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR. Appendices.
NUREG/CR-4278: TRAC-PF1/MOD1 DEVELOPMENT ASSESSMENT.
NUREG/CR-4294: LEAK RATE ANALYSIS OF THE WESTINGHOUSE REACTOR COOLANT PUMP.
NUREG/CR-4304: PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO THE PWR THERMAL-SHOCK ISSUE.Experiment TSE-7.
NUREG/CR-4321: FULL SCALE MEASUREMENTS OF SMOKE TRANSPORT AND DEPOSITION IN VENTILATION SYSTEM DUCTWORK.
NUREG/CR-4325: A PARAMETRIC STUDY OF PWR PRESSURE VESSEL INTEGRITY DURING OVERCOOLING ACCIDENTS,CONSIDERING BOTH 2-D AND 3-D FLAWS.
NUREG/CR-4326 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Main Report.
NUREG/CR-4326 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Appendices.
NUREG/CR-4342: UNCERTAINTY AND SENSITIVITY ANALYSIS OF A MODEL FOR MULTICOMPONENT AEROSOL DYNAMICS.
NUREG/CR-4360 V01: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.
NUREG/CR-4360 V02: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.
NUREG/CR-4383: HIGH PRESSURE INJECTION OF MELT FROM A REACTOR PRESSURE VESSEL - THE DISCHARGE PHASE.
NUREG/CR-4385: EFFECTS OF CONTROL SYSTEM FAILURES IN TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A WESTINGHOUSE PRESSURIZED WATER REACTOR.
NUREG/CR-4386: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A BABCOCK AND WILCOX PRESSURIZED WATER REACTOR.
NUREG/CR-4387: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS AND CORE-MELT FREQUENCIES AT A GENERAL ELECTRIC PRESSURIZED WATER REACTOR.
NUREG/CR-4388: AEROSOL BEHAVIOR MODELING (TASK 3) - SUPPORT SERVICES FOR RESEARCH AND EVALUATION OF SEVERE ACCIDENT PHENOMENA AND MITIGATION.
NUREG/CR-4399: POSSIBLE OPTIONS FOR REDUCING OCCUPATIONAL DOSE FROM THE TMI-2 BASEMENT.

Accident Analysis
NUREG-1125 V04: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 4,Part 2:ACRS Reports On Generic Subjects (Accident Analysis - Generic Items).
NUREG/CR-3887: HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS.
NUREG/CR-4214: HEALTH EFFECTS MODEL FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Part I:Introduction,Integration & Summary.Part II:Scientific Basis For Health Effects Models.

Accident Management
NUREG/CR-4177 V01: MANAGEMENT OF SEVERE ACCIDENTS.Perspectives On Managing Severe Accidents In Commercial Nuclear Power Plants.
NUREG/CR-4177 V02: MANAGEMENT OF SEVERE ACCIDENTS.Extending Plant Operating Procedures Into The Severe Accident Regime.

Accident Mitigation
NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences.

Accident Sequence
NUREG/CR-4140: DOMINANT ACCIDENT SEQUENCES IN OCONEE-1 PRESSURIZED WATER REACTOR.

Acid Digestion
NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.

Acid Waste
NUREG/CR-3906: URANIUM MILL TAILINGS NEUTRALIZATION:CONTAMINANT COMPLEXATION AND TAILINGS LEACHING STUDY.

Acoustic Emission
NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984.
NUREG-1155 V04: RESEARCH PROGRAM PLAN.Non-Destructive Examination.
NUREG/CR-3825 V03-4: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Quarterly Report, April 1984 - September 1984.Volumes 3 and 4.
NUREG/CR-3915: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL.
NUREG/CR-4300 V01: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Progress Report, October-March 1985.

Acoustic Leak Detection
NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984.

Acronyms And Initialisms
NUREG-0544 R02: A HANDBOOK OF ACRONYMS AND INITIALISMS.

Actinide Element
NUREG/CR-4094: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN SULFATE LAKE ENVIRONMENTS.

Administration
- NUREG/CR-4125 V01: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 1:Guidelines For Utility Organization And Administration Plan.
- NUREG/CR-4125 V02: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 2:Workbook For Assessment Of Organization And Management.

Adsorption
- NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.
- NUREG/CR-4094: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN SULFATE LAKE ENVIRONMENTS.
- NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.

Adverse Systems Interaction
- NUREG/CR-3922 V01: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Main Report And Appendices A And B.
- NUREG/CR-3922 V02: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Appendices C And D.

Aerosol
- NUREG/CR-3537: EXPEDIENT METHODS OF RESPIRATORY PROTECTION:III. SUBMICRON PARTICLE TESTS AND SUMMARY OF QUALITY FACTORS.
- NUREG/CR-3830 V02: AEROSOL RELEASE AND TRANSPORT PROGRAM.Semiannual Progress Report For April 1984-September 1984.
- NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984.
- NUREG/CR-4045: LITERATURE REVIEW ON AEROSOL-SAMPLING DEVICES FOR RESPIRATORY FIELD STUDIES.
- NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0:A Computer Code for Severe Reactor Accident Containment Analysis.
- NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET-GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE,AND SODIUM CHLORIDE.
- NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.
- NUREG/CR-4205: TRAP-MELT2 USER'S MANUAL.
- NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS.
- NUREG/CR-4255 V01: AEROSAL RELEASE AND TRANSPORT PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.
- NUREG/CR-4264: INVESTIGATION ON HIGH-EFFICIENCY PARTICULATE AIR FILTER PLUGGING BY COMBUSTION AEROSOLS.
- NUREG/CR-4342: UNCERTAINTY AND SENSITIVITY ANALYSIS OF A MODEL FOR MULTICOMPONENT AEROSOL DYNAMICS.
- NUREG/CR-4346: AEROSOL RELEASE EXPERIMENTS IN THE FUEL AEROSOL SIMULANT TEST FACILITY:UNDERSODIUM EXPERIMENTS.
- NUREG/CR-4388: AEROSOL BEHAVIOR MODELING (TASK 3) - SUPPORT SERVICES FOR RESEARCH AND EVALUATION OF SEVERE ACCIDENT PHENOMENA AND MITIGATION.

Agenda
- NUREG-0936 V03 N04: NRC REGULATORY AGENDA.Quarterly Report,October-December 1984.

Aging
- NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.
- NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984.
- NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.
- NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT.
- NUREG/CR-3819: SURVEY OF AGED POWER PLANT FACILITIES.
- NUREG/CR-3998 V03: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report.October-December 1984.
- NUREG/CR-4008: GENERAL EXTRAPOLATION MODEL FOR AN IMPORTANT CHEMICAL DOSE-RATE EFFECT.
- NUREG/CR-4091: THE EFFECT OF ALTERNATIVE AGING AND ACCIDENT SIMULATIONS ON POLYMER PROPERTIES.
- NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.
- NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.
- NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.

Air Cleaning System
- NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.
- NUREG/CR-4225: SUMMARY OF EFFICIENCY TESTING OF STANDARD AND HIGH-CAPACITY HIGH-EFFICIENCY PARTICULATE AIR FILTERS SUBJECTED TO SIMULATED TORNADO DEPRESSURIZATION AND EXPLOSIVE SHOCK WAVES.

Air Sampler
- NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

Airborne Release
- NUREG/CR-4088: METHODS FOR ESTIMATING RADIOACTIVE AND TOXIC AIRBORNE SOURCE TERMS FOR URANIUM MILLING OPERATIONS.

Alarm Resolution
- NUREG/CR-4108: DEVELOPMENT OF MC&A ALARM RESOLUTION PROCEDURES.

Alloy Steels
- NUREG/CR-4437: EXPLORATORY STUDIES OF ELEMENT INTERACTIONS AND COMPOSITION DEPENDENCIES IN RADIATION SENSITIVITY DEVELOPMENT.

Alternating Current
- NUREG/CR-4294: LEAK RATE ANALYSIS OF THE WESTINGHOUSE REACTOR COOLANT PUMP.

Alternative Disposal Method
- NUREG/CR-3774 V02: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2A:Technical Requirements For Belowground Vault Disposal Of Low Level Radioactive Waste.
- NUREG/CR-3774 V03: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2B:Technical Requirements For Aboveground Vault Disposal Of Low Level Radioactive Waste.
- NUREG/CR-3774 V04: ALTERNATIVE METHOD FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2C:Technical Requirements For Earth Mounded Concrete Bunker Disposal Of Low Level Radioactive Waste.
- NUREG/CR-3774 V05: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2E:Technical Requirements For Shaft Disposal Of Low Level Radioactive Waste.

Ambient Radiation Level
- NUREG-0837 V04 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, July-September 1984.
- NUREG-0837 V04 N04: NRC TLD DIRECT RADIATION MONITORING REPORT.Progress Report, October-December 1984.

Americium-241
- NUREG-1153: INSPECTION REPORT OF UNAUTHORIZED POSSESSION AND USE OF UNSEALED AMERICIUM-241 AND SUBSEQUENT CONFISCATION. J.C. Haynes Company,Newark,Ohio.

Analysis
- NUREG-1140 DRFT FC: A REGULATORY ANALYSIS ON EMERGENCY PREPAREDNESS FOR FUEL CYCLE AND OTHER RADIOACTIVE MATERIAL LICENSEES.Draft Report For Comment.
- NUREG/CR-4360 V01: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.
- NUREG/CR-4360 V02: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.

Annual Report
- NUREG-1145 V01: U.S. NUCLEAR REGULATORY COMMISSION 1984 ANNUAL REPORT.

Annunciator System
NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS.

Anodic Dissolution
NUREG/CR-4422: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN PWR ENVIRONMENTS.

Anonymity
NUREG/CR-4133: NUCLEAR POWER SAFETY REPORTING SYSTEM IMPLEMENTATION AND OPERATIONAL SPECIFICATIONS.

Anticipated Transient Without Scram
NUREG/CR-3633 V01 S1: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.
NUREG/CR-4046: DETERMINING CRITICAL FLOW VALVE CHARACTERISTICS USING EXTRAPOLATION TECHNIQUES.

Antitrust
NUREG-0970: PROCEDURES FOR MEETING NRC ANTITRUST RESPONSIBILITIES.

Aqueous Iodine Chemistry
NUREG/CR-3514 V02: THE CHEMICAL BEHAVIOR OF IODINE IN AQUEOUS SOLUTIONS UP TO 150 C.II.Radiation-Redox Conditions.

Aquifer Restoration
NUREG/CR-3709: METHODS OF MINIMIZING GROUND-WATER CONTAMINATION FROM IN SITU LEACH URANIUM MINING.Final Report.

Artificial Intelligence
NUREG/CR-3481 V02: NUCLEAR POWER PLANT PERSONNEL QUALIFICATIONS AND TRAINING: TAPS -- The Task Analysis Profiling System.
NUREG/CR-4272: RESPONSE TREE EVALUATION:EXPERIMENTAL ASSESSMENT OF AN EXPERT SYSTEM FOR NUCLEAR REACTOR OPERATORS.

Asiatic Clam
NUREG/CR-4233: DISTRIBUTION OF CORBICULA FLUMINEA AT NUCLEAR FACILITIES.

Atmosphere
NUREG/CR-4074: THE PERFORMANCE OF DEFECTED SPENT LWR FUEL RODS IN INERT GAS AND DRY AIR STORAGE ATMOSPHERES.

Atmospheric Dispersion
NUREG/CR-4157: A SCIENTIFIC CRITIQUE OF AVAILABLE MODELS FOR REAL-TIME SIMULATIONS OF DISPERSION.
NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING.
NUREG/CR-4159: COMPARISON OF THE 1981 INEL DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS.

Atomization
NUREG/CR-3937: STEAM GENERATOR TUBE RUPTURE IODINE TRANSPORT MECHANISMS. Task 1:Experimental Studies.
NUREG/CR-4383: HIGH PRESSURE INJECTION OF MELT FROM A REACTOR PRESSURE VESSEL - THE DISCHARGE PHASE.

Austenitic Stainless Steel
NUREG/CR-3613 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Annual Report For 1984.
NUREG/CR-3613 V03 N1: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Semiannual Report For October 1984 Through March 1985.

Auxiliary Feedwater System
NUREG/CR-4228: REVIEW OF THE VOGTLE UNITS 1 AND 2 AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS.

B-Value
NUREG/CR-4145: EARTHQUAKE RECURRENCE INTERVALS AT NUCLEAR POWER PLANTS: ANALYSIS AND RANKING.

BWIP
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.

BWR-LTAS
NUREG/CR-3764: BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE.

Balance Study
NUREG/CR-4355 V01: 238 PU(IV) IN MONKEYS.Overview Of Metabolism.

Barrier Degradation Event
NUREG/CR-4067: SUMMARY OF BARRIER DEGRADATION EVENTS AND SMALL ACCIDENTS IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.

Basalt
NUREG/CR-2663 V01: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Main Report.
NUREG/CR-2663 V02: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Appendices.
NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4114: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS.II.
NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984.
NUREG/CR-4303: HIGH-LEVEL WASTE PRECLOSURE SYSTEMS SAFETY ANALYSIS.Phase 1, Final Report.

Bayesian Method
NUREG/CR-4145: EARTHQUAKE RECURRENCE INTERVALS AT NUCLEAR POWER PLANTS: ANALYSIS AND RANKING.

Below-Water-Table
NUREG/CR-4061: LEACHATE PLUME MIGRATION DOWNGRADIENT FROM URANIUM TAILINGS DISPOSAL IN MINE STOPES.

Belowground Vault
NUREG/CR-3774 V02: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2A:Technical Requirements For Belowground Vault Disposal Of Low Level Radioactive Waste.

Benchmark Problems
NUREG/CR-1677 V02: PIPING BENCHMARK PROBLEMS.VOLUME II DYNAMIC ANALYSIS INDEPENDENT SUPPORT MOTION RESPONSE SPECTRUM METHOD.

Bentonite
NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT.

Beta Dose Rate
NUREG/CR-4203: A CALCULATIONAL METHOD FOR DETERMINING BIOLOGICAL DOSE RATES FROM IRRADIATED RESEARCH REACTOR FUEL.

Beta Radiation
NUREG/CR-4266: STANDARD BETA-PARTICLE AND MONOENERGETIC ELECTRON SOURCES FOR THE CALIBRATION OF BETA-RADIATION PROTECTION INSTRUMENTATION.

Binary Liquid Activity
NUREG/CR-4401: BEHAVIOR OF CONTROL RODS DURING CORE DEGRADATION: PRESSURIZATION OF SILVER-INDIUM-CADMIUM CONTROL RODS.

Bioaccumulation
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.

Bioassay
NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984.

Biodegradation
NUREG/CR-4200: BIODEGRADATION TESTING OF SOLIDIFIED LOW-LEVEL WASTE STREAMS.

Biofouling
NUREG/CR-4070 V02: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Volume 2:Current Status Of Biofouling Surveillance And Control Techniques.

Bivalve Fouling
NUREG/CR-4070 V03: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Factors That May Intensify The Safety Consequences Of Biofouling.

Blowdown
NUREG/CR-4196: OVERVIEW OF TRAC-BD1 (VERSION 12) ASSESSMENT STUDIES.

Body Waves
NUREG/CR-4354: A STUDY OF SEISMICITY AND TECTONICS IN NEW ENGLAND.Final Report.

Bomb Threat
NUREG-0525 R10: SAFEGUARDS SUMMARY EVENT LIST (SSEL),REVISION 10.

Boolean Algebra
NUREG/CR-4213: SETS REFERENCE MANUAL.

Borehole
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

Borosilicate Glass
NUREG/CR-3900 V04: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Annual Report,April 1984 - April 1985.

Break Flow Rate
NUREG/CR-4041: SYSTEM ANALYSIS HANDBOOK.

Breaker
NUREG/CR-3791: CLOSEOUT OF IE BULLETIN 79-09:FAILURE OF GE TYPE AK-2 CIRCUIT BREAKERS IN SAFETY-RELATED SYSTEMS.

Bubble Behavior
NUREG/CR-3830 V02: AEROSOL RELEASE AND TRANSPORT PROGRAM.Semiannual Progress Report For April 1984-September 1984.

Budget
NUREG-1100 V01: FY 1986 BUDGET ESTIMATES.

Buoyant Plume
NUREG/CR-3426 V01: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY. Facility And Test Design Report.
NUREG/CR-3426 V02: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY.Data Report.

Burial
NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984.

Burial Environment
NUREG/CR-4083: ANALYSES OF SOILS FROM THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITES AT BARNWELL,SC AND RICHLAND,WA.

Burial Site
NUREG/CR-3865: EVALUATION OF THE RADIOACTIVE INVENTORY IN,AND ESTIMATION OF ISOTOPIC RELEASE FROM,THE WASTE IN EIGHT TRENCHES AT THE SHEFFIELD LOW-LEVEL WASTE BURIAL SITE.

Buried Waste
NUREG/CR-4083: ANALYSES OF SOILS FROM THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITES AT BARNWELL,SC AND RICHLAND,WA.
NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA.

Burner Location
NUREG/CR-4112 V01: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task A:IEEE Flame Test.

By-Product Material Facility
NUREG/CR-3657: PRELIMINARY SCREENING OF FUEL CYCLE AND BY-PRODUCT MATERIAL LICENSES FOR EMERGENCY PLANNING.

Bypass Test
NUREG/CR-4252: INDEPENDENT ASSESSMENT OF TRAC-PD2/MOD1 CODE WITH BCL ECC BYPASS TESTS.

C&D LCU-13 Battery Cell
NUREG/CR-4096: TEST SERIES 3:SEISMIC FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E C&D LCU-13 BATTERY CELLS.

CDA
NUREG/CR-3944: TRAN B-3:EXPERIMENTAL INVESTIGATION OF FUEL CRUST STABILITY ON MELTING SURFACES OF AN ANNULAR FLOW CHANNEL.

CITADEL
NUREG-1108: RADIOACTIVITY TRANSPORT FOLLOWING STEAM GENERATOR TUBE RUPTURE.

COBRA
NUREG/CR-3810 V03: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,July-September 1984.
NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,October-December 1984.
NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,January-March 1985.

COBRA-TF
NUREG/CR-4166: ANALYSIS OF FLECHT-SEASET 163-ROD BLOCKED BUNDLE DATA USING COBRA-TF.

CONTAIN
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.
NUREG/CR-4340 V01: REACTOR SAFETY RESEARCH SEMIANNUAL REPORT.January - June 1985.

CONTING
NUREG/CR-4071: EXPLORATORY TREND AND PATTERN ANALYSIS FOR 1981 LICENSEE EVENT REPORT DATA.

CORCON
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).
NUREG/CR-4340 V01: REACTOR SAFETY RESEARCH SEMIANNUAL REPORT.January - June 1985.

CORRAL
NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS.

CORRAL-2
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.

CORSOR
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).
NUREG/CR-4173: CORSOR USER'S MANUAL.

CRAC2
NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE.
NUREG/CR-4199: A DEMONSTRATION UNCERTAINTY/SENSITIVITY ANALYSIS USING THE HEALTH AND ECONOMIC CONSEQUENCE MODEL CRAC2.

CRT Display
NUREG/CR-3767: INTERACTIVE SIMULATOR EVALUATION FOR CRT-GENERATED DISPLAYS.
NUREG/CR-4227: HUMAN ENGINEERING GUIDELINES FOR THE EVALUATION AND ASSESSMENT OF VIDEO DISPLAY UNITS.

CRW
NUREG/CR-4070 V02: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Volume 2:Current Status Of Biofouling Surveillance And Control Techniques.
NUREG/CR-4070 V03: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Factors That May Intensify The Safety Consequences Of Biofouling.

Cable
NUREG/CR-4112 V01: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task A:IEEE Flame Test.
NUREG/CR-4112 V02: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task B:Firestop Test Method.

Calibration
NUREG/CR-4266: STANDARD BETA-PARTICLE AND MONOENERGETIC ELECTRON SOURCES FOR THE CALIBRATION OF BETA-RADIATION PROTECTION INSTRUMENTATION.

Canada
NUREG/CR-4317 V01: CANADIAN SEISMIC AGREEMENT.Technical Report Covering 1979-1985.

Capacitor Discharge Vaporization Experiment
NUREG/CR-4346: AEROSOL RELEASE EXPERIMENTS IN THE FUEL AEROSOL SIMULANT TEST FACILITY:UNDERSODIUM EXPERIMENTS.

Cash Flow Analysis
NUREG-1131: FINANCIAL ANALYSIS OF POTENTIAL RETROSPECTIVE PREMIUM ASSESSMENTS UNDER THE PRICE-ANDERSON SYSTEM.

Cast Stainless Steel
NUREG/CR-3998 V02: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,April-June 1984.
NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984.
NUREG/CR-4204: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS:Annual Report,October 1983 - September 1984.

Cement
NUREG/CR-4181: LEACHABILITY OF RADIONUCLIDES FROM CEMENT SOLIDIFIED WASTE FORMS PRODUCED AT OPERATING NUCLEAR POWER REACTORS.

Cesium Hydroxide
NUREG/CR-3197 V01: REACTION BETWEEN SOME CESIUM-IODINE COMPOUNDS AND THE REACTOR MATERIALS 304 STAINLESS STEEL,INCONEL 600 & SILVER. Volume I:Cesium Hydroxide Reactions.

Charpy
NUREG/CR-4092: ORNL CHARACTERIZATION OF HEAVY-SECTION STEEL TECHNOLOGY PROGRAM PLATES 01,02,AND 03.

Chelating Agent
NUREG/CR-4435: ORGANIC COMPLEXANT-ENHANCED MOBILITY OF TOXIC ELEMENTS IN LOW-LEVEL WASTES.Annual Report,July 1984 - June 1985.

Chemical Cleaning
NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.
NUREG/CR-4276: VIBRATION AND WEAR IN STEAM GENERATOR TUBES FOLLOWING CHEMICAL CLEANING - SEMIANNUAL REPORT.

Chemical Dose Rate
NUREG/CR-4008: GENERAL EXTRAPOLATION MODEL FOR AN IMPORTANT CHEMICAL DOSE-RATE EFFECT.

Chemical Stabilization
NUREG/CR-4089: EVALUATION OF FIELD-TESTED FUGITIVE DUST CONTROL TECHNIQUES FOR URANIUM MILL TAILINGS PILES.

Chloride
NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.

Circulating Raw-Water
NUREG/CR-4070 V02: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Volume 2:Current Status Of Biofouling Surveillance And Control Techniques.
NUREG/CR-4070 V03: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Factors That May Intensify The Safety Consequences Of Biofouling.

Cladding
NUREG/CR-3744 V02: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR APRIL-SEPTEMBER 1984.
NUREG/CR-3980 V02: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,April-June 1984.
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.
NUREG/CR-4218: LOCA SIMULATION IN THE NATIONAL RESEARCH UNIVERSAL REACTOR PROGRAM.Postirradiation Examination Results For The Third Materials Test (MT-3) - Second Campaign.
NUREG/CR-4219 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.
NUREG/CR-4275: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM - FIVE-YEAR PLAN FY 1984-1988.

Class 1E Pressure Transmitter
NUREG/CR-3863: ASSESSMENT OF CLASS 1E PRESSURE TRANSMITTER RESPONSE WHEN SUBJECTED TO HARSH ENVIRONMENT SCREENING TESTS.

Cleanup
NUREG/CR-4399: POSSIBLE OPTIONS FOR REDUCING OCCUPATIONAL DOSE FROM THE TMI-2 BASEMENT.

Closeout
NUREG-0905: CLOSEOUT OF IE BULLETIN 79-12:SHORT-PERIOD SCRAMS AT BOILING-WATER REACTORS.
NUREG/CR-3791: CLOSEOUT OF IE BULLETIN 79-09:FAILURE OF GE TYPE AK-2 CIRCUIT BREAKERS IN SAFETY-RELATED SYSTEMS.
NUREG/CR-3794: CLOSEOUT OF IE BULLETIN 80-25:OPERATING PROBLEMS WITH TARGET ROCK SAFETY-RELIEF VALVES AT BWRS.
NUREG/CR-4004: CLOSEOUT OF IE BULLETIN 79-25:FAILURES OF WESTINGHOUSE BFD RELAYS IN SAFETY-RELATED SYSTEMS.
NUREG/CR-4006: CLOSEOUT OF IE BULLETIN 81-01:SURVEILLANCE OF MECHANICAL SNUBBERS.

Cobalt-60 Teletherapy Incident
NUREG-1103: CONTAMINATED MEXICAN STEEL.Importation Of Steel Into The United States That Had Been Inadvertently Contaminated With Cobalt-60 As A Result Of Scrapping Of A Teletherapy Unit.

Code
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment)
NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984.
NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.
NUREG/CR-3091 V06: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1984 - March 1985.
NUREG/CR-3208: TRAC-PD2 DEVELOPMENTAL ASSESSMENT.
NUREG/CR-3319: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM:LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium.
NUREG/CR-3498: TWO-DIMENSIONAL MODELING OF INTRA-SUBASSEMBLY HEAT TRANSFER AND BUOYANCY-INDUCED FLOW REDISTRIBUTION IN LMFBRS.
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-3741 V02: EVALUATION OF POWER REACTOR FUEL ROD ANALYSIS CAPABILITIES. Phase 2 Topical Report.Volume 2:Code Evaluation.
NUREG/CR-3866: TRAC-PD2 INDEPENDENT ASSESSMENT.
NUREG/CR-3901: DOCUMENTATION AND USER'S GUIDE:GS2 & GS3 - VARIABLY SATURATED FLOW AND MASS TRANSPORT MODELS.
NUREG/CR-3972: SETTLEMENT OF URANIUM MILL TAILINGS PILES. A COMPARISON OF ANALYSIS TECHNIQUES.
NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.
NUREG/CR-4169: AN APPROACH TO TREATING RADIONUCLIDE DECAY HEATING FOR USE IN THE MELCOR CODE SYSTEM.
NUREG/CR-4172: A USER'S GUIDE FOR MERGE.

NUREG/CR-4189: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:Semiscale MOD-2A Feedwater-Line Break (S-SF-3) And Steam-Line Break (S-SF-5) Tests.
NUREG/CR-4195: OVERVIEW OF TRAC-PD2 ASSESSMENT CALCULATIONS.
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.
NUREG/CR-4252: INDEPENDENT ASSESSMENT OF TRAC-PD2/MOD1 CODE WITH BCL ECC BYPASS TESTS.
NUREG/CR-4253: REVIEW OF TRAC CALCULATIONS FOR CALVERT CLIFFS PTS STUDY.
NUREG/CR-4267: VESSEL INTEGRITY SIMULATION (VISA) CODE SENSITIVITY STUDY.
NUREG/CR-4292: A COMPARATIVE ANALYSIS OF CONSTITUTIVE RELATIONS IN TRAC-PFL AND RELAP5/MOD1.
NUREG/CR-4367: ORVIRT.PC:A 2-D FINITE-ELEMENT FRACTURE ANALYSIS PROGRAM FOR A MICROCOMPUTER.
NUREG/CR-4388: AEROSOL BEHAVIOR MODELING (TASK 3) - SUPPORT SERVICES FOR RESEARCH AND EVALUATION OF SEVERE ACCIDENT PHENOMENA AND MITIGATION.
NUREG/CR-4402 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31,1985.

Code Listing
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.

Coder's Manual
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.

Cold Leg
NUREG/CR-4115: INTERNATIONAL STANDARD PROBLEM 13 (LOFT EXPERIMENT L2-5). Final Comparison Report.

Cold-Leg Break
NUREG/CR-4044: TRAC-PF1 LOCA CALCULATIONS USING FINE-NODE AND COARSE-NODE INPUT MODELS.

Collection Efficiency
NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

Combustion
NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.
NUREG/CR-4136: SMOKE:A Data Reduction Package For Analysis Of Combustion Experiments.
NUREG/CR-4231: EVALUATION OF AVAILABLE DATA FOR PROBABILISTIC RISK ASSESSMENTS (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.

Commitment
NUREG/CR-2850 V03: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1981.

Common Cause Failure
NUREG/CR-4314: BRIEF SURVEY AND COMPARISON OF COMMON CAUSE FAILURE ANALYSIS.

Compact Specimen
NUREG/CR-4283: STUDY OF THE EFFECTS OF ELASTIC UNLOADINGS ON THE JI-R CURVES FROM COMPACT SPECIMENS.

Comparison Of Licensing Activities
NUREG-1110: COMPARISON OF LICENSING ACTIVITIES FOR OPERATING PLANTS DESIGNED BY BABCOCK & WILCOX.

Comparison Studies
NUREG/CR-4090: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Annual Summary Report - Fiscal Year 1984.

Compilation Of Rules
NUREG-0936 V04 N02: NRC REGULATORY AGENDA.Quarterly Report,April-June 1985.
NUREG-0936 V04 N03: NRC REGULATORY AGENDA.Quarterly Report,July-September 1985.
NUREG-0936 V03 N04: NRC REGULATORY AGENDA.Quarterly Report,October-December 1984.

Compliance
NUREG-0970: PROCEDURES FOR MEETING NRC ANTITRUST RESPONSIBILITIES.
NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION.

Component Degradation
NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.

Compressed Gas
NUREG/CR-3551: SAFETY IMPLICATIONS ASSOCIATED WITH IN-PLANT PRESSURIZED GAS STORAGE AND DISTRIBUTION SYSTEMS IN NUCLEAR POWER PLANTS.

Computer Code
NUREG-1108: RADIOACTIVITY TRANSPORT FOLLOWING STEAM GENERATOR TUBE RUPTURE.
NUREG/CR-3442: RADTWO:A COMPUTER CODE FOR SIMULATING FAST-TRANSIENT, TWO-DIMENSIONAL,TWO-LAYER RADIONUCLIDE CONCENTRATION CONDITIONS IN LAKES,RESERVOIRS,RIVERS,ESTUARIES,AND COASTAL REGIONS.
NUREG/CR-3633 V01 S1: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.
NUREG/CR-3633 V04: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.Volume 4: Developmental Assessment.
NUREG/CR-3764: BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE.
NUREG/CR-3772: RELAP5 ASSESSMENT:SEMISCALE SMALL BREAK TESTS S-UT-1,S-UT-2, S-UT-6,S-UT-7 AND S-UT-8.
NUREG/CR-3802: RELAP5 ASSESSMENT:QUANTITATIVE KEY PARAMETERS AND RUN TIME STATISTICS.
NUREG/CR-3810 V03: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,July-September 1984.
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.
NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT.
NUREG/CR-3913: HECTR VERSION 1.0 USER'S MANUAL.
NUREG/CR-3919: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:NEPTUNUS PRESSURIZER TEST Y05.
NUREG/CR-3936: RELAP5 ASSESSMENT:CONCLUSIONS AND USER GUIDELINES.
NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE.
NUREG/CR-4044: TRAC-PF1 LOCA CALCULATIONS USING FINE-NODE AND COARSE-NODE INPUT MODELS.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0:A Computer Code for Severe Reactor Accident Containment Analysis.
NUREG/CR-4109: TRAC-PF1 ANALYSES OF POTENTIAL PRESSURIZED-THERMAL SHOCK TRANSIENTS AT CALVERT CLIFFS/UNIT 1:A Combustion Engineering PWR.
NUREG/CR-4116: NUFEGQ-NP:A DIGITAL COMPUTER CODE FOR THE LINEAR STABILITY ANALYSIS OF BOILING WATER NUCLEAR REACTORS.
NUREG/CR-4127 V01: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 1:Numerical Methods.
NUREG/CR-4136: SMOKE:A Data Reduction Package For Analysis Of Combustion Experiments.
NUREG/CR-4155: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:NORTHWESTERN UNIVERSITY PERFORATED-PLATE CCFL TESTS.
NUREG/CR-4168: GT2F:A COMPUTER CODE FOR ESTIMATING LIGHT WATER REACTOR FUEL ROD FAILURES.
NUREG/CR-4185: AN ASSESSMENT OF DOSIMETRY DATA FOR ACCIDENTAL RADIONUCLIDE RELEASES FROM NUCLEAR REACTORS.
NUREG/CR-4192: THE ANALYSIS OF DRAINAGE AND CONSOLIDATION AT TYPICAL URANIUM MILL TAILINGS SITES.
NUREG/CR-4196: OVERVIEW OF TRAC-BD1 (VERSION 12) ASSESSMENT STUDIES.
NUREG/CR-4199: A DEMONSTRATION UNCERTAINTY/SENSITIVITY ANALYSIS USING THE HEALTH AND ECONOMIC CONSEQUENCE MODEL CRAC2.
NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS.
NUREG/CR-4260: TORAC USER'S MANUAL:A Computer Code For Analyzing Tornado-Induced Flow And Material Transport In Nuclear Facilities.

NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,January-March 1985.
NUREG/CR-4321: FULL SCALE MEASUREMENTS OF SMOKE TRANSPORT AND DEPOSITION IN VENTILATION SYSTEM DUCTWORK.
NUREG/CR-4340 V01: REACTOR SAFETY RESEARCH SEMIANNUAL REPORT.January - June 1985.
NUREG/CR-4360 V01: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.
NUREG/CR-4360 V02: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.
NUREG/CR-4396: SIMMER POSTPROCESSOR MANUAL.
NUREG/CR-4429: TRAC-BD1/MOD1 USERS GUIDELINE.

Computer Display
NUREG/CR-3767: INTERACTIVE SIMULATOR EVALUATION FOR CRT-GENERATED DISPLAYS.

Computer Model
NUREG/CR-3904: A COMPARISON OF UNCERTAINTY AND SENSITIVITY ANALYSIS TECHNIQUES FOR COMPUTER MODELS.
NUREG/CR-4042: A 3-DIMENSIONAL COMPUTER MODEL TO SIMULATE FLUID FLOW AND CONTAINMENT TRANSPORT THROUGH A ROCK FRACTURE SYSTEM.
NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUNDWATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions.

Computer Operator Aides
NUREG/CR-4227: HUMAN ENGINEERING GUIDELINES FOR THE EVALUATION AND ASSESSMENT OF VIDEO DISPLAY UNITS.

Computer Program
NUREG-1167: TPDWR2:THERMAL POWER DETERMINATION FOR WESTINGHOUSE REACTORS,VERSION2.User's Guide.
NUREG/CR-3413: OFF-SITE CONSEQUENCES OF RADIOLOGICAL ACCIDENTS:METHODS, COSTS AND SCHEDULES FOR DECONTAMINATION.
NUREG/CR-4020: HMS:A COMPUTER PROGRAM FOR TRANSIENT,THREE-DIMENSIONAL MIXING GASES.
NUREG/CR-4122: A FORTRAN 77 PROGRAM AND USER'S GUIDE FOR THE CALCULATION OF PARTIAL CORRELATION AND STANDARDIZED REGRESSION COEFFICIENTS.
NUREG/CR-4288: FOCAL MECHANISM ANALYSES FOR VIRGINIA AND EASTERN TENNESSEE EARTHQUAKES (1978-1984).
NUREG/CR-4376: HEAT TRANSFER,CARRYOVER AND FALL BACK IN PWR STEAM GENERATORS DURING TRANSIENTS.

Computer Security
NUREG/CR-4298: DESIGN AND INSTALLATION OF COMPUTER SYSTEMS TO MEET THE REQUIREMENTS OF 10 CFR 73.55.

Computer System
NUREG/CR-4298: DESIGN AND INSTALLATION OF COMPUTER SYSTEMS TO MEET THE REQUIREMENTS OF 10 CFR 73.55.

Computerized System
NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS.

Concentration
NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.

Concrete
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.

Concrete Bunker
NUREG/CR-3774 V04: ALTERNATIVE METHOD FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2C:Technical Requirements For Earth Mounded Concrete Bunker Disposal Of Low Level Radioactive Waste.

Concrete Containment
NUREG/CR-4149: ULTIMATE PRESSURE CAPACITY OF REINFORCED AND PRESTRESSED CONCRETE CONTAINMENT.

Condensation
NUREG/CR-4414: DIRECT-CONTACT CONDENSATION OF STEAM ON COLD WATER IN STRATIFIED COUNTERCURRENT FLOW.
NUREG/CR-4416: STABILITY OF STEAM-WATER COUNTERCURRENT STRATIFIED FLOW.
NUREG/CR-4417: LOCAL PROPERTIES OF COUNTERCURRENT STRATIFIED STEAM-WATER FLOW.

Cone Penetration Test
NUREG/CR-4430: CURRENT METHODOLOGIES FOR ASSESSING THE POTENTIAL FOR EARTHQUARKE-INDUCED LIQUEFACTION IN SOILS.

Confiscation
NUREG-1153: INSPECTION REPORT OF UNAUTHORIZED POSSESSION AND USE OF UNSEALED AMERICIUM-241 AND SUBSEQUENT CONFISCATION. J.C. Haynes Company,Newark,Ohio.

Congress
NUREG-0090 V08 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.Janurary-March 1985.

Consequence Analysis
NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences.

Consequence Model
NUREG/CR-4350 V07: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 7 - Environmental Transport And Consequence Analysis.

Consolidation Characteristics
NUREG/CR-4087: MEASUREMENTS OF URANIUM MILL TAILINGS CONSOLIDATION CHARACTERISTICS.

Container
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.
NUREG/CR-3900 V02: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,July-September 1984.
NUREG/CR-3900 V04: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Annual Report,April 1984 - April 1985.
NUREG/CR-4379 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report,Year Four April-June 1985.

Containment
NUREG-0800 06.2.2 R4: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 4 To Section 6.2.2, "Containment Heat Removal System."
NUREG-0897 R01: CONTAINMENT EMERGENCY SUMP PERFORMANCE.(Technical Findings Related To Unresolved Safety Issues).
NUREG-1037 DRFT FC: CONTAINMENT PERFORMANCE WORKING GROUP REPORT.Draft Report For Comment.
NUREG-1079 DRFT FC: ESTIMATES OF EARLY CONTAINMENT FROM CORE MELT ACCIDENTS. Draft Report for Comment.
NUREG-1116: A REVIEW OF THE CURRENT UNDERSTANDING OF THE POTENTIAL FOR CONTAINMENT FAILURE FROM IN-VESSEL STEAM EXPLOSIONS.
NUREG/CR-3647: DESIGN AND FABRICATION OF A 1/8-SCALE STEEL CONTAINMENT MODEL.
NUREG/CR-3803: THE EFFECTS OF POST-LOCA CONDITIONS ON A PROTECTIVE COATING (PAINT) FOR THE NUCLEAR POWER INDUSTRY.
NUREG/CR-3816 V02: REACTOR SAFETY RESEARCH.Quarterly Report,April-June 1984.
NUREG/CR-3855: CHARACTERIZATION OF NUCLEAR REACTOR CONTAINMENT PENETRATION - FINAL REPORT.
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.
NUREG/CR-3952: SEQUOYAH EQUIPMENT HATCH SEAL LEAKAGE.
NUREG/CR-3954: HECTR ANALYSIS OF EQUIPMENT TEMPERATURE RESPONSES TO SELECTED HYDROGEN BURNS IN AN ICE CONDENSER CONTAINMENT.
NUREG/CR-4020: HMS:A COMPUTER PROGRAM FOR TRANSIENT,THREE-DIMENSIONAL MIXING GASES.
NUREG/CR-4042: A 3-DIMENSIONAL COMPUTER MODEL TO SIMULATE FLUID FLOW AND CONTAINMENT TRANSPORT THROUGH A ROCK FRACTURE SYSTEM.
NUREG/CR-4064: STRUCTURAL RESPONSE OF LARGE PENETRATIONS AND CLOSURES FOR CONTAINMENT VESSELS SUBJECTED TO LOADINGS BEYOND DESIGN BASIS.
NUREG/CR-4081: ABSORPTION OF GASEOUS IODINE BY WATER DROPLETS.
NUREG/CR-4119: INTEGRITY OF CONTAINMENT PENETRATIONS UNDER SEVERE ACCIDENT CONDITIONS FY84 ANNUAL REPORT.
NUREG/CR-4137: PRETEST PREDICTIONS FOR THE RESPONSE OF A 1:8-SCALE STEEL LWR CONTAINMENT BUILDING MODEL TO STATIC OVERPRESSURIZATION.
NUREG/CR-4141: CONTAINMENT PURGE AND VENT VALVE TEST PROGRAM FINAL REPORT.

NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences.
NUREG/CR-4151: INTEGRATION OF EMERGENCY ACTION LEVELS WITH COMBUSTION ENGINEERING EMERGENCY OPERATING PROCEDURES.By Use Of Combustion Engineering Owners Group Emergency Operating Procedure Technical Guidelines.
NUREG/CR-4209: COMPARISON OF ANALYTICAL PREDICTIONS AND EXPERIMENTAL RESULTS FOR A 1:8-SCALE STEEL CONTAINMENT MODEL PRESSURIZED TO FAILURE.
NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS.
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.
NUREG/CR-4220: RELIABILITY ANALYSIS OF CONTAINMENT ISOLATION SYSTEMS.
NUREG/CR-4329: RELIABILITY EVALUATION OF CONTAINMENTS INCLUDING SOIL-STRUCTURE INTERACTION.
NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.
NUREG/CR-4432: COMPARISON OF DYNAMIC CHARACTERISTICS OF FUKUSHIMA NUCLEAR POWER PLANT CONTAINMENT BUILDING DETERMINED FROM TESTS AND EARTHQUAKES.

Contaminant
NUREG/CR-3906: URANIUM MILL TAILINGS NEUTRALIZATION:CONTAMINANT COMPLEXATION AND TAILINGS LEACHING STUDY.

Contaminated Steel Products
NUREG-1103: CONTAMINATED MEXICAN STEEL.Importation Of Steel Into The United States That Had Been Inadvertently Contaminated With Cobalt-60 As A Result Of Scrapping Of A Teletherapy Unit.

Contamination
NUREG/CR-3709: METHODS OF MINIMIZING GROUND-WATER CONTAMINATION FROM IN SITU LEACH URANIUM MINING.Final Report.
NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions.
NUREG/CR-4251 V02: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 2:Case Study Analysis Of Hydrologic Characterization And Mitigative Schemes.

Contract
NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984.

Control Rod
NUREG-0090 V07 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1984.
NUREG/CR-4401: BEHAVIOR OF CONTROL RODS DURING CORE DEGRADATION: PRESSURIZATION OF SILVER-INDIUM-CADMIUM CONTROL RODS.

Control Room
NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS.
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.
NUREG/CR-4280: THE EFFECTS OF SUPERVISOR EXPERIENCE AND ASSISTANCE OF A SHIFT TECHNICAL ADVISOR (STA) ON CREW PERFORMANCE IN CONTROL ROOM SIMULATORS.

Control System
NUREG/CR-4262 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR.Main Report.
NUREG/CR-4326 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Main Report.
NUREG/CR-4385: EFFECTS OF CONTROL SYSTEM FAILURES IN TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A WESTINGHOUSE PRESSURIZED WATER REACTOR.
NUREG/CR-4386: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A BABCOCK AND WILCOX PRESSURIZED WATER REACTOR.
NUREG/CR-4387: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS AND CORE-MELT FREQUENCIES AT A GENERAL ELECTRIC PRESSURIZED WATER REACTOR.

Control System Failure
NUREG/CR-4262 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR. Appendices.
NUREG/CR-4326 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Appendices.

Coolant Boilaway
NUREG/CR-3810 V03: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,July-September 1984.

Coolant Boilaway And Damage
NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,January-March 1985.

Cooling Pond Modeling
NUREG/CR-4120: MATHEMATICAL MODELING OF ULTIMATE HEAT SINK COOLING PONDS.

Corbicula Fluminea
NUREG/CR-4233: DISTRIBUTION OF CORBICULA FLUMINEA AT NUCLEAR FACILITIES.

Core
NUREG/CR-3485: PRA REVIEW MANUAL.
NUREG/CR-3757: TRAN B-2:THE EFFECT OF LOW STEEL CONTENT ON FUEL PENETRATION IN A NON-MELTING CYLINDRICAL FLOW CHANNEL.
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.
NUREG/CR-3948: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY.
NUREG/CR-4080: DETERMINATION OF THE AVAILABILITY OF CORE EXIT THERMOCOUPLES DURING SEVERE ACCIDENT SITUATIONS.

Core Cooling
NUREG/CR-4060: THE DC-1 AND DC-2 DEBRIS COOLABILITY AND MELT DYNAMICS EXPERIMENTS.

Core Damage
NUREG-1037 DRFT FC: CONTAINMENT PERFORMANCE WORKING GROUP REPORT.Draft Report For Comment.
NUREG/CR-4050: A REVIEW OF THE SHOREHAM NUCLEAR POWER STATION PROBABILISTIC RISK ASSESSMENT.Internal Events And Core Damage Frequency.

Core Degradation
NUREG/CR-4173: CORSOR USER'S MANUAL.
NUREG/CR-4401: BEHAVIOR OF CONTROL RODS DURING CORE DEGRADATION: PRESSURIZATION OF SILVER-INDIUM-CADMIUM CONTROL RODS.

Core Disruptive Accident
NUREG/CR-4240 V01: PHYSICS OF REACTOR SAFETY.Quarterly Report,January-March 1985.
NUREG/CR-3804 V03: PHYSICS OF REACTOR SAFETY.Quarterly Report,July-September 1984.
NUREG/CR-3804 V04: PHYSICS OF REACTOR SAFETY.Quarterly Report,October-December 1984.
NUREG/CR-3944: TRAN B-3:EXPERIMENTAL INVESTIGATION OF FUEL CRUST STABILITY ON MELTING SURFACES OF AN ANNULAR FLOW CHANNEL.

Core Melt
NUREG-1079 DRFT FC: ESTIMATES OF EARLY CONTAINMENT FROM CORE MELT ACCIDENTS. Draft Report for Comment.
NUREG-1116: A REVIEW OF THE CURRENT UNDERSTANDING OF THE POTENTIAL FOR CONTAINMENT FAILURE FROM IN-VESSEL STEAM EXPLOSIONS.
NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT.
NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.
NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions.
NUREG/CR-4251 V02: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 2:Case Study Analysis Of Hydrologic Characterization And Mitigative Schemes.

NUREG/CR-4385: EFFECTS OF CONTROL SYSTEM FAILURES IN TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A WESTINGHOUSE PRESSURIZED WATER REACTOR.
NUREG/CR-4386: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A BABCOCK AND WILCOX PRESSURIZED WATER REACTOR.
NUREG/CR-4387: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS AND CORE-MELT FREQUENCIES AT A GENERAL ELECTRIC PRESSURIZED WATER REACTOR.
NUREG/CR-4400: THE IMPACT OF MECHANICAL-AND MAINTENANCE-INDUCED FAILURES OF MAIN REACTOR COOLANT PUMP SEALS ON PLANT SAFETY.

Core Reflooding System
NUREG/CR-4277: INVERTED ANNUAL FLOW EXPERIMENTAL STUDY.

Corporate Data Network
NUREG/CR-4322 V01: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 1: Enterprise Model.
NUREG/CR-4322 V02: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 2: Data Entity Dictionary.
NUREG/CR-4322 V03: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 3: Data Model.
NUREG/CR-4322 V04: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 4: Preliminary Strategic Data Plan.

Corrective Maintenance
NUREG/CR-4377: EVALUATIONS AND UTILIZATIONS OF RISK IMPORTANCES.

Corrosion
NUREG-1095: EVALUATION OF RESPONSES TO IE BULLETIN 82-02.Degradation Of Threaded Fasteners In Reactor Coolant Pressure Boundary Of Pressurized Water-Reactor Plants.
NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.
NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.

Corrosion Assisted Fatigue
NUREG-1155 V01: RESEARCH PROGRAM PLAN.Reactor Vessels.
NUREG-1155 V03: RESEARCH PROGRAM PLAN.Piping.

Corrosion Fatigue
NUREG/CR-4422: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN PWR ENVIRONMENTS.

Cost
NUREG/CR-3293 V01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Main Report.
NUREG/CR-3293 V02: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Appendices.
NUREG/CR-4090: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Annual Summary Report - Fiscal Year 1984.
NUREG/CR-4268: RATIO METHODS FOR COST-EFFECTIVE FIELD SAMPLING OF COMMERCIAL RADIOACTIVE LOW-LEVEL WASTES.
NUREG/CR-4373: COMPENDIUM OF COST-EFFECTIVENESS EVALUATIONS OF MODIFICATIONS FOR DOSE REDUCTION AT NUCLEAR POWER PLANTS.
NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.
NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

Counter-Current Steam
NUREG/CR-4415: COUNTER-CURRENT STEAM/WATER FLOW ABOVE A PERFORATED PLATE-VERTICAL INJECTION OF WATER.

Countercurrent Flow
NUREG/CR-4414: DIRECT-CONTACT CONDENSATION OF STEAM ON COLD WATER IN STRATIFIED COUNTERCURRENT FLOW.
NUREG/CR-4417: LOCAL PROPERTIES OF COUNTERCURRENT STRATIFIED STEAM-WATER FLOW.

Coupled-Process Model
NUREG/CP-0062: PROCEEDINGS OF THE CONFERENCE ON THE APPLICATION OF GEOCHEMICAL MODELS TO HIGH-LEVEL NUCLEAR WASTE REPOSITORY ASSESSMENT.

Crack
NUREG/CR-3228 V03: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1984.
NUREG/CR-3660 V04: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 4:Pipe Failure Induced By Crack Growth In West Coast Plants.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4275: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM - FIVE-YEAR PLAN FY 1984-1988.
NUREG/CR-4305: COMMENTS ON THE LEAK-BEFORE-BREAK CONCEPT FOR NUCLEAR POWER PLANT PIPING SYSTEMS.

Crack Arrest
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1.Investigation Of Warm Prestressing And Upper-Shelf Arrest.

Crack Growth
NUREG/CR-3744 V02: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR APRIL-SEPTEMBER 1984.
NUREG/CR-3998 V03: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,October-December 1984.
NUREG/CR-4219 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.
NUREG/CR-4287: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Annual Report,October 1983 - September 1984.

Crew Performance
NUREG/CR-4280: THE EFFECTS OF SUPERVISOR EXPERIENCE AND ASSISTANCE OF A SHIFT TECHNICAL ADVISOR (STA) ON CREW PERFORMANCE IN CONTROL ROOM SIMULATORS.

Crew Training
NUREG/CR-4258: AN APPROACH TO TEAM SKILLS TRAINING OF NUCLEAR POWER PLANT CONTROL ROOM CREWS.

Critical Flow
NUREG/CR-3866: TRAC-PD2 INDEPENDENT ASSESSMENT.

Crust Stability
NUREG/CR-3944: TRAN B-3.EXPERIMENTAL INVESTIGATION OF FUEL CRUST STABILITY ON MELTING SURFACES OF AN ANNULAR FLOW CHANNEL.

D10 Experiment
NUREG/CR-4055: THE D10 EXPERIMENT.COOLABILITY OF UO2 DEBRIS IN SODIUM WITH DOWNWARD HEAT REMOVAL.

DEGB
NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3663 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 1:Summary Report.
NUREG/CR-4290 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF BABCOCK AND WILCOX PWR PLANTS.Volume 2:Guillotine Break Indirectly Induced By Earthquakes.

Damage Control
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP).A Methodology For Estimating Risk Impacts Of Safeguards Measures.

Damage Definition
NUREG/CR-4112 V01: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task A:IEEE Flame Test.

Damage Mitigation
NUREG-1155 V03: RESEARCH PROGRAM PLAN.Piping.

Damper Response Time
NUREG/CR-4232: THE RESPONSE OF VENTILATION DAMPERS TO LARGE AIRFLOW PULSES.

Data
NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS.

Data Analysis
NUREG/CR-4138: DATA ANALYSES FOR NEVADA TEST SITE (NTS) PREMIXED COMBUSTION TESTS.

Data Bank
NUREG/CR-2531 R03: INTRODUCTORY USER'S MANUAL FOR THE U.S. NUCLEAR REGULATORY COMMISSION REACTOR SAFETY RESEARCH DATA BANK.
NUREG/CR-4009: HUMAN RELIABILITY DATA BANK.Evaluation Results.

Data Base
NUREG-1148: NUCLEAR POWER PLANT FIRE PROTECTION RESEARCH PROGRAM.
NUREG/CR-3319: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium.
NUREG/CR-3413: OFF-SITE CONSEQUENCES OF RADIOLOGICAL ACCIDENTS:METHODS, COSTS AND SCHEDULES FOR DECONTAMINATION.
NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.

Data Development
NUREG/CR-4350 V06: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 6 - Data Development.

Data Entity Dictionary
NUREG/CR-4322 V02: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 2: Data Entity Dictionary.

Data Model
NUREG/CR-4322 V03: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 3: Data Model.

Data Processing
NUREG/CR-4322 V01: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 1: Enterprise Model.
NUREG/CR-4322 V02: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 2: Data Entity Dictionary.
NUREG/CR-4322 V03: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 3: Data Model.
NUREG/CR-4322 V04: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 4: Preliminary Strategic Data Plan.

Debris Bed
NUREG/CR-4060: THE DC-1 AND DC-2 DEBRIS COOLABILITY AND MELT DYNAMICS EXPERIMENTS.

Debris Cooling
NUREG/CR-4055: THE D10 EXPERIMENT:COOLABILITY OF UO2 DEBRIS IN SODIUM WITH DOWNWARD HEAT REMOVAL.

Decay Heat
NUREG/CR-2951: THE D9 EXPERIMENT.Heat Removal From Stratified UO2 Debris.
NUREG/CR-4041: SYSTEM ANALYSIS HANDBOOK.
NUREG/CR-4055: THE D10 EXPERIMENT:COOLABILITY OF UO2 DEBRIS IN SODIUM WITH DOWNWARD HEAT REMOVAL.
NUREG/CR-4169: AN APPROACH TO TREATING RADIONUCLIDE DECAY HEATING FOR USE IN THE MELCOR CODE SYSTEM.

Decision Aid
NUREG/CR-4272: RESPONSE TREE EVALUATION:EXPERIMENTAL ASSESSMENT OF AN EXPERT SYSTEM FOR NUCLEAR REACTOR OPERATORS.

Decisionmaking
NUREG/CR-4040: OPERATIONAL DECISIONMAKING AND ACTION SELECTION UNDER PSYCHOLOGICAL STRESS IN NUCLEAR POWER PLANTS.

Decommissioning
NUREG/CR-1755 ADD01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING NUCLEAR REACTORS AT MULTIPLE-REACTOR STATIONS.Effects On Decommissioning Of Interim Inability To Dispose Of Wastes Offsite.
NUREG/CR-3293 V01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Main Report.
NUREG/CR-3293 V02: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Appendices.
NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.
NUREG/CR-4090: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Annual Summary Report - Fiscal Year 1984.

Decontamination
NUREG-1153: INSPECTION REPORT OF UNAUTHORIZED POSSESSION AND USE OF UNSEALED AMERICIUM-241 AND SUBSEQUENT CONFISCATION. J.C. Haynes Company,Newark,Ohio.
NUREG/CR-3413: OFF-SITE CONSEQUENCES OF RADIOLOGICAL ACCIDENTS:METHODS, COSTS AND SCHEDULES FOR DECONTAMINATION.
NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.
NUREG/CR-4399: POSSIBLE OPTIONS FOR REDUCING OCCUPATIONAL DOSE FROM THE TMI-2 BASEMENT.

Defect
NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.

Degradation
NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.
NUREG/CR-4067: SUMMARY OF BARRIER DEGRADATION EVENTS AND SMALL ACCIDENTS IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.
NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.
NUREG/CR-4256: MEASUREMENT OF RESPONSE TIME AND DETECTION OF DEGRADATION IN PRESSURE SENSOR/SENSING LINE SYSTEMS.

Degraded Core
NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT.

Demineralization System
NUREG/CR-4150: EPICOR-II RESIN DEGRADATION RESULTS FROM FIRST RESIN SAMPLES OF PF-8 AND PF-20.

Density Profile
NUREG/CR-4358: APPLICATIONS OF DENSITY PROFILING TO EQUIPMENT QUALIFICATION ISSUES.

Density-Wave
NUREG/CR-4116: NUFEGQ-NP:A DIGITAL COMPUTER CODE FOR THE LINEAR STABILITY ANALYSIS OF BOILING WATER NUCLEAR REACTORS.

Deposition
NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984.

Deposition Modeling
NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING.

Deposition Velocity
NUREG/CR-4157: A SCIENTIFIC CRITIQUE OF AVAILABLE MODELS FOR REAL-TIME SIMULATIONS OF DISPERSION.

Design Accident
NUREG/CR-3952: SEQUOYAH EQUIPMENT HATCH SEAL LEAKAGE.

Design Basis Event
NUREG/CR-3660 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 1:Summary Report.

Dewatering
NUREG/CR-4192: THE ANALYSIS OF DRAINAGE AND CONSOLIDATION AT TYPICAL URANIUM MILL TAILINGS SITES.

Diesel Generator
NUREG/CR-3831: THE IN-PLANT RELIABILITY DATA BASE FOR NUCLEAR PLANT COMPONENTS.Interim Report - Diesel Generators,Batteries, Chargers And Inverters.
NUREG/CR-4347: EMERGENCY DIESEL GENERATOR OPERATING EXPERIENCE,1981-1983.
NUREG/CR-4440: A REVIEW OF EMERGENCY DIESEL GENERATOR PERFORMANCE AT NUCLEAR POWER PLANTS.

Differential Integral Conservation Equation
NUREG/CR-3989: TIME- AND VOLUME-AVERAGED CONSERVATION EQUATIONS FOR MULTIPHASE FLOW.Part One:System Without Internal Solid Structures.

Diffusion
NUREG/CR-4072: THE ESTIMATION OF ATMOSPHERIC DISPERSION AT NUCLEAR POWER PLANTS UTILIZING REAL TIME ANEMOMETER STATISTICS.

Diffusion Flame
NUREG/CR-3638: HYDROGEN-STEAM JET-FLAME FACILITY AND EXPERIMENTS.

Digests And Indexes
NUREG-0750 V20 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY-SEPTEMBER 1984.
NUREG-0750 V20 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY-DECEMBER 1984.
NUREG-0750 V22 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-September 1985.

Diode Radiation Detector
NUREG/CR-4131: INVESTIGATION OF ALTERNATIVE MEANS TO ACCOMPLISH THE GOALS OF BIENNIAL ION CHAMBER CALIBRATION.

Directory Of Certificate Of Compliance
NUREG-0383 V01 R08: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Packages.
NUREG-0383 V02 R08: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Certificates of Compliance.
NUREG-0383 V03 R05: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Quality Assurance Programs For Radioactive Material Packages.

Discrete Element
NUREG/CR-3442: RADTWO:A COMPUTER CODE FOR SIMULATING FAST-TRANSIENT, TWO-DIMENSIONAL,TWO-LAYER RADIONUCLIDE CONCENTRATION CONDITIONS IN LAKES,RESERVOIRS,RIVERS,ESTUARIES,AND COASTAL REGIONS.

Dispersion Model
NUREG/CR-4157: A SCIENTIFIC CRITIQUE OF AVAILABLE MODELS FOR REAL-TIME SIMULATIONS OF DISPERSION.
NUREG/CR-4159: COMPARISON OF THE 1981 INEL DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS.

Disposal
NUREG-1046: DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN UNSATURATED ZONE:TECHNICAL CONSIDERATIONS AND RESPONSE TO COMMENTS.
NUREG-1164: INFORMATION ON THE CONFINEMENT CAPABILITY OF THE FACILITY DISPOSAL AREA AT WEST VALLEY,NEW YORK.
NUREG/CR-1755 ADD01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING NUCLEAR REACTORS AT MULTIPLE-REACTOR STATIONS.Effects On Decommissioning Of Interim Inability To Dispose Of Wastes Offsite.
NUREG/CR-4061: LEACHATE PLUME MIGRATION DOWNGRADIENT FROM URANIUM TAILINGS DISPOSAL IN MINE STOPES.
NUREG/CR-4352: SUGGESTED STATE REQUIREMENTS AND CRITERIA FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE REGULATORY PROGRAM.

Disposal Site
NUREG/CR-4069: ANALYSES OF SOILS FROM AN AREA ADJACENT TO THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE AT SHEFFIELD,ILLINOIS.
NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION.

Disposition Schedule
NUREG-0910 R01 S01: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE.
NUREG-0910 R01 S02: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE.
NUREG-0910 R01 S03: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE.
NUREG-0910 R01 S04: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE.

Distribution Coefficient
NUREG/CR-4094: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN SULFATE LAKE ENVIRONMENTS.

Dose
NUREG/CP-0066: PROCEEDINGS OF AN INTERNATIONAL WORKSHOP ON HISTORIC DOSE EXPERIENCE AND DOSE REDUCTION (ALARA) AT NUCLEAR POWER PLANTS,MAY 29-JUNE 1,1984.
NUREG/CR-2850 V03: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1981.
NUREG/CR-3469 V02: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS. Annotated Bibliography Of Selected Readings In Radiation Protection And ALARA.
NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984.
NUREG/CR-4068: SUMMARY OF HISTORICAL EXPERIENCE WITH RELEASES OF RADIOACTIVE MATERIALS FROM COMMERCIAL NUCLEAR POWER PLANTS IN THE UNITED STATES.
NUREG/CR-4159: COMPARISON OF THE 1981 INEL DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS.
NUREG/CR-4160: HISTORICAL SUMMARY OF OCCUPATIONAL RADIATION EXPOSURE EXPERIENCE IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.
NUREG/CR-4203: A CALCULATIONAL METHOD FOR DETERMINING BIOLOGICAL DOSE RATES FROM IRRADIATED RESEARCH REACTOR FUEL.
NUREG/CR-4239: ANALYSIS OF THE ABILITY OF CURRENT HEALTH PHYSICS INSTRUMENTS TO PREDICT DOSE IN EXPOSED INDIVIDUALS.
NUREG/CR-4266: STANDARD BETA-PARTICLE AND MONOENERGETIC ELECTRON SOURCES FOR THE CALIBRATION OF BETA-RADIATION PROTECTION INSTRUMENTATION.
NUREG/CR-4373: COMPENDIUM OF COST-EFFECTIVENESS EVALUATIONS OF MODIFICATIONS FOR DOSE REDUCTION AT NUCLEAR POWER PLANTS.

Dose Reduction
NUREG/CR-4254: OCCUPATIONAL DOSE REDUCTION AND ALARA AT NUCLEAR POWER PLANTS.Study On High-Dose Jobs,Radwaste Handling,And ALARA Incentives.

Dosimeter
NUREG/CR-4239: ANALYSIS OF THE ABILITY OF CURRENT HEALTH PHYSICS INSTRUMENTS TO PREDICT DOSE IN EXPOSED INDIVIDUALS.
NUREG/CR-4297: EXTREMITY MONITORING:Considerations For Use,Dosimeter Placement,And Evaluation.

Dosimetry
NUREG/CR-3319: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM:LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium.
NUREG/CR-3746 V02: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.Semiannual Progress Report,April 1984 - September 1984.
NUREG/CR-3746 V03: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.1984 Annual Report,October 1,1983 - September 30, 1984.
NUREG/CR-4131: INVESTIGATION OF ALTERNATIVE MEANS TO ACCOMPLISH THE GOALS OF BIENNIAL ION CHAMBER CALIBRATION.
NUREG/CR-4185: AN ASSESSMENT OF DOSIMETRY DATA FOR ACCIDENTAL RADIONUCLIDE RELEASES FROM NUCLEAR REACTORS.

Double-Ended Guillotine Break
NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3663 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 1:Summary Report.
NUREG/CR-3663 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR

PLANTS.Volume 3:Double Ended Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-4290 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF BABCOCK AND WILCOX PWR PLANTS.Volume 2:Guillotine Break Indirectly Induced By Earthquakes.

Downcomer Pressure
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.

Downward Heat Removal
NUREG/CR-4055: THE D10 EXPERIMENT:COOLABILITY OF UO2 DEBRIS IN SODIUM WITH DOWNWARD HEAT REMOVAL.

Drainage
NUREG/CR-4192: THE ANALYSIS OF DRAINAGE AND CONSOLIDATION AT TYPICAL URANIUM MILL TAILINGS SITES.

Drills
NUREG/CR-4258: AN APPROACH TO TEAM SKILLS TRAINING OF NUCLEAR POWER PLANT CONTROL ROOM CREWS.

Drop-Size Distribution
NUREG/CR-4424: DROPLET SIZES,DYNAMICS AND DEPOSITION IN VERTICAL ANNULAR FLOW.

Dry Air Storage
NUREG/CR-4074: THE PERFORMANCE OF DEFECTED SPENT LWR FUEL RODS IN INERT GAS AND DRY AIR STORAGE ATMOSPHERES.

Dry Deposition
NUREG/CR-4157: A SCIENTIFIC CRITIQUE OF AVAILABLE MODELS FOR REAL-TIME SIMULATIONS OF DISPERSION.
NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING.

Dry Spent Fuel Storage
NUREG/CR-4084: DRY SPENT FUEL STORAGE TEST PLAN FOR DESTRUCTIVE FUEL ROD EXAMINATIONS.

Dry Storage
NUREG/CR-4345: INVESTIGATION OF THE STABILITY OF LWR SPENT FUEL RODS BELOW 250 C.

Dynamic Analysis
NUREG/CR-1677 V02: PIPING BENCHMARK PROBLEMS.VOLUME II DYNAMIC ANALYSIS INDEPENDENT SUPPORT MOTION RESPONSE SPECTRUM METHOD.

Dynamic Soil Test
NUREG/CR-4430: CURRENT METHODOLOGIES FOR ASSESSING THE POTENTIAL FOR EARTHQUARKE-INDUCED LIQUEFACTION IN SOILS.

EBR
NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.

ECCS
NUREG/CR-4277: INVERTED ANNUAL FLOW EXPERIMENTAL STUDY.

EOCMEC
NUREG/CR-4288: FOCAL MECHANISM ANALYSES FOR VIRGINIA AND EASTERN TENNESSEE EARTHQUAKES (1978-1984).

EPICOR-II
NUREG/CR-4150: EPICOR-II RESIN DEGRADATION RESULTS FROM FIRST RESIN SAMPLES OF PF-8 AND PF-20.

ESF
NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.

Earth Mounded Concrete Bunker Disposal
NUREG/CR-3774 V04: ALTERNATIVE METHOD FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2C:Technical Requirements For Earth Mounded Concrete Bunker Disposal Of Low Level Radioactive Waste.

Earthen Radon Suppression Cover
NUREG/CR-3752: EFFECTS OF HYDROLOGIC VARIABLES ON ROCK RIPRAP DESIGN FOR URANIUM TAILINGS IMPOUNDMENTS.

Earthquake
NUREG-1061 V02: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2:Evaluation Of Seismic Designs - A Review Of Seismic Design Requirements For Nuclear Power Plant Piping.
NUREG-1061 V02 ADD: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2 Addendum:Summary And Evaluation Of Historical Strong-Motion Earthquake Seismic Response And Damage To Aboveground Industrial Piping.
NUREG/CP-0070: PROCEEDINGS OF THE WORKSHOP ON SEISMIC AND DYNAMIC FRAGILITY OF NUCLEAR POWER PLANT COMPONENTS.
NUREG/CR-3178: STRUCTURAL AND TECTONIC STUDIES IN NEW YORK STATE.Final Report,July 1981 - June 1982.
NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3805 V02: ENGINEERING CHARACTERIZATION OF GROUND MOTION.Task II: Effects Of Ground Motion Characteristics On Structural Response Considering Localized Structural Nonlinearities And Soil-Structure Interaction Effects.
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.
NUREG/CR-4117: FAULTING AND JOINTING IN AND NEAR SURFACE MINES OF SOUTHWESTERN INDIANA.
NUREG/CR-4145: EARTHQUAKE RECURRENCE INTERVALS AT NUCLEAR POWER PLANTS: ANALYSIS AND RANKING.
NUREG/CR-4226: NEW MADRID SEISMOTECTONIC STUDY.Activities During Fiscal Year 1983.
NUREG/CR-4290 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF BABCOCK AND WILCOX PWR PLANTS.Volume 2:Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-4329: RELIABILITY EVALUATION OF CONTAINMENTS INCLUDING SOIL-STRUCTURE INTERACTION.
NUREG/CR-4334: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS.
NUREG/CR-4339: A REVIEW OF RECENT RESEARCH ON THE SEISMOTECTONICS OF THE SOUTHEASTERN SEABOARD AND AN EVALUATION OF HYPOTHESES ON THE SOURCE OF THE 1886 CHARLESTON,SOUTH CAROLINA EARTHQUAKE.
NUREG/CR-4354: A STUDY OF SEISMICITY AND TECTONICS IN NEW ENGLAND.Final Report.
NUREG/CR-4430: CURRENT METHODOLOGIES FOR ASSESSING THE POTENTIAL FOR EARTHQUARKE-INDUCED LIQUEFACTION IN SOILS.
NUREG/CR-4432: COMPARISON OF DYNAMIC CHARACTERISTICS OF FUKUSHIMA NUCLEAR POWER PLANT CONTAINMENT BUILDING DETERMINED FROM TESTS AND EARTHQUAKES.

Econometric Model
NUREG/CR-4373: COMPENDIUM OF COST-EFFECTIVENESS EVALUATIONS OF MODIFICATIONS FOR DOSE REDUCTION AT NUCLEAR POWER PLANTS.

Eddy Current
NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984.
NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.
NUREG-1155 V04: RESEARCH PROGRAM PLAN.Non-Destructive Examination.
NUREG/CR-3949 V01: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Semiannual Progress Report For Period Ending June 30,1984.
NUREG/CR-3949 V02: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Annual Progress Report For Period Ending December 31,1984.

Educational Qualification
NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.

Effective Peak Acceleration
NUREG/CR-3805 V02: ENGINEERING CHARACTERIZATION OF GROUND MOTION.Task II: Effects Of Ground Motion Characteristics On Structural Response Considering Localized Structural Nonlinearities And Soil-Structure Interaction Effects.

Effluent
NUREG-0017 R01: CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM PRESSURIZED WATER REACTORS (PWR-GALE CODE).

NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION.
NUREG/CR-4397: IN-PLANT SOURCE TERM MEASUREMENTS AT PRAIRIE ISLAND NUCLEAR GENERATING STATION.

Elastic Unloading
NUREG/CR-4283: STUDY OF THE EFFECTS OF ELASTIC UNLOADINGS ON THE JI-R CURVES FROM COMPACT SPECIMENS.

Elastic-Plastic Fracture Mechanics
NUREG-1155 V03: RESEARCH PROGRAM PLAN.Piping.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

Electric Motor
NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.

Electric Motor-Operated Valves
NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.

Electrical System Failure
NUREG/CR-3991: FAILURE MODES AND EFFECTS ANALYSIS (FMEA) OF THE ICS/NNI ELECTRIC POWER DISTRIBUTION CIRCUITRY AT THE OCONEE-1 NUCLEAR PLANT.

Electronic Module Aging
NUREG/CR-3863: ASSESSMENT OF CLASS 1E PRESSURE TRANSMITTER RESPONSE WHEN SUBJECTED TO HARSH ENVIRONMENT SCREENING TESTS.

Embrittlement
NUREG/CR-3228 V03: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1984.
NUREG/CR-3998 V02: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,April-June 1984.
NUREG/CR-3998 V03: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,October-December 1984.
NUREG/CR-4204: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Annual Report,October 1983 - September 1984.
NUREG/CR-4212: IN-PLACE THERMAL ANNEALING OF NUCLEAR REACTOR PRESSURE VESSELS.
NUREG/CR-4395: CORRELATION OF CV AND KIC/KJC TRANSITION TEMPERATURE INCREASES DUE TO IRRADIATION.

Emergency Core Cooling
NUREG/CR-3866: TRAC-PD2 INDEPENDENT ASSESSMENT.
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.
NUREG/CR-4252: INDEPENDENT ASSESSMENT OF TRAC-PD2/MOD1 CODE WITH BCL ECC BYPASS TESTS.

Emergency Diesel Generator
NUREG/CR-4347: EMERGENCY DIESEL GENERATOR OPERATING EXPERIENCE,1981-1983.

Emergency Operating Procedure
NUREG-0800 13.5.2 R1: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 1 To Section 13.5.2, "Operating And Maintenance Procedures," and Revision 0 of Appendix A to Section 13.5.2, "Review...."

Emergency Preparedness
NUREG-1140 DRFT FC: A REGULATORY ANALYSIS ON EMERGENCY PREPAREDNESS FOR FUEL CYCLE AND OTHER RADIOACTIVE MATERIAL LICENSEES.Draft Report For Comment.
NUREG/CR-4151: INTEGRATION OF EMERGENCY ACTION LEVELS WITH COMBUSTION ENGINEERING EMERGENCY OPERATING PROCEDURES.By Use Of Combustion Engineering Owners Group Emergency Operating Procedure Technical Guidelines.

Emergency Response
NUREG-0981 R01: NRC/FEMA OPERATIONAL RESPONSE PROCEDURES FOR RESPONSE TO A COMMERCIAL NUCLEAR REACTOR ACCIDENT.

Emergency Response Planning
NUREG/CR-3657: PRELIMINARY SCREENING OF FUEL CYCLE AND BY-PRODUCT MATERIAL LICENSES FOR EMERGENCY PLANNING.

Emergency Sump Performance
NUREG-0897 R01: CONTAINMENT EMERGENCY SUMP PERFORMANCE.(Technical Findings Related To Unresolved Safety Issues).

Emission Control
NUREG/CR-4176: EMISSION CONTROL TECHNOLOGY AND QUALITY ASSURANCE NEEDS AT URANIUM MILLING FACILITIES.Includes Supporting Methods For Testing,Operating,And Maintaining Air Pollution Control Devices.

Emissions
NUREG/CR-4100: EVALUATION OF INSTRUMENTAL METHODS FOR THE MEASUREMENT OF YELLOWCAKE EMISSIONS.

Energetic Core Disruptive Accident
NUREG/CR-4346: AEROSOL RELEASE EXPERIMENTS IN THE FUEL AEROSOL SIMULANT TEST FACILITY.UNDERSODIUM EXPERIMENTS.

Energy Transport
NUREG/CR-4203: A CALCULATIONAL METHOD FOR DETERMINING BIOLOGICAL DOSE RATES FROM IRRADIATED RESEARCH REACTOR FUEL.

Enforcement Actions
NUREG-0940 V03 N04: ENFORCEMENT ACTIONS.SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,October-December 1984.
NUREG-0940 V04 N01: ENFORCEMENT ACTIONS.SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,January-March 1985.
NUREG-0940 V04 N02: ENFORCEMENT ACTIONS.SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June,1985.
NUREG-0940 V04 N03: ENFORCEMENT ACTIONS.SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,July-September 1985.

Engineered Disposal
NUREG/CR-3774 V02: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2A:Technical Requirements For Belowground Vault Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V03: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2B:Technical Requirements For Aboveground Vault Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V04: ALTERNATIVE METHOD FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2C:Technical Requirements For Earth Mounded Concrete Bunker Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V05: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2E:Technical Requirements For Shaft Disposal Of Low Level Radioactive Waste.

Engineered Safety Feature
NUREG/CR-3317: TECHNICAL BASES AND USER'S MANUAL FOR THE PROTOTYPE OF SPARC - A SUPPRESSION POOL AEROSOL REMOVAL CODE.
NUREG/CR-4130: ICEDF.A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.
NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.

Engineered Safety System
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.
NUREG/CR-4210: MATADOR.A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS.
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.

Engineering Curriculum
NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.

Enterprise Model
NUREG/CR-4322 V01: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 1: Enterprise Model.

Entrainment
NUREG/CR-4424: DROPLET SIZES,DYNAMICS AND DEPOSITION IN VERTICAL ANNULAR FLOW.

Environment
NUREG/CR-3738: ENVIRONMENTAL EFFECTS OF THE URANIUM FUEL CYCLE.A Review Of Data For Technetium.
NUREG/CR-4138: DATA ANALYSES FOR NEVADA TEST SITE (NTS) PREMIXED COMBUSTION TESTS.

Environmental Assessment
NUREG-1118: ENVIRONMENTAL ASSESSMENT FOR RENEWAL OF SPECIAL NUCLEAR MATERIAL LICENSE NO. SNM-1107.Docket No. 70-1151. (Westinghouse Electric Corporation)
NUREG-1130: ENVIRONMENTAL ASSESSMENT FOR RENEWAL AND CONSOLIDATION OF MATERIALS LICENSE NOS. SNM-362,SMB-405,08-00566-05, 08-00566-10,AND 08-00566-12.
NUREG-1157: ENVIRONMENTAL ASSESSMENT FOR RENEWAL OF SOURCE MATERIAL LICENSE NO. SUB-1010.Docket No. 40-8027. (Sequoyah Fuels Corporation)
NUREG-1112: ENVIRONMENTAL ASSESSMENT FOR RENEWAL OF SPECIAL NUCLEAR MATERIAL LICENSE NO. SNM-368.(UNC Naval Products Division Of UNC Resources,Inc)

Environmental Consequences
NUREG/CR-4061: LEACHATE PLUME MIGRATION DOWNGRADIENT FROM URANIUM TAILINGS DISPOSAL IN MINE STOPES.

Environmental Effect
NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984.
NUREG/CR-3945: FATIGUE CRACK GROWTH RATES OF LOW-CARBON AND STAINLESS PIPING STEELS IN PWR ENVIRONMENT.

Environmental Impact
NUREG-1165: ESRP 7.1.1 "ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RELEASES OF RADIOACTIVE MATERIALS TO GROUNDWATER."

Environmental Transport
NUREG/CR-4350 V07: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 7 - Environmental Transport And Consequence Analysis.

Equation Transformation System
NUREG/CR-4213: SETS REFERENCE MANUAL.

Equipment
NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.

Equipment Hatch
NUREG/CR-4137: PRETEST PREDICTIONS FOR THE RESPONSE OF A 1:8-SCALE STEEL LWR CONTAINMENT BUILDING MODEL TO STATIC OVERPRESSURIZATION.

Equipment Qualification
NUREG/CR-3914: PUMP AND VALVE QUALIFICATION REVIEW GUIDE.
NUREG/CR-4358: APPLICATIONS OF DENSITY PROFILING TO EQUIPMENT QUALIFICATION ISSUES.

Equipment Survivability
NUREG/CR-4146: SIMULATION OF AN EPRI-NEVADA TEST SITE (NTS) HYDROGEN BURN TEST AT THE CENTRAL RECEIVER TEST FACILITY.

Event Tree
NUREG/CR-4050: A REVIEW OF THE SHOREHAM NUCLEAR POWER STATION PROBABILISTIC RISK ASSESSMENT.Internal Events And Core Damage Frequency.
NUREG/CR-4350 V04: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 4 - System Reliability And Analysis Techniques,Sessions B/C - Event Trees/Fault Trees.
NUREG/CR-4377: EVALUATIONS AND UTILIZATIONS OF RISK IMPORTANCES.

Examination
NUREG-1122: KNOWLEDGES AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS.Pressurized Water Reactors.

Examiner Standard
NUREG-1021 R01: OPERATOR LICENSING EXAMINER STANDARDS.

Exhaust Flow
NUREG/CR-4112 V01: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task A:IEEE Flame Test.

Exide EMP-13 Battery Cell
NUREG/CR-4097: TEST SERIES 4:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED EXIDE EMP-13 BATTERY CELLS.

Expert System
NUREG/CR-4272: RESPONSE TREE EVALUATION:EXPERIMENTAL ASSESSMENT OF AN EXPERT SYSTEM FOR NUCLEAR REACTOR OPERATORS.

Explosive Gas
NUREG/CR-3237: CONTROL OF EXPLOSIVE MIXTURES IN PWR WASTE GAS SYSTEMS.

Exposure
NUREG-0713 V05: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS - 1983 ANNUAL REPORT.
NUREG-1134: RADIATION PROTECTION TRAINING FOR PERSONNEL EMPLOYED IN MEDICAL FACILITIES.
NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.
NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984.
NUREG/CR-3990: CHARCOAL PERFORMANCE UNDER ACCIDENT CONDITIONS IN LIGHT-WATER REACTORS.
NUREG/CR-4031 V03: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutron Exposure Parameters."
NUREG/CR-4147: THE EFFECT OF ENVIRONMENTAL STRESS ON SYLGARD 70 SILICONE ELASTOMER.

Extended Storage
NUREG/CR-4062: EXTENDED STORAGE OF LOW-LEVEL RADIOACTIVE WASTES:Potential Problem Areas.

Extremity Dosimetry
NUREG/CR-4297: EXTREMITY MONITORING:Considerations For Use,Dosimeter Placement,And Evaluation.

FAST
NUREG/CR-3830 V02: AEROSOL RELEASE AND TRANSPORT PROGRAM.Semiannual Progress Report For April 1984-September 1984.
NUREG/CR-4346: AEROSOL RELEASE EXPERIMENTS IN THE FUEL AEROSOL SIMULANT TEST FACILITY:UNDERSODIUM EXPERIMENTS.

FEMA
NUREG-0981 R01: NRC/FEMA OPERATIONAL RESPONSE PROCEDURES FOR RESPONSE TO A COMMERCIAL NUCLEAR REACTOR ACCIDENT.

FERRET-SAND II
NUREG/CR-3319: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM:LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium.

FIRAC
NUREG/CR-4321: FULL SCALE MEASUREMENTS OF SMOKE TRANSPORT AND DEPOSITION IN VENTILATION SYSTEM DUCTWORK.

FIST
NUREG/CR-4127 V01: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 1:Numerical Methods.

FITS
NUREG/CR-3721 V01: PRESSURE MEASUREMENTS IN A HYDROGEN COMBUSTION ENVIRONMENT. Hydrogen-Air Combustion Test Series 1 And 2 In The FITS Tank.

FLECHT
NUREG/CR-4167: FLECHT SEASET PROGRAM.Final Report.NRC/EPRI Westinghouse Report Number 16.

FLECHT SEASET
NUREG/CR-4166: ANALYSIS OF FLECHT-SEASET 163-ROD BLOCKED BUNDLE DATA USING COBRA-TF.

FORTRAN
NUREG/CR-4292: A COMPARATIVE ANALYSIS OF CONSTITUTIVE RELATIONS IN TRAC-PFL AND RELAP5/MOD1.

FORTRAN 77
NUREG/CR-4122: A FORTRAN 77 PROGRAM AND USER'S GUIDE FOR THE CALCULATION OF PARTIAL CORRELATION AND STANDARDIZED REGRESSION COEFFICIENTS.

FRAC
NUREG/CR-4217: A STATISTICAL ANALYSIS OF NUCLEAR POWER PLANT VALVE FAILURE-RATE VARIABILITY--SOME PRELIMINARY RESULTS.

FRAPCON
NUREG/CR-3810 V03: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,July-September 1984.
NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,October-December 1984.
NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,January-March 1985.
NUREG/CR-4318 V02: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,April-June 1985.

FRAPCON-2
NUREG/CR-3741 V02: EVALUATION OF POWER REACTOR FUEL ROD ANALYSIS CAPABILITIES. Phase 2 Topical Report.Volume 2:Code Evaluation.

Fabrication Criteria
NUREG/CR-3854: FABRICATION CRITERIA FOR SHIPPING CONTAINERS.

Failure
NUREG-1116: A REVIEW OF THE CURRENT UNDERSTANDING OF THE POTENTIAL FOR CONTAINMENT FAILURE FROM IN-VESSEL STEAM EXPLOSIONS.
NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.
NUREG/CP-0070: PROCEEDINGS OF THE WORKSHOP ON SEISMIC AND DYNAMIC FRAGILITY OF NUCLEAR POWER PLANT COMPONENTS.
NUREG/CR-3647: DESIGN AND FABRICATION OF A 1/8-SCALE STEEL CONTAINMENT MODEL.
NUREG/CR-3660 V04: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 4:Pipe Failure Induced By Crack Growth In West Coast Plants.
NUREG/CR-3663 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 3:Double Ended Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3831: THE IN-PLANT RELIABILITY DATA BASE FOR NUCLEAR PLANT COMPONENTS.Interim Report - Diesel Generators,Batteries, Chargers And Inverters.
NUREG/CR-3943: THE BWR PLAN ANALYZER.
NUREG/CR-3952: SEQUOYAH EQUIPMENT HATCH SEAL LEAKAGE.
NUREG/CR-3991: FAILURE MODES AND EFFECTS ANALYSIS (FMEA) OF THE ICS/NNI ELECTRIC POWER DISTRIBUTION CIRCUITRY AT THE OCONEE-1 NUCLEAR PLANT.
NUREG/CR-4004: CLOSEOUT OF IE BULLETIN 79-25:FAILURES OF WESTINGHOUSE BFD RELAYS IN SAFETY-RELATED SYSTEMS.
NUREG/CR-4123: SEISMIC FRAGILITY OF REINFORCED CONCRETE STRUCTURES AND COMPONENTS FOR APPLICATION TO NUCLEAR FACILITIES.
NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.
NUREG/CR-4180: STATE-OF-THE-ART OF SOLID-STATE MOTOR CONTROLLERS.
NUREG/CR-4209: COMPARISON OF ANALYTICAL PREDICTIONS AND EXPERIMENTAL RESULTS FOR A 1:8-SCALE STEEL CONTAINMENT MODEL PRESSURIZED TO FAILURE.
NUREG/CR-4217: A STATISTICAL ANALYSIS OF NUCLEAR POWER PLANT VALVE FAILURE-RATE VARIABILITY--SOME PRELIMINARY RESULTS.
NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.
NUREG/CR-4305: COMMENTS ON THE LEAK-BEFORE-BREAK CONCEPT FOR NUCLEAR POWER PLANT PIPING SYSTEMS.
NUREG/CR-4314: BRIEF SURVEY AND COMPARISON OF COMMON CAUSE FAILURE ANALYSIS.
NUREG/CR-4326 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Main Report.
NUREG/CR-4326 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Appendices.
NUREG/CR-4383: HIGH PRESSURE INJECTION OF MELT FROM A REACTOR PRESSURE VESSEL - THE DISCHARGE PHASE.
NUREG/CR-4385: EFFECTS OF CONTROL SYSTEM FAILURES IN TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A WESTINGHOUSE PRESSURIZED WATER REACTOR.
NUREG/CR-4386: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A BABCOCK AND WILCOX PRESSURIZED WATER REACTOR.
NUREG/CR-4387: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS AND CORE-MELT FREQUENCIES AT A GENERAL ELECTRIC PRESSURIZED WATER REACTOR.
NUREG/CR-4440: A REVIEW OF EMERGENCY DIESEL GENERATOR PERFORMANCE AT NUCLEAR POWER PLANTS.

Failure Analysis
NUREG/CR-4149: ULTIMATE PRESSURE CAPACITY OF REINFORCED AND PRESTRESSED CONCRETE CONTAINMENT.

Failure Mechanism
NUREG/CR-3819: SURVEY OF AGED POWER PLANT FACILITIES.

Failure Rate Analysis Code
NUREG/CR-4217: A STATISTICAL ANALYSIS OF NUCLEAR POWER PLANT VALVE FAILURE-RATE VARIABILITY--SOME PRELIMINARY RESULTS.

Failure Rates
NUREG/CR-4228: REVIEW OF THE VOGTLE UNITS 1 AND 2 AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS.

Fast Breeder Reactor
NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.

Fast Flux Test Facility
NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.

Fastener
NUREG/CR-4112 V01: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task A:IEEE Flame Test.

Fatigue
NUREG/CR-3228 V03: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1984.

Fatigue Crack Growth
NUREG/CR-3945: FATIGUE CRACK GROWTH RATES OF LOW-CARBON AND STAINLESS PIPING STEELS IN PWR ENVIRONMENT.
NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS.

Fault
NUREG/CR-3178: STRUCTURAL AND TECTONIC STUDIES IN NEW YORK STATE.Final Report,July 1981 - June 1982.
NUREG/CR-4117: FAULTING AND JOINTING IN AND NEAR SURFACE MINES OF SOUTHWESTERN INDIANA.
NUREG/CR-4339: A REVIEW OF RECENT RESEARCH ON THE SEISMOTECTONICS OF THE SOUTHEASTERN SEABOARD AND AN EVALUATION OF HYPOTHESES ON THE SOURCE OF THE 1886 CHARLESTON,SOUTH CAROLINA EARTHQUAKE.

Fault Tree
NUREG/CR-3301: CATALOG OF PRA DOMINANT ACCIDENT SEQUENCE INFORMATION.
NUREG/CR-4213: SETS REFERENCE MANUAL.
NUREG/CR-4350 V04: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 4 - System Reliability And Analysis Techniques,Sessions B/C - Event Trees/Fault Trees.

Fault Zone
NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983.
NUREG/CR-4333: STE. GENEVIEVE FAULT ZONE,MISSOURI AND ILLINOIS.

Ferritic Steel
NUREG/CR-3744 V02: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR APRIL-SEPTEMBER 1984.
NUREG/CR-4219 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.
NUREG/CR-4275: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM - FIVE-YEAR PLAN FY 1984-1988.

Film Flow
NUREG/CR-3651: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES.

Film Rupture
NUREG/CR-4422: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN PWR ENVIRONMENTS.

Filter
NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET-GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE,AND SODIUM CHLORIDE.

Filter Flow Test
NUREG/CR-4056: PARTICULATE AND GAS RELEASE FROM LIGHT-WATER-REACTOR (LWR) FUEL RODS STORED IN INERT AND DRY AIR ATMOSPHERES.

Filtration
NUREG/CR-3537: EXPEDIENT METHODS OF RESPIRATORY PROTECTION:III. SUBMICRON PARTICLE TESTS AND SUMMARY OF QUALITY FACTORS.

Final Environmental Statement
NUREG-1033: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF WPPSS NUCLEAR PROJECT NO. 3.Docket No. 50-508.(Washington Public Power Supply System)
NUREG-1073: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF RIVER BEND STATION.Docket No. 50-458.(Gulf States Utilities And Cajun Electric Power Cooperative)
NUREG-1085: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION,UNIT NO. 2.Docket No. 50-410.(Niagara Mohawk Power Corporation,et al)
NUREG-1087: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF VOGTLE ELECTRIC GENERATING PLANT,UNITS 1 AND 2.Docket Nos. 50-424 And 50-425.(Georgia Power Company)
NUREG-1094: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF BEAVER VALLEY POWER STATION,UNIT 2.Docket No. 50-412. (Duquesne Light Company)

Financial Analysis
NUREG-1131: FINANCIAL ANALYSIS OF POTENTIAL RETROSPECTIVE PREMIUM ASSESSMENTS UNDER THE PRICE-ANDERSON SYSTEM.

Finite Element Analysis
NUREG/CR-4367: ORVIRT.PC:A 2-D FINITE-ELEMENT FRACTURE ANALYSIS PROGRAM FOR A MICROCOMPUTER.

Finite Element Method
NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS.

Fire Accident Analysis Computer Code
NUREG/CR-4321: FULL SCALE MEASUREMENTS OF SMOKE TRANSPORT AND DEPOSITION IN VENTILATION SYSTEM DUCTWORK.

Fire Exposure
NUREG/CR-4112 V02: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task B:Firestop Test Method.

Fire Protection Research
NUREG-1148: NUCLEAR POWER PLANT FIRE PROTECTION RESEARCH PROGRAM.

Fire Protection System
NUREG/CR-4230: PROBABILITY-BASED EVALUATION OF SELECTED FIRE PROTECTION FEATURES IN NUCLEAR POWER PLANTS.

Fire Risk
NUREG/CR-4229: EVALUATION OF CURRENT METHODOLOGY EMPLOYED IN PROBABILISTIC RISK ASSESSMENT (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.
NUREG/CR-4231: EVALUATION OF AVAILABLE DATA FOR PROBABILISTIC RISK ASSESSMENTS (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.

Firestop
NUREG/CR-4112 V02: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task B:Firestop Test Method.

Fish
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.

Fission
NUREG/CR-3930: OBSERVED BEHAVIOR OF CESIUM,IODINE,AND TELLURIUM IN THE ORNL FISSION PRODUCT RELEASE PROGRAM.

Fission Gas Release
NUREG/CR-4056: PARTICULATE AND GAS RELEASE FROM LIGHT-WATER-REACTOR (LWR) FUEL RODS STORED IN INERT AND DRY AIR ATMOSPHERES.

Fission Product
NUREG/CR-3197 V01: REACTION BETWEEN SOME CESIUM-IODINE COMPOUNDS AND THE REACTOR MATERIALS 304 STAINLESS STEEL,INCONEL 600 & SILVER. Volume I:Cesium Hydroxide Reactions.
NUREG/CR-3980 V03: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,July-September 1984.
NUREG/CR-3980 V04: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,October-December 1984.
NUREG/CR-4037: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-5.
NUREG/CR-4081: ABSORPTION OF GASEOUS IODINE BY WATER DROPLETS.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0:A Computer Code for Severe Reactor Accident Containment Analysis.
NUREG/CR-4105: AN ASSESSMENT OF THERMAL GRADIENT TUBE RESULTS FROM THE HI SERIES OF FISSION PRODUCT RELEASE TESTS.
NUREG/CR-4169: AN APPROACH TO TREATING RADIONUCLIDE DECAY HEATING FOR USE IN THE MELCOR CODE SYSTEM.
NUREG/CR-4172: A USER'S GUIDE FOR MERGE.

Fission Product Model
NUREG/CR-3980 V02: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,April-June 1984.

Fission Product Release
NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1-September 30,1984.
NUREG/CR-4402 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31,1985.

Fission Product Retention
NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.

Fission Product Transport
NUREG/CR-4205: TRAP-MELT2 USER'S MANUAL.

Five-Year Research Plan
NUREG-1080 V02: LONG-RANGE RESEARCH PLAN FY 1986-FY 1990.

Flame Test
NUREG/CR-4112 V01: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task A:IEEE Flame Test.

Flashing
NUREG/CR-3937: STEAM GENERATOR TUBE RUPTURE IODINE TRANSPORT MECHANISMS. Task 1:Experimental Studies.
NUREG/CR-4079: ANALYTIC STUDIES PERTAINING TO STEAM GENERATOR TUBE RUPTURE ACCIDENTS.

Flaw
NUREG/CR-3723: STRESS-INTENSITY FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS.
NUREG/CR-3744 V02: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR APRIL-SEPTEMBER 1984.
NUREG/CR-3825 V03-4: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Quarterly Report, April 1984 - September 1984.Volumes 3 and 4.
NUREG/CR-3915: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL.
NUREG/CR-4219 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.
NUREG/CR-4275: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM - FIVE-YEAR PLAN FY 1984-1988.

NUREG/CR-4284: NEUTRON EXPOSURE PARAMETERS FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY IRRADIATION SERIES.
NUREG/CR-4304: PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO THE PWR THERMAL-SHOCK ISSUE.Experiment TSE-7.
NUREG/CR-4325: A PARAMETRIC STUDY OF PWR PRESSURE VESSEL INTEGRITY DURING OVERCOOLING ACCIDENTS,CONSIDERING BOTH 2-D AND 3-D FLAWS.
NUREG/CR-4367: ORVIRT.PC:A 2-D FINITE-ELEMENT FRACTURE ANALYSIS PROGRAM FOR A MICROCOMPUTER.

Flaw Detection
NUREG/CR-4300 V01: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Progress Report, October-March 1985.

Flawed Steel Plate
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.

Flexible Piping Design
NUREG/CR-4263: RELIABILITY ANALYSIS OF STIFF VERSUS FLEXIBLE PIPING FINAL PROJECT REPORT.

Flow Blockage Heat Transfer
NUREG/CR-4166: ANALYSIS OF FLECHT-SEASET 163-ROD BLOCKED BUNDLE DATA USING COBRA-TF.
NUREG/CR-4167: FLECHT SEASET PROGRAM.Final Report.NRC/EPRI Westinghouse Report Number 16.

Flow Rate
NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS.

Flow Redistribution
NUREG/CR-3498: TWO-DIMENSIONAL MODELING OF INTRA-SUBASSEMBLY HEAT TRANSFER AND BUOYANCY-INDUCED FLOW REDISTRIBUTION IN LMFBRS.

Focal Mechanism
NUREG/CR-3145 V03: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION - ANNUAL REPORT.(October 1982 - September 1983, Volume 3).
NUREG/CR-4288: FOCAL MECHANISM ANALYSES FOR VIRGINIA AND EASTERN TENNESSEE EARTHQUAKES (1978-1984).

Force-Balance Pressure Transmitters
NUREG/CR-4256: MEASUREMENT OF RESPONSE TIME AND DETECTION OF DEGRADATION IN PRESSURE SENSOR/SENSING LINE SYSTEMS.

Forced Convective
NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT.

Foreign Experience
NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE.

Fractography
NUREG/CR-3945: FATIGUE CRACK GROWTH RATES OF LOW-CARBON AND STAINLESS PIPING STEELS IN PWR ENVIRONMENT.
NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS.

Fracture
NUREG/CR-3178: STRUCTURAL AND TECTONIC STUDIES IN NEW YORK STATE.Final Report,July 1981 - June 1982.
NUREG/CR-3228 V03: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1984.
NUREG/CR-4042: A 3-DIMENSIONAL COMPUTER MODEL TO SIMULATE FLUID FLOW AND CONTAINMENT TRANSPORT THROUGH A ROCK FRACTURE SYSTEM.
NUREG/CR-4198: FRACTURE IN GLASS/HIGH LEVEL WASTE CANISTERS.

Fracture Analysis
NUREG/CR-4367: ORVIRT.PC:A 2-D FINITE-ELEMENT FRACTURE ANALYSIS PROGRAM FOR A MICROCOMPUTER.

Fracture Mechanics
NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984.
NUREG-1155 V01: RESEARCH PROGRAM PLAN.Reactor Vessels.
NUREG/CR-3723: STRESS-INTENSITY-FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS.
NUREG/CR-3744 V02: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR APRIL-SEPTEMBER 1984.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.
NUREG/CR-4219 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.
NUREG/CR-4249: PRESSURE VESSEL FRACTURE STUDIES PENETRATING TO THE PWR THERMAL-SHOCK ISSUE.EXPERIMENTS TSE-5,TSE-5A AND TSE-6.
NUREG/CR-4275: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM - FIVE-YEAR PLAN FY 1984-1988.
NUREG/CR-4304: PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO THE PWR THERMAL-SHOCK ISSUE.Experiment TSE-7.
NUREG/CR-4305: COMMENTS ON THE LEAK-BEFORE-BREAK CONCEPT FOR NUCLEAR POWER PLANT PIPING SYSTEMS.
NUREG/CR-4325: A PARAMETRIC STUDY OF PWR PRESSURE VESSEL INTEGRITY DURING OVERCOOLING ACCIDENTS,CONSIDERING BOTH 2-D AND 3-D FLAWS.

Fracture Toughness
NUREG/CR-3978: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL PLATE AND WELDS FOR THE FOURTH HSST IRRADIATION SERIES.
NUREG/CR-4395: CORRELATION OF CV AND KIC/KJC TRANSITION TEMPERATURE INCREASES DUE TO IRRADIATION.

Fragility
NUREG/CR-3558: HANDBOOK OF NUCLEAR POWER PLANT SEISMIC FRAGILITIES. Seismic Safety Margins Research Program.
NUREG/CR-4123: SEISMIC FRAGILITY OF REINFORCED CONCRETE STRUCTURES AND COMPONENTS FOR APPLICATION TO NUCLEAR FACILITIES.

Free Convection
NUREG/CR-3498: TWO-DIMENSIONAL MODELING OF INTRA-SUBASSEMBLY HEAT TRANSFER AND BUOYANCY-INDUCED FLOW REDISTRIBUTION IN LMFBRS.

Froude Number
NUREG/CR-3426 V01: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY. Facility And Test Design Report.
NUREG/CR-3426 V02: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY.Data Report.

Fuel
NUREG-1127: RADIATION PROTECTION TRAINING AT URANIUM HEXAFLUORIDE AND FUEL FABRICATION PLANTS.
NUREG/CR-3944: TRAN B-3:EXPERIMENTAL INVESTIGATION OF FUEL CRUST STABILITY ON MELTING SURFACES OF AN ANNULAR FLOW CHANNEL.

Fuel Aerosol Simulant Test
NUREG/CR-3830 V02: AEROSOL RELEASE AND TRANSPORT PROGRAM.Semiannual Progress Report For April 1984-September 1984.
NUREG/CR-4348: AEROSOL RELEASE EXPERIMENTS IN THE FUEL AEROSOL SIMULANT TEST FACILITY:UNDERSODIUM EXPERIMENTS.

Fuel Cladding
NUREG/CR-3900 V04: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Annual Report,April 1984 - April 1985.
NUREG/CR-4151: INTEGRATION OF EMERGENCY ACTION LEVELS WITH COMBUSTION ENGINEERING EMERGENCY OPERATING PROCEDURES.By Use Of Combustion Engineering Owners Group Emergency Operating Procedure Technical Guidelines.

Fuel Cycle
NUREG/CR-3293 V01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Main Report.
NUREG/CR-3293 V02: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Appendices.
NUREG/CR-4232: THE RESPONSE OF VENTILATION DAMPERS TO LARGE AIRFLOW PULSES.

Fuel Cycle Facility
NUREG-0525 R10: SAFEGUARDS SUMMARY EVENT LIST (SSEL),REVISION 10.

Fuel Damage
NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,October-December 1984.
NUREG/CR-3816 V02: REACTOR SAFETY RESEARCH.Quarterly Report,April-June 1984.
NUREG/CR-4037: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-5.

Fuel Failure
NUREG/CR-4168: GT2F:A COMPUTER CODE FOR ESTIMATING LIGHT WATER REACTOR FUEL ROD FAILURES.

Fuel Flow
NUREG/CR-4112 V01: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task A:IEEE Flame Test.

Fuel Performance
NUREG/CR-3950 V01: FUEL PERFORMANCE ANNUAL REPORT FOR 1983.

Fuel Removal
NUREG/CR-3757: TRAN B-2:THE EFFECT OF LOW STEEL CONTENT ON FUEL PENETRATION IN A NON-MELTING CYLINDRICAL FLOW CHANNEL.

Fuel Rod
NUREG/CR-3741 V02: EVALUATION OF POWER REACTOR FUEL ROD ANALYSIS CAPABILITIES. Phase 2 Topical Report.Volume 2:Code Evaluation.
NUREG/CR-3948: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY.
NUREG/CR-4037: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-5.
NUREG/CR-4056: PARTICULATE AND GAS RELEASE FROM LIGHT-WATER-REACTOR (LWR) FUEL RODS STORED IN INERT AND DRY AIR ATMOSPHERES.

Fuel Rod Behavior
NUREG/CR-4345: INVESTIGATION OF THE STABILITY OF LWR SPENT FUEL RODS BELOW 250 C.

Fuel Rod Cladding
NUREG/CR-3999: ELECTRICALLY HEATED EX-REACTOR PELLET-CLADDING INTERACTION (PCI) SIMULATIONS UTILIZING IRRADIATED ZIRCALOY CLADDING.

Fugitive Dust
NUREG/CR-4089: EVALUATION OF FIELD-TESTED FUGITIVE DUST CONTROL TECHNIQUES FOR URANIUM MILL TAILINGS PILES.

Fukushima Data
NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS.

Full Integral Simulation Test
NUREG/CR-4127 V01: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 1:Numerical Methods.

Fully Instrumented Test Site
NUREG/CR-3721 V01: PRESSURE MEASUREMENTS IN A HYDROGEN COMBUSTION ENVIRONMENT. Hydrogen-Air Combustion Test Series 1 And 2 In The FITS Tank.

GE Type AK-2 Circuit
NUREG/CR-3791: CLOSEOUT OF IE BULLETIN 79-09:FAILURE OF GE TYPE AK-2 CIRCUIT BREAKERS IN SAFETY-RELATED SYSTEMS.

GS2
NUREG/CR-3901: DOCUMENTATION AND USER'S GUIDE:GS2 & GS3 - VARIABLY SATURATED FLOW AND MASS TRANSPORT MODELS.

GS3
NUREG/CR-3901: DOCUMENTATION AND USER'S GUIDE:GS2 & GS3 - VARIABLY SATURATED FLOW AND MASS TRANSPORT MODELS.

GT2F
NUREG/CR-4168: GT2F:A COMPUTER CODE FOR ESTIMATING LIGHT WATER REACTOR FUEL ROD FAILURES.

Galvanized Steel Corrosion
NUREG/CR-3361: THE EFFECT OF WATER CHEMISTRY ON THE RATES OF HYDROGEN GENERATION FROM GALVANIZED STEEL CORROSION AT POST-LOCA CONDITIONS.

Gamma Dose Rate
NUREG/CR-4203: A CALCULATIONAL METHOD FOR DETERMINING BIOLOGICAL DOSE RATES FROM IRRADIATED RESEARCH REACTOR FUEL.

Gamma-Ray
NUREG/CR-4039: GAMMA-RAY CHARACTERIZATION OF THE TWO-YEAR IRRADIATION EXPERIMENT PERFORMED AT THE POOLSIDE FACILITY.

Gas Analyzer
NUREG/CR-3237: CONTROL OF EXPLOSIVE MIXTURES IN PWR WASTE GAS SYSTEMS.

Gas Blowthrough
NUREG/CR-4383: HIGH PRESSURE INJECTION OF MELT FROM A REACTOR PRESSURE VESSEL - THE DISCHARGE PHASE.

Gas Bubble
NUREG/CR-2718: STEAM EXPLOSION EXPERIMENTS WITH SINGLE DROPS OF IRON OXIDE MELTED WITH A CO2 LASER.Part II:Parametric Studies.

Gas Distribution System
NUREG/CR-3551: SAFETY IMPLICATIONS ASSOCIATED WITH IN-PLANT PRESSURIZED GAS STORAGE AND DISTRIBUTION SYSTEMS IN NUCLEAR POWER PLANTS.

Gaseous Tracer
NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.

Gastrointestinal Absorption
NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS.Application To Establishing Values Of f1 For Soluble Plutonium.

Gaussian Plume Model
NUREG/CR-4159: COMPARISON OF THE 1981 INEL DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS.

Geiger-Mueller
NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.

Generic Issue A-29
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP):A Methodology For Estimating Risk Impacts Of Safeguards Measures.

Generic Safety issue
NUREG-0933 S03: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

Generic Safety Issues
NUREG-0933 S02: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG/CR-4103: USES OF HUMAN RELIABILITY ANALYSIS PROBABILISTIC RISK ASSESSMENT RESULTS TO RESOLVE PERSONNEL PERFORMANCE ISSUES THAT COULD AFFECT SAFETY.

Genetic Effects
NUREG/CR-4214: HEALTH EFFECTS MODEL FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Part I:Introduction,Integration & Summary.Part II:Scientific Basis For Health Effects Models.

Geochemical Condition
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.

Geochemical Modeling
NUREG/CP-0062: PROCEEDINGS OF THE CONFERENCE ON THE APPLICATION OF GEOCHEMICAL MODELS TO HIGH-LEVEL NUCLEAR WASTE REPOSITORY ASSESSMENT.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.

Geochemistry
NUREG-1164: INFORMATION ON THE CONFINEMENT CAPABILITY OF THE FACILITY DISPOSAL AREA AT WEST VALLEY,NEW YORK.
NUREG/CR-4110: REPOSITORY SITE DATA REPORT FOR UNSATURATED TUFF,YUCCA MOUNTAIN,NEVADA.
NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.

Geomorphic Process
NUREG-1164: INFORMATION ON THE CONFINEMENT CAPABILITY OF THE FACILITY DISPOSAL AREA AT WEST VALLEY,NEW YORK.

Geophysical Interpretation
NUREG/CR-3736: FIELD AND THEORETICAL INVESTIGATIONS OF FRACTURED CRYSTALLINE ROCK NEAR ORACLE,ARIZONA.

Glacial Hydrogeology
NUREG-1164: INFORMATION ON THE CONFINEMENT CAPABILITY OF THE FACILITY DISPOSAL AREA AT WEST VALLEY,NEW YORK.

Glass Waste Form
NUREG/CR-4198: FRACTURE IN GLASS/HIGH LEVEL WASTE CANISTERS.

Granite
NUREG/CR-2663 V01: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Main Report.
NUREG/CR-2663 V02: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Appendices.

Gravitational Sedimentation
NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING.

Gravity
NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983.

Ground Motion
NUREG/CR-3805 V02: ENGINEERING CHARACTERIZATION OF GROUND MOTION.Task II: Effects Of Ground Motion Characteristics On Structural Response Considering Localized Structural Nonlinearities And Soil-Structure Interaction Effects.

Groundwater
NUREG-1046: DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN UNSATURATED ZONE.TECHNICAL CONSIDERATIONS AND RESPONSE TO COMMENTS.
NUREG-1164: INFORMATION ON THE CONFINEMENT CAPABILITY OF THE FACILITY DISPOSAL AREA AT WEST VALLEY,NEW YORK.
NUREG-1165: ESRP 7.1.1 "ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RELEASES OF RADIOACTIVE MATERIALS TO GROUNDWATER."
NUREG/CR-3709: METHODS OF MINIMIZING GROUND-WATER CONTAMINATION FROM IN SITU LEACH URANIUM MINING.Final Report.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER.(Final Report)
NUREG/CR-4110: REPOSITORY SITE DATA REPORT FOR UNSATURATED TUFF,YUCCA MOUNTAIN,NEVADA.
NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.
NUREG/CR-4198: FRACTURE IN GLASS/HIGH LEVEL WASTE CANISTERS.
NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions.
NUREG/CR-4251 V02: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 2:Case Study Analysis Of Hydrologic Characterization And Mitigative Schemes.
NUREG/CR-4403: SUMMARY OF THE WASTE MANAGEMENT PROGRAMS AT URANIUM RECOVERY FACILITIES AS THEY RELATE TO THE 40 CFR PART 192 STANDARDS.

Guide
NUREG-1021 R01: OPERATOR LICENSING EXAMINER STANDARDS.
NUREG/CR-3914: PUMP AND VALVE QUALIFICATION REVIEW GUIDE.

Guidelines
NUREG/CR-2800 S03: GUIDELINES FOR NUCLEAR POWER PLANT SAFETY ISSUE PRIORITIZATION INFORMATION DEVELOPMENT.
NUREG/CR-4125 V01: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 1:Guidelines For Utility Organization And Administration Plan.
NUREG/CR-4125 V02: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 2:Workbook For Assessment Of Organization And Management.
NUREG/CR-4151: INTEGRATION OF EMERGENCY ACTION LEVELS WITH COMBUSTION ENGINEERING EMERGENCY OPERATING PROCEDURES.By Use Of Combustion Engineering Owners Group Emergency Operating Procedure Technical Guidelines.
NUREG/CR-4227: HUMAN ENGINEERING GUIDELINES FOR THE EVALUATION AND ASSESSMENT OF VIDEO DISPLAY UNITS.

HECTR
NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT.
NUREG/CR-3913: HECTR VERSION 1.0 USER'S MANUAL.
NUREG/CR-3954: HECTR ANALYSIS OF EQUIPMENT TEMPERATURE RESPONSES TO SELECTED HYDROGEN BURNS IN AN ICE CONDENSER CONTAINMENT.

HEPA
NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET-GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE,AND SODIUM CHLORIDE.
NUREG/CR-4225: SUMMARY OF EFFICIENCY TESTING OF STANDARD AND HIGH-CAPACITY HIGH-EFFICIENCY PARTICULATE AIR FILTERS SUBJECTED TO SIMULATED TORNADO DEPRESSURIZATION AND EXPLOSIVE SHOCK WAVES.
NUREG/CR-4264: INVESTIGATION ON HIGH-EFFICIENCY PARTICULATE AIR FILTER PLUGGING BY COMBUSTION AEROSOLS.

HMS
NUREG/CR-4020: HMS:A COMPUTER PROGRAM FOR TRANSIENT,THREE-DIMENSIONAL MIXING GASES.

HPI
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE 1 PRESSURIZED THERMAL SHOCK STUDY.

HPI Jet
NUREG/CR-3426 V01: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY. Facility And Test Design Report.
NUREG/CR-3426 V02: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY.Data Report.

HRA
NUREG/CR-4010: SPECIFICATION OF A HUMAN RELIABILITY DATA BANK FOR CONDUCTING HRA SEGMENTS OF PRAS FOR NUCLEAR POWER PLANTS.

HSST
NUREG/CR-3978: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL PLATE AND WELDS FOR THE FOURTH HSST IRRADIATION SERIES.
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.

NUREG/CR-4086: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL WELDS FOR THE THIRD HSST IRRADIATION SERIES.
NUREG/CR-4284: NEUTRON EXPOSURE PARAMETERS FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY IRRADIATION SERIES.

HTGR
NUREG-1125 V05: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 5,Part 2:ACRS Reports On Generic Subjects (HTGR - Regulatory Guides).
NUREG/CR-3885 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, October 1-December 31,1984.

Handbook
NUREG-0544 R02: A HANDBOOK OF ACRONYMS AND INITIALISMS.
NUREG/CR-3558: HANDBOOK OF NUCLEAR POWER PLANT SEISMIC FRAGILITIES. Seismic Safety Margins Research Program.
NUREG/CR-4041: SYSTEM ANALYSIS HANDBOOK.
NUREG/CR-4041 R01: SYSTEM ANALYSIS HANDBOOK.

Harsh Environment
NUREG/CR-3863: ASSESSMENT OF CLASS 1E PRESSURE TRANSMITTER RESPONSE WHEN SUBJECTED TO HARSH ENVIRONMENT SCREENING TESTS.

Hatch Seal
NUREG/CR-3952: SEQUOYAH EQUIPMENT HATCH SEAL LEAKAGE.

Hazardous Waste Regulation
NUREG/CR-4406: AN ANALYSIS OF LOW-LEVEL WASTES:Review of Hazardous Waste Regulations And Identification of Radioactive Mixed Wastes. Final Report.

Health
NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984.

Health Effects Model
NUREG/CR-4214: HEALTH EFFECTS MODEL FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Part I:Introduction,Integration & Summary.Part II:Scientific Basis For Health Effects Models.

Health Physics
NUREG/CR-4239: ANALYSIS OF THE ABILITY OF CURRENT HEALTH PHYSICS INSTRUMENTS TO PREDICT DOSE IN EXPOSED INDIVIDUALS.
NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.

Heat Affected Zone
NUREG/CR-3613 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Annual Report For 1984.

Heat Balance
NUREG-1167: TPDWR2:THERMAL POWER DETERMINATION FOR WESTINGHOUSE REACTORS,VERSION2.User's Guide.

Heat Removal
NUREG-0800 06.2.2 R4: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 4 To Section 6.2.2, "Containment Heat Removal System."
NUREG/CR-2951: THE D9 EXPERIMENT.Heat Removal From Stratified UO2 Debris.
NUREG/CR-3816 V02: REACTOR SAFETY RESEARCH.Quarterly Report,April-June 1984.

Heat Transfer
NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT.
NUREG/CR-3208: TRAC-PD2 DEVELOPMENTAL ASSESSMENT.
NUREG/CR-3498: TWO-DIMENSIONAL MODELING OF INTRA-SUBASSEMBLY HEAT TRANSFER AND BUOYANCY-INDUCED FLOW REDISTRIBUTION IN LMFBRS.
NUREG/CR-3633 V04: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.Volume 4: Developmental Assessment.
NUREG/CR-4376: HEAT TRANSFER,CARRYOVER AND FALL BACK IN PWR STEAM GENERATORS DURING TRANSIENTS.
NUREG/CR-4414: DIRECT-CONTACT CONDENSATION OF STEAM ON COLD WATER IN STRATIFIED COUNTERCURRENT FLOW.
NUREG/CR-4417: LOCAL PROPERTIES OF COUNTERCURRENT STRATIFIED STEAM-WATER FLOW.

Heat-Affected Zone
NUREG/CR-3911 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Quarterly Report,April-June 1984.

Heavy Section Steel Technology
NUREG/CR-3978: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL PLATE AND WELDS FOR THE FOURTH HSST IRRADIATION SERIES.
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.
NUREG/CR-4086: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL WELDS FOR THE THIRD HSST IRRADIATION SERIES.
NUREG/CR-4284: NEUTRON EXPOSURE PARAMETERS FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY IRRADIATION SERIES.

High Confidence Of Low Probability Of Failure
NUREG/CR-4334: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS.

High Efficiency Particulate Air Filter
NUREG/CR-4225: SUMMARY OF EFFICIENCY TESTING OF STANDARD AND HIGH-CAPACITY HIGH-EFFICIENCY PARTICULATE AIR FILTERS SUBJECTED TO SIMULATED TORNADO DEPRESSURIZATION AND EXPLOSIVE SHOCK WAVES.

High Integrity Container
NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA.

High Pressure Ejection
NUREG/CR-4383: HIGH PRESSURE INJECTION OF MELT FROM A REACTOR PRESSURE VESSEL - THE DISCHARGE PHASE.

High Pressure Injection
NUREG/CR-3426 V02: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY.Data Report.
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.

High Temperature
NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.
NUREG/CR-3197 V01: REACTION BETWEEN SOME CESIUM-IODINE COMPOUNDS AND THE REACTOR MATERIALS 304 STAINLESS STEEL,INCONEL 600 & SILVER: Volume 1:Cesium Hydroxide Reactions.
NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1-September 30,1984.

High Temperature Combustion
NUREG/CR-3638: HYDROGEN-STEAM JET-FLAME FACILITY AND EXPERIMENTS.

High Temperature Graphite Reactor
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.

High-Burnup Fuel
NUREG/CR-3950 V01: FUEL PERFORMANCE ANNUAL REPORT FOR 1983.

High-Dose Job
NUREG/CR-4254: OCCUPATIONAL DOSE REDUCTION AND ALARA AT NUCLEAR POWER PLANTS.Study On High-Dose Jobs,Radwaste Handling,And ALARA Incentives.

High-Efficiency Particulate
NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET-GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE AND SODIUM CHLORIDE.
NUREG/CR-4264: INVESTIGATION ON HIGH-EFFICIENCY PARTICULATE AIR FILTER PLUGGING BY COMBUSTION AEROSOLS.

High-Level Waste
NUREG-0946: AN EVALUATION OF RADIONUCLIDE CONCENTRATIONS IN HIGH-LEVEL RADIOACTIVE WASTES.
NUREG-1046: DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN UNSATURATED Z0NE:TECHNICAL CONSIDERATIONS AND RESPONSE TO COMMENTS.
NUREG/CP-0062: PROCEEDINGS OF THE CONFERENCE ON THE APPLICATION OF GEOCHEMICAL MODELS TO HIGH-LEVEL NUCLEAR WASTE REPOSITORY ASSESSMENT.
NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-3900 V02: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,July-September 1984.
NUREG/CR-3900 V03: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,October-December 1984.
NUREG/CR-3900 V04: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Annual Report,April 1984 - April 1985.
NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.
NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt.
NUREG/CR-4198: FRACTURE IN GLASS/HIGH LEVEL WASTE CANISTERS.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.REPORT FOR OCTOBER-DECEMBER 1984.
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.
NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.
NUREG/CR-4303: HIGH-LEVEL WASTE PRECLOSURE SYSTEMS SAFETY ANALYSIS.Phase 1: Final Report.
NUREG/CR-4379 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report,Year Four April-June 1985.

High-Speed Simulation
NUREG/CR-3943: THE BWR PLAN ANALYZER.

High-Temperature
NUREG/CR-4037: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-5.

High-Temperature Gas-Cooled Reactor
NUREG/CR-3885 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, October 1-December 31,1984.
NUREG/CR-4402 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31,1985.

Hole Ablation
NUREG/CR-4383: HIGH PRESSURE INJECTION OF MELT FROM A REACTOR PRESSURE VESSEL - THE DISCHARGE PHASE.

Human Engineering
NUREG/CR-4227: HUMAN ENGINEERING GUIDELINES FOR THE EVALUATION AND ASSESSMENT OF VIDEO DISPLAY UNITS.

Human Error
NUREG/CR-3519: HUMAN ERROR PROBABILITY ESTIMATION USING LICENSEE EVENT REPORTS.
NUREG/CR-3688 V01: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 1:Main Report.
NUREG/CR-3688 V02: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 2:Appendices.
NUREG/CR-4016 V01: APPLICATION OF SLIM-MAUD:A TEST OF AN INTERACTIVE COMPUTER-BASED METHOD FOR ORGANIZING EXPERT ASSESSMENT OF HUMAN PERFORMANCE AND RELIABILITY.Volume I:Main Report.
NUREG/CR-4133: NUCLEAR POWER SAFETY REPORTING SYSTEM IMPLEMENTATION AND OPERATIONAL SPECIFICATIONS.
NUREG/CR-4280: THE EFFECTS OF SUPERVISOR EXPERIENCE AND ASSISTANCE OF A SHIFT TECHNICAL ADVISOR (STA) ON CREW PERFORMANCE IN CONTROL ROOM SIMULATORS.

Human Factors
NUREG-0800 18.2 R0: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition Revision 0 To SRP Section 18.2, "Safety Parameter Display System (SPDS)."
NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984.
NUREG/CR-2331 V04 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July 1 -September 30,1984.
NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.
NUREG/CR-3481 V02: NUCLEAR POWER PLANT PERSONNEL QUALIFICATIONS AND TRAINING: TAPS -- The Task Analysis Profiling System.
NUREG/CR-3634: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: USER'S MANUAL.
NUREG/CR-3837: MULTIPLE-SEQUENTIAL FAILURE MODEL.Evaluation Of And Procedures For Human Error Dependency.
NUREG/CR-3887: HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS.
NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS.
NUREG/CR-4016 V01: APPLICATION OF SLIM-MAUD:A TEST OF AN INTERACTIVE COMPUTER-BASED METHOD FOR ORGANIZING EXPERT ASSESSMENT OF HUMAN PERFORMANCE AND RELIABILITY.Volume I:Main Report.
NUREG/CR-4093: SAFETY/SAFEGUARDS INTERACTIONS DURING SAFETY-RELATED EMERGENCIES AT NUCLEAR POWER REACTOR FACILITIES.
NUREG/CR-4103: USES OF HUMAN RELIABILITY ANALYSIS PROBABILISTIC RISK ASSESSMENT RESULTS TO RESOLVE PERSONNEL PERFORMANCE ISSUES THAT COULD AFFECT SAFETY.
NUREG/CR-4104: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL. Field Evaluation/Validation.
NUREG/CR-4206: A SELECT REVIEW OF THE RECENT (1979-1983) BEHAVIORAL RESEARCH LITERATURE ON TRAINING SIMULATORS.
NUREG/CR-4227: HUMAN ENGINEERING GUIDELINES FOR THE EVALUATION AND ASSESSMENT OF VIDEO DISPLAY UNITS.
NUREG/CR-4248: RECOMMENDATIONS FOR NRC POLICY ON SHIFT SCHEDULING AND OVERTIME AT NUCLEAR POWER PLANTS.
NUREG/CR-4272: RESPONSE TREE EVALUATION:EXPERIMENTAL ASSESSMENT OF AN EXPERT SYSTEM FOR NUCLEAR REACTOR OPERATORS.

Human Performance Data
NUREG/CR-4010: SPECIFICATION OF A HUMAN RELIABILITY DATA BANK FOR CONDUCTING HRA SEGMENTS OF PRAS FOR NUCLEAR POWER PLANTS.

Human Reliability
NUREG/CR-3519: HUMAN ERROR PROBABILITY ESTIMATION USING LICENSEE EVENT REPORTS.
NUREG/CR-3626 V02: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: DESCRIPTION OF MODEL CONTENT,STRUCTURE,AND SENSITIVITY TESTING.
NUREG/CR-4009: HUMAN RELIABILITY DATA BANK.Evaluation Results.
NUREG/CR-4010: SPECIFICATION OF A HUMAN RELIABILITY DATA BANK FOR CONDUCTING HRA SEGMENTS OF PRAS FOR NUCLEAR POWER PLANTS.

Human Reliability Analysis
NUREG/CR-4103: USES OF HUMAN RELIABILITY ANALYSIS PROBABILISTIC RISK ASSESSMENT RESULTS TO RESOLVE PERSONNEL PERFORMANCE ISSUES THAT COULD AFFECT SAFETY.

Human Reliability Estimate
NUREG/CR-3688 V01: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 1:Main Report.
NUREG/CR-3688 V02: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 2:Appendices.

Hydrogen
NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT.
NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS.

Hydrogen Absorption
NUREG/CR-4422: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN PWR ENVIRONMENTS.

Hydrogen Burn
NUREG/CR-3954: HECTR ANALYSIS OF EQUIPMENT TEMPERATURE RESPONSES TO SELECTED HYDROGEN BURNS IN AN ICE CONDENSER CONTAINMENT.
NUREG/CR-4146: SIMULATION OF AN EPRI-NEVADA TEST SITE (NTS) HYDROGEN BURN TEST AT THE CENTRAL RECEIVER TEST FACILITY.

Hydrogen Combustion
NUREG/CR-3721 V01: PRESSURE MEASUREMENTS IN A HYDROGEN COMBUSTION ENVIRONMENT. Hydrogen-Air Combustion Test Series 1 And 2 In The FITS Tank.

Hydrogen Generation
NUREG/CR-3361: THE EFFECT OF WATER CHEMISTRY ON THE RATES OF HYDROGEN GENERATION FROM GALVANIZED STEEL CORROSION AT POST-LOCA CONDITIONS.
NUREG/CR-3803: THE EFFECTS OF POST-LOCA CONDITIONS ON A PROTECTIVE COATING (PAINT) FOR THE NUCLEAR POWER INDUSTRY.

Hydrogen Transport
NUREG/CR-4020: HMS:A COMPUTER PROGRAM FOR TRANSIENT,THREE-DIMENSIONAL MIXING GASES.

Hydrogen-Oxygen Flammability
NUREG/CR-3237: CONTROL OF EXPLOSIVE MIXTURES IN PWR WASTE GAS SYSTEMS.

Hydrogeologic Site Characterization
NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions.
NUREG/CR-4251 V02: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 2:Case Study Analysis Of Hydrologic Characterization And Mitigative Schemes.

Hydrology
NUREG/CR-3736: FIELD AND THEORETICAL INVESTIGATIONS OF FRACTURED CRYSTALLINE ROCK NEAR ORACLE,ARIZONA.

ICEDF
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).
NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.

IE Bulletin 79-04
NUREG/CR-4003: CLOSEOUT OF IE BULLETIN 79-04:INCORRECT WEIGHTS FOR SWING CHECK VALVES MANUFACTURED BY VELAN ENGINEERING CORPORATION.

IE Bulletin 79-09
NUREG/CR-3791: CLOSEOUT OF IE BULLETIN 79-09:FAILURE OF GE TYPE AK-2 CIRCUIT BREAKERS IN SAFETY-RELATED SYSTEMS.

IE Bulletin 79-12
NUREG-0905: CLOSEOUT OF IE BULLETIN 79-12:SHORT-PERIOD SCRAMS AT BOILING-WATER REACTORS.

IE Bulletin 79-25
NUREG/CR-4004: CLOSEOUT OF IE BULLETIN 79-25:FAILURES OF WESTINGHOUSE BFD RELAYS IN SAFETY-RELATED SYSTEMS.

IE Bulletin 80-12
NUREG/CR-4005: CLOSEOUT OF IE BULLETIN 80-12:DECAY HEAT REMOVAL SYSTEM OPERABILITY.

IE Bulletin 80-25
NUREG/CR-3794: CLOSEOUT OF IE BULLETIN 80-25:OPERATING PROBLEMS WITH TARGET ROCK SAFETY-RELIEF VALVES AT BWRS.

IE Bulletin 81-01
NUREG/CR-4006: CLOSEOUT OF IE BULLETIN 81-01:SURVEILLANCE OF MECHANICAL SNUBBERS.

IE Bulletin 82-02
NUREG-1095: EVALUATION OF RESPONSES TO IE BULLETIN 82-02.Degradation Of Threaded Fasteners In Reactor Coolant Pressure Boundary Of Pressurized Water-Reactor Plants.

IEEE 383 Flame Test
NUREG/CR-4112 V01: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task A:IEEE Flame Test.

IGSCC
NUREG-1061 V05: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 5:Summary - Piping Review Committee Conclusions and Recommendations.

ILRT
NUREG/CR-4220: RELIABILITY ANALYSIS OF CONTAINMENT ISOLATION SYSTEMS.

IPRD
NUREG/CR-3831: THE IN-PLANT RELIABILITY DATA BASE FOR NUCLEAR PLANT COMPONENTS.Interim Report - Diesel Generators,Batteries, Chargers And Inverters.

Ice Condenser
NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT.
NUREG/CR-3954: HECTR ANALYSIS OF EQUIPMENT TEMPERATURE RESPONSES TO SELECTED HYDROGEN BURNS IN AN ICE CONDENSER CONTAINMENT.
NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.

Illinois
NUREG/CR-4333: STE. GENEVIEVE FAULT ZONE,MISSOURI AND ILLINOIS.

Immunity
NUREG/CR-4133: NUCLEAR POWER SAFETY REPORTING SYSTEM IMPLEMENTATION AND OPERATIONAL SPECIFICATIONS.

Importance Ranking
NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.

Importation
NUREG/CR-4357: THE FEASIBILITY OF DETECTING THE IMPORT OF UNAUTHORIZED RADIOACTIVE MATERIALS INTO THE UNITED STATES.

Impoundment
NUREG/CR-4075: DESIGNING PROTECTIVE COVERS FOR URANIUM MILL TAILINGS PILES. A Review.

In Situ Leaching
NUREG/CR-3709: METHODS OF MINIMIZING GROUND-WATER CONTAMINATION FROM IN SITU LEACH URANIUM MINING.Final Report.

In-Plant Reliability Data
NUREG/CR-3831: THE IN-PLANT RELIABILITY DATA BASE FOR NUCLEAR PLANT COMPONENTS.Interim Report - Diesel Generators,Batteries, Chargers And Inverters.

In-Situ Soil Test
NUREG/CR-4430: CURRENT METHODOLOGIES FOR ASSESSING THE POTENTIAL FOR EARTHQUARKE-INDUCED LIQUEFACTION IN SOILS.

Incident
NUREG-0090 V07 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1984.

Incident Data Analysis
NUREG/CR-3611: RADIOACTIVE MATERIAL (RAM) ACCIDENT/INCIDENT DATA ANALYSIS PROGRAM.

Inconel 600
NUREG/CR-3197 V01: REACTION BETWEEN SOME CESIUM-IODINE COMPOUNDS AND THE REACTOR MATERIALS 304 STAINLESS STEEL,INCONEL 600 & SILVER. Volume I:Cesium Hydroxide Reactions.

Independent Assessment
NUREG/CR-3919: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:NEPTUNUS PRESSURIZER TEST Y05.
NUREG/CR-4252: INDEPENDENT ASSESSMENT OF TRAC-PD2/MOD1 CODE WITH BCL ECC BYPASS TESTS.

Independent Safety Organization
NUREG/CR-4152: AN INDEPENDENT SAFETY ORGANIZATION.

Indexes
NUREG-0304 V09 N04: REGULATORY AND TECHNICAL REPORTS.Annual Compilation For 1984.
NUREG-0304 V10 N02: REGULATORY AND TECHNICAL REPORTS.Compilation For Second Quarter 1985,April-June.
NUREG-0750 V20 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY-SEPTEMBER 1984.
NUREG-0750 V20 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY-DECEMBER 1984.
NUREG-0750 V21 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY-MARCH 1985.
NUREG-0750 V21 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY-MARCH 1985.

Indiana
NUREG/CR-3145 V03: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION - ANNUAL REPORT.(October 1982 - September 1983. Volume 3).

Inert Gas
NUREG/CR-4074: THE PERFORMANCE OF DEFECTED SPENT LWR FUEL RODS IN INERT GAS AND DRY AIR STORAGE ATMOSPHERES.

Influence Coefficient
NUREG/CR-3723: STRESS-INTENSITY-FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS.

Information Engineering
NUREG/CR-4322 V01: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 1: Enterprise Model.
NUREG/CR-4322 V02: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 2: Data Entity Dictionary.
NUREG/CR-4322 V03: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 3: Data Model.
NUREG/CR-4322 V04: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 4: Preliminary Strategic Data Plan.

Informed Opinion
NUREG/CP-0063: PROCEEDINGS OF THE 1984 STATISTICAL SYMPOSIUM ON NATIONAL ENERGY ISSUES.

Infrared Analysis
NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984.

Inhalation
NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984.

Inservice Inspection
NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.
NUREG-1155 V04: RESEARCH PROGRAM PLAN.Non-Destructive Examination.

Inspection
NUREG-0040 V08 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1984. (White Book)
NUREG-0040 V09 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1985.(White Book)
NUREG-0040 V09 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,April-June 1985. (White Book)
NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.
NUREG-1153: INSPECTION REPORT OF UNAUTHORIZED POSSESSION AND USE OF UNSEALED AMERICIUM-241 AND SUBSEQUENT CONFISCATION. J.C. Haynes Company,Newark,Ohio.
NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.
NUREG/CR-3949 V02: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Annual Progress Report For Period Ending December 31,1984.
NUREG/CR-4170: AN ULTRA-HIGH SPEED RESIDUE PROCESSOR FOR SAFT INSPECTION SYSTEM IMAGE ENHANCEMENT.
NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.
NUREG/CR-4440: A REVIEW OF EMERGENCY DIESEL GENERATOR PERFORMANCE AT NUCLEAR POWER PLANTS.

Instructor Training
NUREG/CR-4344: INSTRUCTIONAL SKILLS EVALUATION IN NUCLEAR INDUSTRY TRAINING.

Instrumentation
NUREG/CR-4239: ANALYSIS OF THE ABILITY OF CURRENT HEALTH PHYSICS INSTRUMENTS TO PREDICT DOSE IN EXPOSED INDIVIDUALS.

Instrumentation Circuit
NUREG/CR-3863: ASSESSMENT OF CLASS 1E PRESSURE TRANSMITTER RESPONSE WHEN SUBJECTED TO HARSH ENVIRONMENT SCREENING TESTS.

Integral Systems
NUREG/CP-0058 V01: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.

Integrated Control System
NUREG/CR-3991: FAILURE MODES AND EFFECTS ANALYSIS (FMEA) OF THE ICS/NNI ELECTRIC POWER DISTRIBUTION CIRCUITRY AT THE OCONEE-1 NUCLEAR PLANT.

Integrated Leak Rate Test
NUREG/CR-4220: RELIABILITY ANALYSIS OF CONTAINMENT ISOLATION SYSTEMS.

Integrated Plant Safety Assessment
NUREG-0824 S01: INTEGRATED PLANT SAFETY ASSESSMENT REPORT.SYSTEMATIC EVALUATION PROGRAM-MILLSTONE NUCLEAR POWER STATION,UNIT 1. Docket No. 50-245. (Northeast Nuclear Energy Company)
NUREG-0829 DRFT: INTEGRATED PLANT SAFETY ASSESSMENT REPORT.SYSTEMATIC EVALUATION PROGRAM - SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1.Docket No. 50-206.(Southern California Edison Company)

Interactions Testing
NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT.

Interfacial Transfer Integral
NUREG/CR-3989: TIME- AND VOLUME-AVERAGED CONSERVATION EQUATIONS FOR MULTIPHASE FLOW.Part One:System Without Internal Solid Structures.

Intergranular Stress Corrosion Cracking
NUREG/CR-3613 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Annual Report For 1984.
NUREG/CR-3613 V03 N1: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Semiannual Report For October 1984 Through March 1985.

International Probabilistic Safety Assessment
NUREG/CR-4153: APPLICATIONS OF FOREIGN PROBABILISTIC SAFETY ASSESSMENT EXPERIENCE TO THE U.S. NUCLEAR REGULATORY PROCESS.

International Standard Problem 13
NUREG/CR-4115: INTERNATIONAL STANDARD PROBLEM 13 (LOFT EXPERIMENT L2-5). Final Comparison Report.

Intestinal Absorption
NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS:Application To Establishing Values Of f1 For Soluble Plutonium.

Inventory
NUREG/CR-3865: EVALUATION OF THE RADIOACTIVE INVENTORY IN,AND ESTIMATION OF ISOTOPIC RELEASE FROM,THE WASTE IN

EIGHT TRENCHES AT THE SHEFFIELD LOW-LEVEL WASTE BURIAL SITE.

Inventory Difference Data

NUREG-0430 V05 N01: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.January 1984 - June 1984.(Gray Book II)

Inverted Annular Flow

NUREG/CR-4277: INVERTED ANNUAL FLOW EXPERIMENTAL STUDY.

Iodine

NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

NUREG/CR-3514 V02: THE CHEMICAL BEHAVIOR OF IODINE IN AQUEOUS SOLUTIONS UP TO 150 C.II.Radiation-Redox Conditions.

NUREG/CR-3990: CHARCOAL PERFORMANCE UNDER ACCIDENT CONDITIONS IN LIGHT-WATER REACTORS.

NUREG/CR-4081: ABSORPTION OF GASEOUS IODINE BY WATER DROPLETS.

Ion Exchange Resins

NUREG/CR-4150: EPICOR-II RESIN DEGRADATION RESULTS FROM FIRST RESIN SAMPLES OF PF-8 AND PF-20.

Ionization Chamber

NUREG/CR-4131: INVESTIGATION OF ALTERNATIVE MEANS TO ACCOMPLISH THE GOALS OF BIENNIAL ION CHAMBER CALIBRATION.

Iron Oxide

NUREG/CR-2718: STEAM EXPLOSION EXPERIMENTS WITH SINGLE DROPS OF IRON OXIDE MELTED WITH A CO2 LASER.Part II.Parametric Studies.

Irradiated Fuel

NUREG-0725 R05: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL.

NUREG/CR-4203: A CALCULATIONAL METHOD FOR DETERMINING BIOLOGICAL DOSE RATES FROM IRRADIATED RESEARCH REACTOR FUEL.

Irradiated Zircaloy Cladding

NUREG/CR-3980 V03: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report.July-September 1984.

NUREG/CR-3980 V04: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report.October-December 1984.

Irradiation

NUREG-1155 V01: RESEARCH PROGRAM PLAN.Reactor Vessels.

NUREG/CR-3872: DATA ACQUISITION AND CONTROL OF THE HSST SERIES V IRRADIATION EXPERIMENT AT THE ORR.

NUREG/CR-3978: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL PLATE AND WELDS FOR THE FOURTH HSST IRRADIATION SERIES.

NUREG/CR-3999: ELECTRICALLY HEATED EX-REACTOR PELLET-CLADDING INTERACTION (PCI) SIMULATIONS UTILIZING IRRADIATED ZIRCALOY CLADDING.

NUREG/CR-4039: GAMMA-RAY CHARACTERIZATION OF THE TWO-YEAR IRRADIATION EXPERIMENT PERFORMED AT THE POOLSIDE FACILITY.

NUREG/CR-4219 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.

NUREG/CR-4239: ANALYSIS OF THE ABILITY OF CURRENT HEALTH PHYSICS INSTRUMENTS TO PREDICT DOSE IN EXPOSED INDIVIDUALS.

NUREG/CR-4275: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM - FIVE-YEAR PLAN FY 1984-1988.

NUREG/CR-4284: NEUTRON EXPOSURE PARAMETERS FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY IRRADIATION SERIES.

NUREG/CR-4395: CORRELATION OF CV AND KIC/KJC TRANSITION TEMPERATURE INCREASES DUE TO IRRADIATION.

Isotope Exchange

NUREG/CR-3990: CHARCOAL PERFORMANCE UNDER ACCIDENT CONDITIONS IN LIGHT-WATER REACTORS.

Isotopic Release Rates

NUREG/CR-3865: EVALUATION OF THE RADIOACTIVE INVENTORY IN,AND ESTIMATION OF ISOTOPIC RELEASE FROM,THE WASTE IN EIGHT TRENCHES AT THE SHEFFIELD LOW-LEVEL WASTE BURIAL SITE.

Issues

NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.

J-Integral Resistance Curve

NUREG/CR-4283: STUDY OF THE EFFECTS OF ELASTIC UNLOADINGS ON THE JI-R CURVES FROM COMPACT SPECIMENS.

J-Integral/Tearing Modulus

NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.

NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

Japan

NUREG/CP-0059 V01: PROCEEDINGS OF THE MITI-NRC SEISMIC INFORMATION EXCHANGE MEETING,VOLUME I.

NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE.

Job Performance Aid

NUREG/CR-3817: DEVELOPMENT,USE AND CONTROL OF MAINTENANCE PROCEDURES IN NUCLEAR POWER PLANTS.Problems And Recommendations.

KNK II

NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.

Krypton

NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

LAMPRE

NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.

LER

NUREG-1022 S02: LICENSEE EVENT REPORT SYSTEM.Evaluation Of First Year Results And Recommendations For Improvements.

NUREG/CR-2000 V03N12: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of December 1984.

NUREG/CR-2000 V04 N1: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of January 1985.

NUREG/CR-2000 V04 N2: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of February 1985.

NUREG/CR-2000 V04 N3: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of March 1985.

NUREG/CR-2000 V04 N4: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of April 1985.

NUREG/CR-2000 V04 N5: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of May 1985.

NUREG/CR-2000 V04 N6: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of June 1985.

NUREG/CR-2000 V04 N7: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of July 1985.

NUREG/CR-2000 V04 N8: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of August 1985.

NUREG/CR-2000 V04 N9: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of September 1985.

NUREG/CR-2000 V04N10: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of October 1985.

NUREG/CR-2000 V04N11: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of November 1985.

NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.

NUREG/CR-4178 DRFT: AN EVALUATION OF SELECTED LICENSEE EVENT REPORTS PREPARED PURSUANT TO 10 CFR 50.73.Draft Report.

NUREG/CR-4220: RELIABILITY ANALYSIS OF CONTAINMENT ISOLATION SYSTEMS.

LIDAR

NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3.Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.

LOCA
NUREG/CR-3633 V01 S1: TRAC-BC1/MOD1 AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.
NUREG/CR-3802: RELAP5 ASSESSMENT:QUANTITATIVE KEY PARAMETERS AND RUN TIME STATISTICS.
NUREG/CR-4044: TRAC-PF1 LOCA CALCULATIONS USING FINE-NODE AND COARSE-NODE INPUT MODELS.
NUREG/CR-4127 V01: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 1:Numerical Methods.
NUREG/CR-4167: FLECHT SEASET PROGRAM.Final Report.NRC/EPRI Westinghouse Report Number 16.
NUREG/CR-4196: OVERVIEW OF TRAC-BD1 (VERSION 12) ASSESSMENT STUDIES.
NUREG/CR-4218: LOCA SIMULATION IN THE NATIONAL RESEARCH UNIVERSAL REACTOR PROGRAM.Postirradiation Examination Results For The Third Materials Test (MT-3) - Second Campaign.

LOFT
NUREG/CR-3005: SUMMARY OF THE NUCLEAR REGULATORY COMMISSION'S LOFT PROGRAM RESEARCH FINDINGS.

LOFT L2-5
NUREG/CR-4115: INTERNATIONAL STANDARD PROBLEM 13 (LOFT EXPERIMENT L2-5). Final Comparison Report.

Leach Test
NUREG/CR-4379 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report.Year Four April-June 1985.

Leachability
NUREG/CR-4181: LEACHABILITY OF RADIONUCLIDES FROM CEMENT SOLIDIFIED WASTE FORMS PRODUCED AT OPERATING NUCLEAR POWER REACTORS.

Leachate Movement
NUREG/CR-4061: LEACHATE PLUME MIGRATION DOWNGRADIENT FROM URANIUM TAILINGS DISPOSAL IN MINE STOPES.

Leaching
NUREG/CR-2482 V06: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program.October 1983 - March 1984.
NUREG/CR-4198: FRACTURE IN GLASS/HIGH LEVEL WASTE CANISTERS.
NUREG/CR-4259: TAILINGS NEUTRALIZATION AND OTHER ALTERNATIVES FOR IMMOBILIZING TOXIC MATERIALS IN TAILINGS.Final Report.

Leak
NUREG-1155 V04: RESEARCH PROGRAM PLAN.Non-Destructive Examination.
NUREG/CR-3855: CHARACTERIZATION OF NUCLEAR REACTOR CONTAINMENT PENETRATION - FINAL REPORT.
NUREG/CR-3952: SEQUOYAH EQUIPMENT HATCH SEAL LEAKAGE.
NUREG/CR-4294: LEAK RATE ANALYSIS OF THE WESTINGHOUSE REACTOR COOLANT PUMP.
NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

Leak Rate
NUREG-1037 DRFT FC: CONTAINMENT PERFORMANCE WORKING GROUP REPORT.Draft Report For Comment.

Leak-Before-Break
NUREG-1155 V03: RESEARCH PROGRAM PLAN.Piping.
NUREG/CR-4305: COMMENTS ON THE LEAK-BEFORE-BREAK CONCEPT FOR NUCLEAR POWER PLANT PIPING SYSTEMS.

Leakage
NUREG-1095: EVALUATION OF RESPONSES TO IE BULLETIN 82-02.Degradation Of Threaded Fasteners In Reactor Coolant Pressure Boundary Of Pressurized Water Reactor Plants.
NUREG/CR-4209: COMPARISON OF ANALYTICAL PREDICTIONS AND EXPERIMENTAL RESULTS FOR A 1:8-SCALE STEEL CONTAINMENT MODEL PRESSURIZED TO FAILURE.

Legal Issuances
NUREG-0750 V20 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1984. Pages 1,055-1,436.
NUREG-0750 V20 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1984. Pages 1,437-1,572.
NUREG-0750 V20 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1984. Pages 1,573-1,706.
NUREG-0750 V21 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1985. Pages 1-273.
NUREG-0750 V21 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1985. Pages 275-469.
NUREG-0750 V21 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1985. Pages 471-559.
NUREG-0750 V21 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1985. Pages 561-1,041.
NUREG-0750 V21 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1985. Pages 1,043-1,567.
NUREG-0750 V21 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1985. Pages 1,569-1,786.
NUREG-0750 V22 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1985. Pages 1-176.
NUREG-0750 V22 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1985. Pages 177-457.
NUREG-0750 V22 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1985. Pages 459-649.
NUREG-0750 V22 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1985. Pages 651-769.

Licensed Operating Reactors
NUREG-0020 V08 N12: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of November 30,1984.(Gray Book I)
NUREG-0020 V09 N01: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of December 31,1984.(Gray Book I)
NUREG-0020 V09 N02: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of January 31,1985.(Gray Book I)
NUREG-0020 V09 N03: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of Febuary 28,1985.(Gray Book I)
NUREG-0020 V09 N04: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of March 31,1985.(Gray Book I)
NUREG-0020 V09 N05: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of April 30, 1985.(Gray Book I)
NUREG-0020 V09 N06: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of May 31,1985.(Gray Book I)
NUREG-0020 V09 N07: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of June 30,1985.(Gray Book I)
NUREG-0020 V09 N08: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of July 31,1985.(Gray Book I)
NUREG-0020 V09 N09: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of August 31,1985.(Gray Book I)
NUREG-0020 V09 N10: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of September 30,1985.(Gray Book I)
NUREG-0020 V09 N11: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of October 31,1985.(Gray Book I)

Licensee Contractor And Vendor Inspection
NUREG-0040 V09 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report.January-March 1985.(White Book)
NUREG-0040 V09 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report.April-June 1985.(White Book)
NUREG-0040 V09 N03: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report.July 1985-September 1985.(White Book)

Licensee Event Report
NUREG-1022 S02: LICENSEE EVENT REPORT SYSTEM.Evaluation Of First Year Results And Recommendations For Improvements.
NUREG/CR-2000 V03N12: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of December 1984.
NUREG/CR-2000 V04 N1: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of January 1985.
NUREG/CR-2000 V04 N2: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of February 1985.
NUREG/CR-2000 V04 N3: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of March 1985.
NUREG/CR-2000 V04 N4: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of April 1985.
NUREG/CR-2000 V04 N5: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of May 1985.
NUREG/CR-2000 V04 N6: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of June 1985.
NUREG/CR-2000 V04 N7: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of July 1985.
NUREG/CR-2000 V04 N8: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of August 1985.
NUREG/CR-2000 V04 N9: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of September 1985.

NUREG/CR-2000 V04N10: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of October 1985.
NUREG/CR-2000 V04N11: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of November 1985.
NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.
NUREG/CR-3519: HUMAN ERROR PROBABILITY ESTIMATION USING LICENSEE EVENT REPORTS.
NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-4071: EXPLORATORY TREND AND PATTERN ANALYSIS FOR 1981 LICENSEE EVENT REPORT DATA.
NUREG/CR-4178 DRFT: AN EVALUATION OF SELECTED LICENSEE EVENT REPORTS PREPARED PURSUANT TO 10 CFR 50.73.Draft Report.
NUREG/CR-4220: RELIABILITY ANALYSIS OF CONTAINMENT ISOLATION SYSTEMS.

Licensee Security
NUREG/CR-4250: VEHICLE BARRIERS:EMPHASIS ON NATURAL FEATURES.

Liquefaction
NUREG/CR-4430: CURRENT METHODOLOGIES FOR ASSESSING THE POTENTIAL FOR EARTHQUARKE-INDUCED LIQUEFACTION IN SOILS.

Liquid Entrainment
NUREG/CR-4164: DATA REPORT FOR THE TPFL TEE/CRITICAL FLOW EXPERIMENTS.

Liquid Metal
NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.

Liquid Pathways
NUREG-1165: ESRP 7.1.1 "ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RELEASES OF RADIOACTIVE MATERIALS TO GROUNDWATER."

Load Combination
NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984.
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.

Load Factor
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.

Load Ratio
NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS.
NUREG/CR-4422: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN PWR ENVIRONMENTS.

Local Interfacial
NUREG/CR-4414: DIRECT-CONTACT CONDENSATION OF STEAM ON COLD WATER IN STRATIFIED COUNTERCURRENT FLOW.

Long-Range Research Plan
NUREG-1080 V02: LONG-RANGE RESEARCH PLAN FY 1986-FY 1990.

Loss Of AC Power
NUREG/CR-4342: UNCERTAINTY AND SENSITIVITY ANALYSIS OF A MODEL FOR MULTICOMPONENT AEROSOL DYNAMICS.

Loss Of Feedwater
NUREG-0090 V08 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1985.

Loss Of Main And Auxiliary Feedwater
NUREG-1154: LOSS OF MAIN AND AUXILIARY FEEDWATER EVENT AT THE DAVIS-BESSE PLANT ON JUNE 9, 1985.

Loss Of Offsite Power
NUREG/CR-3992: COLLECTION AND EVALUATION OF COMPLETE AND PARTIAL LOSSES OF OFF-SITE POWER AT NUCLEAR POWER PLANTS.

Loss-Of-Coolant Accident
NUREG/CR-3426 V02: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY.Data Report.
NUREG/CR-3633 V01 S1: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.
NUREG/CR-3648: TRAC-PF1 INDEPENDENT ASSESSMENT.
NUREG/CR-3651: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES.
NUREG/CR-3764: BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE.
NUREG/CR-3802: RELAP5 ASSESSMENT:QUANTITATIVE KEY PARAMETERS AND RUN TIME STATISTICS.
NUREG/CR-3866: TRAC-PD2 INDEPENDENT ASSESSMENT.
NUREG/CR-3936: RELAP5 ASSESSMENT:CONCLUSIONS AND USER GUIDELINES.
NUREG/CR-4127 V01: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 1:Numerical Methods.
NUREG/CR-4196: OVERVIEW OF TRAC-BD1 (VERSION 12) ASSESSMENT STUDIES.
NUREG/CR-4218: LOCA SIMULATION IN THE NATIONAL RESEARCH UNIVERSAL REACTOR PROGRAM:Postirradiation Examination Results For The Third Materials Test (MT-3) - Second Campaign.
NUREG/CR-4400: THE IMPACT OF MECHANICAL-AND MAINTENANCE-INDUCED FAILURES OF MAIN REACTOR COOLANT PUMP SEALS ON PLANT SAFETY.

Loss-Of-Fluid Test
NUREG/CR-3005: SUMMARY OF THE NUCLEAR REGULATORY COMMISSION'S LOFT PROGRAM RESEARCH FINDINGS.
NUREG/CR-4046: DETERMINING CRITICAL FLOW VALVE CHARACTERISTICS USING EXTRAPOLATION TECHNIQUES.
NUREG/CR-4278: TRAC-PF1/MOD1 DEVELOPMENT ASSESSMENT.

Low Enriched Uranium
NUREG-1065 R01: ACCEPTANCE CRITERIA FOR THE LOW ENRICHED URANIUM REFORM AMENDMENTS.
NUREG-1065 R01: ACCEPTANCE CRITERIA FOR THE LOW ENRICHED URANIUM REFORM AMENDMENTS.

Low Strategic Significance
NUREG-1065 R01: ACCEPTANCE CRITERIA FOR THE LOW ENRICHED URANIUM REFORM AMENDMENTS.

Low-Level Waste
NUREG/CR-3774 V02: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2A:Technical Requirements For Belowground Vault Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V03: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2B:Technical Requirements For Aboveground Vault Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V04: ALTERNATIVE METHOD FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2C:Technical Requirements For Earth Mounded Concrete Bunker Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V05: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2E:Technical Requirements For Shaft Disposal Of Low Level Radioactive Waste.
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.
NUREG/CR-3865: EVALUATION OF THE RADIOACTIVE INVENTORY IN,AND ESTIMATION OF ISOTOPIC RELEASE FROM,THE WASTE IN EIGHT TRENCHES AT THE SHEFFIELD LOW-LEVEL WASTE BURIAL SITE.
NUREG/CR-4062: EXTENDED STORAGE OF LOW-LEVEL RADIOACTIVE WASTES:Potential Problem Areas.
NUREG/CR-4069: ANALYSES OF SOILS FROM AN AREA ADJACENT TO THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE AT SHEFFIELD,ILLINOIS.
NUREG/CR-4101: ASSAY OF LONG-LIVED RADIONUCLIDES IN LOW-LEVEL WASTES FROM POWER REACTORS.
NUREG/CR-4150: EPICOR-II RESIN DEGRADATION RESULTS FROM FIRST RESIN SAMPLES OF PF-8 AND PF-20.
NUREG/CR-4181: LEACHABILITY OF RADIONUCLIDES FROM CEMENT SOLIDIFIED WASTE FORMS PRODUCED AT OPERATING NUCLEAR POWER REACTORS.

NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984.
NUREG/CR-4200: BIODEGRADATION TESTING OF SOLIDIFIED LOW-LEVEL WASTE STREAMS.
NUREG/CR-4201: THERMAL STABILITY TESTING OF LOW-LEVEL WASTE FORMS.
NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA.
NUREG/CR-4268: RATIO METHODS FOR COST-EFFECTIVE FIELD SAMPLING OF COMMERCIAL RADIOACTIVE LOW-LEVEL WASTES.
NUREG/CR-4352: SUGGESTED STATE REQUIREMENTS AND CRITERIA FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE REGULATORY PROGRAM.
NUREG/CR-4406: AN ANALYSIS OF LOW-LEVEL WASTES.Review of Hazardous Waste Regulations And Identification of Radioactive Mixed Wastes. Final Report.
NUREG/CR-4435: ORGANIC COMPLEXANT-ENHANCED MOBILITY OF TOXIC ELEMENTS IN LOW-LEVEL WASTES.Annual Report,July 1984 - June 1985.

Low-Temperature Aging
NUREG/CR-4204: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Annual Report,October 1983 - September 1984.

Lumped Parameter Method
NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS.

MAEROS
NUREG/CR-4342: UNCERTAINTY AND SENSITIVITY ANALYSIS OF A MODEL FOR MULTICOMPONENT AEROSOL DYNAMICS.

MAG-1 Spectacle
NUREG/CR-3953: THE USE OF MAG-1 SPECTACLES WITH POSITIVE- AND NEGATIVE-PRESSURE RESPIRATORS.

MAPPS
NUREG/CR-3626 V02: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: DESCRIPTION OF MODEL CONTENT,STRUCTURE,AND SENSITIVITY TESTING.
NUREG/CR-3634: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: USER'S MANUAL.
NUREG/CR-4104: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL. Field Evaluation/Validation.

MARCH
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).
NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT.

MARCH-2
NUREG/CR-4172: A USER'S GUIDE FOR MERGE.

MATADOR
NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS.
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.

MC&A
NUREG/CR-4108: DEVELOPMENT OF MC&A ALARM RESOLUTION PROCEDURES.

MELCOR
NUREG/CR-4169: AN APPROACH TO TREATING RADIONUCLIDE DECAY HEATING FOR USE IN THE MELCOR CODE SYSTEM.
NUREG/CR-4185: AN ASSESSMENT OF DOSIMETRY DATA FOR ACCIDENTAL RADIONUCLIDE RELEASES FROM NUCLEAR REACTORS.
NUREG/CR-4199: A DEMONSTRATION UNCERTAINTY/SENSITIVITY ANALYSIS USING THE HEALTH AND ECONOMIC CONSEQUENCE MODEL CRAC2.

MELPROG
NUREG/CR-4340 V01: REACTOR SAFETY RESEARCH SEMIANNUAL REPORT.January - June 1985.

MELRPI MOD2
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.

MERGE
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).
NUREG/CR-4172: A USER'S GUIDE FOR MERGE.

MESODIF
NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.

MINET
NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984.
NUREG/CR-2331 V04 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July 1 -September 30,1984.
NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.

MINETQ
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.

MINTEQ
NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984.

MOD1
NUREG/CR-3802: RELAP5 ASSESSMENT:QUANTITATIVE KEY PARAMETERS AND RUN TIME STATISTICS.
NUREG/CR-4189: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:Semiscale MOD-2A Feedwater-Line Break (S-SF-3) And Steam-Line Break (S-SF-5) Tests.
NUREG/CR-4428: OVERVIEW OF TRAC-BD1/MOD1 ASSESSMENT STUDIES.

MOSREP
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP):A Methodology For Estimating Risk Impacts Of Safeguards Measures.

Magnetic
NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983.

Maintenance
NUREG/CR-3626 V02: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: DESCRIPTION OF MODEL CONTENT,STRUCTURE,AND SENSITIVITY TESTING.
NUREG/CR-3634: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: USER'S MANUAL.
NUREG/CR-3817: DEVELOPMENT,USE AND CONTROL OF MAINTENANCE PROCEDURES IN NUCLEAR POWER PLANTS.Problems And Recommendations.
NUREG/CR-3837: MULTIPLE-SEQUENTIAL FAILURE MODEL:Evaluation Of And Procedures For Human Error Dependency.
NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE.
NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.
NUREG/CR-4281: AN EMPIRICAL ANALYSIS OF SELECTED NUCLEAR POWER PLANT MAINTENANCE FACTORS AND PLANT SAFETY.

Maintenance Personnel
NUREG/CR-4104: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL. Field Evaluation/Validation.

Malfunction
NUREG-1154: LOSS OF MAIN AND AUXILIARY FEEDWATER EVENT AT THE DAVIS-BESSE PLANT ON JUNE 9, 1985.

Mandrel Loading Test
NUREG/CR-3980 V02: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,April-June 1984.
NUREG/CR-3980 V03: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,July-September 1984.

Manufacture
NUREG-1127: RADIATION PROTECTION TRAINING AT URANIUM HEXAFLUORIDE AND FUEL FABRICATION PLANTS.

Marine Environment
NUREG/CR-4190: CALIFORNIA OFFSHORE SURVEY OF LICENSEES USING RADIOACTIVE MATERIAL.

Material Control And Accounting
NUREG/CR-4108: DEVELOPMENT OF MC&A ALARM RESOLUTION PROCEDURES.

Material License
NUREG/CR-3657: PRELIMINARY SCREENING OF FUEL CYCLE AND BY-PRODUCT MATERIAL LICENSES FOR EMERGENCY PLANNING.

Material Transport
NUREG/CR-4260: TORAC USER'S MANUAL.A Computer Code For Analyzing Tornado-Induced Flow And Material Transport In Nuclear Facilities.

Materials Control Unit
NUREG/CR-4107: SEQUENTIAL TEST PROCEDURES FOR DETECTING PROTRACTED MATERIALS LOSSES.

Materials Property Test
NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,January-March 1985.

Materials Test
NUREG/CR-4218: LOCA SIMULATION IN THE NATIONAL RESEARCH UNIVERSAL REACTOR PROGRAM.Postirradiation Examination Results For The Third Materials Test (MT-3) - Second Campaign.

Mathematical Model
NUREG/CR-3659: A MATHEMATICAL MODEL FOR ASSESSING THE UNCERTAINTIES OF INSTRUMENTATION MEASUREMENTS FOR POWER AND FLOW OF PWR REACTORS.

Measurement
NUREG/CR-3481 V02: NUCLEAR POWER PLANT PERSONNEL QUALIFICATIONS AND TRAINING: TAPS -- The Task Analysis Profiling System.
NUREG/CR-3609: EVALUATION OF NEUTRON DOSIMETRY TECHNIQUES FOR WELL-LOGGING OPERATIONS.
NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION.
NUREG/CR-4397: IN-PLANT SOURCE TERM MEASUREMENTS AT PRAIRIE ISLAND NUCLEAR GENERATING STATION.

Measurement Uncertainties
NUREG/CR-3659: A MATHEMATICAL MODEL FOR ASSESSING THE UNCERTAINTIES OF INSTRUMENTATION MEASUREMENTS FOR POWER AND FLOW OF PWR REACTORS.

Mechanical Snubbers
NUREG/CR-4006: CLOSEOUT OF IE BULLETIN 81-01:SURVEILLANCE OF MECHANICAL SNUBBERS.

Mechanical Stability
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.

Melt Discharge
NUREG/CR-4383: HIGH PRESSURE INJECTION OF MELT FROM A REACTOR PRESSURE VESSEL - THE DISCHARGE PHASE.

Meltdown
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.
NUREG/CR-4172: A USER'S GUIDE FOR MERGE.
NUREG/CR-4388: AEROSOL BEHAVIOR MODELING (TASK 3) - SUPPORT SERVICES FOR RESEARCH AND EVALUATION OF SEVERE ACCIDENT PHENOMENA AND MITIGATION.

Metallurgical Blind Test
NUREG/CR-4039: GAMMA-RAY CHARACTERIZATION OF THE TWO-YEAR IRRADIATION EXPERIMENT PERFORMED AT THE POOLSIDE FACILITY.

Meteorology
NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.

Methodology For Estimating Risk Impacts
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP).A Methodology For Estimating Risk Impacts Of Safeguards Measures.

Mexican Incident
NUREG-1103: CONTAMINATED MEXICAN STEEL.Importation Of Steel Into The United States That Had Been Inadvertently Contaminated With Cobalt-60 As A Result Of Scrapping Of A Teletherapy Unit.

Micro-R-Meter
NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.

Microcomputer Software
NUREG/CR-4367: ORVIRT.PC:A 2-D FINITE-ELEMENT FRACTURE ANALYSIS PROGRAM FOR A MICROCOMPUTER.

Microstructure
NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984.

Migration
NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER (Final Report).
NUREG/CR-4114: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS.II.

Millers
NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.

Mineralogical Characterization
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984.
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.

Miners
NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.

Minicomputer Simulation
NUREG/CR-3943: THE BWR PLAN ANALYZER.

Missile
NUREG/CR-3551: SAFETY IMPLICATIONS ASSOCIATED WITH IN-PLANT PRESSURIZED GAS STORAGE AND DISTRIBUTION SYSTEMS IN NUCLEAR POWER PLANTS.

Missouri
NUREG/CR-4333: STE. GENEVIEVE FAULT ZONE,MISSOURI AND ILLINOIS.

Mitigation Of Errosion
NUREG/CR-3752: EFFECTS OF HYDROLOGIC VARIABLES ON ROCK RIPRAP DESIGN FOR URANIUM TAILINGS IMPOUNDMENTS.

Mitigation Technique

NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions.

NUREG/CR-4251 V02: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 2:Case Study Analysis Of Hydrologic Characterization And Mitigative Schemes.

Mobility

NUREG-1164: INFORMATION ON THE CONFINEMENT CAPABILITY OF THE FACILITY DISPOSAL AREA AT WEST VALLEY,NEW YORK.

NUREG/CR-4435: ORGANIC COMPLEXANT-ENHANCED MOBILITY OF TOXIC ELEMENTS IN LOW-LEVEL WASTES.Annual Report,July 1984 - June 1985.

Modal Parameters

NUREG/CR-4432: COMPARISON OF DYNAMIC CHARACTERISTICS OF FUKUSHIMA NUCLEAR POWER PLANT CONTAINMENT BUILDING DETERMINED FROM TESTS AND EARTHQUAKES.

Model Development

NUREG/CR-4127 V01: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 1:Numerical Methods.

Model Uncertainty

NUREG/CR-4157: A SCIENTIFIC CRITIQUE OF AVAILABLE MODELS FOR REAL-TIME SIMULATIONS OF DISPERSION.

Modeling

NUREG/CR-2482 V06: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program,October 1983 - March 1984.

NUREG/CR-3626 V02: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: DESCRIPTION OF MODEL CONTENT,STRUCTURE,AND SENSITIVITY TESTING.

NUREG/CR-3634: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: USER'S MANUAL.

NUREG/CR-3741 V02: EVALUATION OF POWER REACTOR FUEL ROD ANALYSIS CAPABILITIES: Phase 2 Topical Report.Volume 2:Code Evaluation.

NUREG/CR-4173: CORSOR USER'S MANUAL.

NUREG/CR-4322 V01: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 1: Enterprise Model.

NUREG/CR-4322 V02: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 2: Data Entity Dictionary.

NUREG/CR-4322 V03: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 3: Data Model.

NUREG/CR-4322 V04: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 4: Preliminary Strategic Data Plan.

NUREG/CR-4350 V01: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 1 - PRA Fundamentals.

NUREG/CR-4360 V01: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.

NUREG/CR-4360 V02: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.

NUREG/CR-4388: AEROSOL BEHAVIOR MODELING (TASK 3) - SUPPORT SERVICES FOR RESEARCH AND EVALUATION OF SEVERE ACCIDENT PHENOMENA AND MITIGATION.

NUREG/CR-4402 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31,1985.

Modification

NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

Monitoring

NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984.

NUREG-0837 V04 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, July-September 1984.

NUREG-0837 V04 N04: NRC TLD DIRECT RADIATION MONITORING REPORT.Progress Report, October-December 1984.

NUREG-0837 V05 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, January-March 1985.

NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.

NUREG-1155 V04: RESEARCH PROGRAM PLAN.Non-Destructive Examination.

NUREG/CR-3916: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL.

NUREG/CR-3943: THE BWR PLAN ANALYZER.

NUREG/CR-4100: EVALUATION OF INSTRUMENTAL METHODS FOR THE MEASUREMENT OF YELLOWCAKE EMISSIONS.

NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.

NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.

NUREG/CR-4297: EXTREMITY MONITORING:Considerations For Use,Dosimeter Placement,And Evaluation.

NUREG/CR-4300 V01: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Progress Report, October-March 1985.

NUREG/CR-4352: SUGGESTED STATE REQUIREMENTS AND CRITERIA FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE REGULATORY PROGRAM.

NUREG/CR-4354: A STUDY OF SEISMICITY AND TECTONICS IN NEW ENGLAND.Final Report.

Monoenergetic Electron Sources

NUREG/CR-4266: STANDARD BETA-PARTICLE AND MONOENERGETIC ELECTRON SOURCES FOR THE CALIBRATION OF BETA-RADIATION PROTECTION INSTRUMENTATION.

Monte-Carlo Simulation

NUREG/CR-3626 V02: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: DESCRIPTION OF MODEL CONTENT,STRUCTURE,AND SENSITIVITY TESTING.

Moodus

NUREG/CR-4354: A STUDY OF SEISMICITY AND TECTONICS IN NEW ENGLAND.Final Report.

Multiphase Flow System

NUREG/CR-3989: TIME- AND VOLUME-AVERAGED CONSERVATION EQUATIONS FOR MULTIPHASE FLOW.Part One:System Without Internal Solid Structures.

Multiple Frequency

NUREG/CR-2949 V01: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Semiannual Progress Report For Period Ending June 30,1984.

Multiple-Sequential Failure Model

NUREG/CR-3837: MULTIPLE-SEQUENTIAL FAILURE MODEL.Evaluation Of And Procedures For Human Error Dependency.

NAUA

NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).

NDE

NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984.

NDT

NUREG/CR-4092: ORNL CHARACTERIZATION OF HEAVY-SECTION STEEL TECHNOLOGY PROGRAM PLATES 01,02,AND 03.

NEPTUNUS

NUREG/CR-3919: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:NEPTUNUS PRESSURIZER TEST Y05.

NNWSI

NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984.

NPRDS

NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.

NTS Hydrogen Dewar

NUREG/CR-4138: DATA ANALYSES FOR NEVADA TEST SITE (NTS) PREMIXED COMBUSTION TESTS.

NUFREQ-NP

NUREG/CR-4116: NUFREQ-NP:A DIGITAL COMPUTER CODE FOR THE LINEAR STABILITY ANALYSIS OF BOILING WATER NUCLEAR REACTORS.

Natural Circulation
NUREG/CR-4167: FLECHT SEASET PROGRAM.Final Report.NRC/EPRI Westinghouse Report Number 16.

Natural Convection Core Cooling
NUREG/CR-3804 V03: PHYSICS OF REACTOR SAFETY.Quarterly Report,July-September 1984.
NUREG/CR-3804 V04: PHYSICS OF REACTOR SAFETY.Quarterly Report,October-December 1984.
NUREG/CR-4240 V01: PHYSICS OF REACTOR SAFETY.Quarterly Report,January-March 1985.

Neutralization
NUREG/CR-4259: TAILINGS NEUTRALIZATION AND OTHER ALTERNATIVES FOR IMMOBILIZING TOXIC MATERIALS IN TAILINGS.Final Report.

Neutron Dosimetry
NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984.
NUREG-1155 V01: RESEARCH PROGRAM PLAN.Reactor Vessels.
NUREG/CR-3609: EVALUATION OF NEUTRON DOSIMETRY TECHNIQUES FOR WELL-LOGGING OPERATIONS.

Neutron Embrittlement
NUREG/CR-4437: EXPLORATORY STUDIES OF ELEMENT INTERACTIONS AND COMPOSITION DEPENDENCIES IN RADIATION SENSITIVITY DEVELOPMENT.

Neutron Irradiation
NUREG/CR-4086: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL WELDS FOR THE THIRD HSST IRRADIATION SERIES.

Neutron Spectral Characterization
NUREG/CR-4031 V02: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutronics Calculations."
NUREG/CR-4031 V03: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutron Exposure Parameters."

Neutronics Calculation
NUREG/CR-4031 V02: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutronics Calculations."

Nevada Nuclear Waste Storage Investigation
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.REPORT FOR OCTOBER-DECEMBER 1984.

New Madrid
NUREG/CR-4333: STE. GENEVIEVE FAULT ZONE,MISSOURI AND ILLINOIS.

Noding
NUREG/CR-4044: TRAC-PF1 LOCA CALCULATIONS USING FINE-NODE AND COARSE-NODE INPUT MODELS.

Nondestructive Evaluation
NUREG-1155 V04: RESEARCH PROGRAM PLAN.Non-Destructive Examination.
NUREG/CR-3998 V02: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,April-June 1984.
NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984.
NUREG/CR-4365: DESIGN AND DEVELOPMENT OF A SPECIAL PURPOSE SAFT SYSTEM FOR NONDESTRUCTIVE EVALUATION OF NUCLEAR REACTOR VESSELS AND PIPING COMPONENTS.

Nondestructive Examination
NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984.

Nonequilibrium
NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT.

Nonnuclear Instrumentation
NUREG/CR-3991: FAILURE MODES AND EFFECTS ANALYSIS (FMEA) OF THE ICS/NNI ELECTRIC POWER DISTRIBUTION CIRCUITRY AT THE OCONEE-1 NUCLEAR PLANT.

Notch Ductility
NUREG/CR-4395: CORRELATION OF CV AND KIC/KJC TRANSITION TEMPERATURE INCREASES DUE TO IRRADIATION.
NUREG/CR-4437: EXPLORATORY STUDIES OF ELEMENT INTERACTIONS AND COMPOSITION DEPENDENCIES IN RADIATION SENSITIVITY DEVELOPMENT.

Nuclear Plant Aging
NUREG/CR-2331 V04 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July 1 -September 30,1984.

Nuclear Waste
NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984.

Nuclear-Grade Carbon
NUREG/CR-3990: CHARCOAL PERFORMANCE UNDER ACCIDENT CONDITIONS IN LIGHT-WATER REACTORS.

OCA-P
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

OPTRAN 1-1
NUREG/CR-3946: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY.

OPTRAN 1-2
NUREG/CR-3946: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY.

ORECA
NUREG/CR-3885 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, October 1-December 31,1984.

ORIGEN
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).

ORVIRT PC
NUREG/CR-4367: ORVIRT.PC:A 2-D FINITE-ELEMENT FRACTURE ANALYSIS PROGRAM FOR A MICROCOMPUTER.

Occupational Dose
NUREG/CR-4399: POSSIBLE OPTIONS FOR REDUCING OCCUPATIONAL DOSE FROM THE TMI-2 BASEMENT.

Occupational Radiation
NUREG-0713 V05: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS - 1983 ANNUAL REPORT.
NUREG/CR-4160: HISTORICAL SUMMARY OF OCCUPATIONAL RADIATION EXPOSURE EXPERIENCE IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.

Occurrence
NUREG-0090 V07 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1984.

Off-Normal Event
NUREG/CR-4168: GT2F:A COMPUTER CODE FOR ESTIMATING LIGHT WATER REACTOR FUEL ROD FAILURES.

Official Record
NUREG-0910 R01 S03: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE.

Official Records
NUREG-0910 R01 S02: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE.
NUREG-0910 R01 S04: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE.

Offshore Radioactive Material
NUREG/CR-4190: CALIFORNIA OFFSHORE SURVEY OF LICENSEES USING RADIOACTIVE MATERIAL.

Ohio
NUREG/CR-3145 V03: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION - ANNUAL REPORT.(October 1982 - September 1983, Volume 3).

Operating Experience
NUREG/CR-3430 V02: NUCLEAR POWER PLANT OPERATING EXPERIENCE - 1982 Annual Report.
NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE.
NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.
NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.

Operating Procedures
NUREG-0800 13.5.2 R1: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 1 To Section 13.5.2, "Operating And Maintenance Procedures," and Revision 0 of Appendix A to Section 13.5.2, "Review..."

Operating Reactors Licensing Actions
NUREG-0748 V04 N12: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of December 31,1984 (Orange Book)
NUREG-0748 V05 N01: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of January 31,1985 (Orange Book)
NUREG-0748 V05 N02: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of February 28,1985.(Orange Book)
NUREG-0748 V05 N03: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of March 31,1985. (Orange Book)
NUREG-0748 V05 N04: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of April 30,1985.(Orange Book)
NUREG-0748 V05 N05: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of May 31,1985. (Orange Book)
NUREG-0748 V05 N06: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of June 30,1985. (Orange Book)
NUREG-0748 V05 N07: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of July 31,1985.(Orange Book)
NUREG-0748 V05 N08: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of August 31,1985. (Orange Book)
NUREG-0748 V05 N10: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of October 31,1985. (Orange Book)

Operational Proceaures
NUREG-0981 R01: NRC/FEMA OPERATIONAL RESPONSE PROCEDURES FOR RESPONSE TO A COMMERCIAL NUCLEAR REACTOR ACCIDENT.

Operations Personnel
NUREG/CR-4139: THE MAILED SURVEY:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL.

Operator
NUREG/CR-4206: A SELECT REVIEW OF THE RECENT (1979-1983) BEHAVIORAL RESEARCH LITERATURE ON TRAINING SIMULATORS.

Operator Error
NUREG-1154: LOSS OF MAIN AND AUXILIARY FEEDWATER EVENT AT THE DAVIS-BESSE PLANT ON JUNE 9, 1985.

Operator Feedback Project
NUREG/CR-4139: THE MAILED SURVEY:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL.

Operator Training
NUREG-0090 V08 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1985.
NUREG/CR-4139: THE MAILED SURVEY:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL.

Organic Complexant
NUREG/CR-4435: ORGANIC COMPLEXANT-ENHANCED MOBILITY OF TOXIC ELEMENTS IN LOW-LEVEL WASTES.Annual Report,July 1984 - June 1985.

Organization
NUREG/CR-4125 V01: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 1:Guidelines For Utility Organization And Administration Plan.
NUREG/CR-4125 V02: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 2:Workbook For Assessment Of Organization And Management.

Organization Chart
NUREG-0325 R07: U.S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS.

Overcooling
NUREG/CR-3935: THERMAL-HYDRAULIC ANALYSES OF OVERCOOLING SEQUENCES FOR THE H.B. ROBINSON UNIT 2 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

Overcooling Accident
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.

Overexposure
NUREG-0090 V07 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1984.

Overpack
NUREG/CR-3900 V04: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Annual Report,April 1984 - April 1985.
NUREG/CR-4379 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report,Year Four April-June 1985.

Overpressurization
NUREG/CR-4137: PRETEST PREDICTIONS FOR THE RESPONSE OF A 1:8-SCALE STEEL LWR CONTAINMENT BUILDING MODEL TO STATIC OVERPRESSURIZATION.
NUREG/CR-4209: COMPARISON OF ANALYTICAL PREDICTIONS AND EXPERIMENTAL RESULTS FOR A 1:8-SCALE STEEL CONTAINMENT MODEL PRESSURIZED TO FAILURE.

Overtime
NUREG/CR-4248: RECOMMENDATIONS FOR NRC POLICY ON SHIFT SCHEDULING AND OVERTIME AT NUCLEAR POWER PLANTS.

Oxygen Diffusion
NUREG/CR-4358: APPLICATIONS OF DENSITY PROFILING TO EQUIPMENT QUALIFICATION ISSUES.

P-Wave Residuals
NUREG/CR-3145 V03: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION - ANNUAL REPORT.(October 1982 - September 1983, Volume 3).

PFR
NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.

PHREEQE
NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.REPORT FOR OCTOBER-DECEMBER 1984.

NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.

PRA
NUREG-1115: CATEGORIZATION OF REACTOR SAFETY ISSUES FROM A RISK PERSPECTIVE.
NUREG/CP-0070: PROCEEDINGS OF THE WORKSHOP ON SEISMIC AND DYNAMIC FRAGILITY OF NUCLEAR POWER PLANT COMPONENTS.
NUREG/CR-3301: CATALOG OF PRA DOMINANT ACCIDENT SEQUENCE INFORMATION.
NUREG/CR-3485: PRA REVIEW MANUAL.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4050: A REVIEW OF THE SHOREHAM NUCLEAR POWER STATION PROBABILISTIC RISK ASSESSMENT.Internal Events And Core Damage Frequency.
NUREG/CR-4103: USES OF HUMAN RELIABILITY ANALYSIS PROBABILISTIC RISK ASSESSMENT RESULTS TO RESOLVE PERSONNEL PERFORMANCE ISSUES THAT COULD AFFECT SAFETY.
NUREG/CR-4331: SIMPLIFIED SEISMIC PROBABILISTIC RISK ASSESSMENT.Procedures And Limitations.
NUREG/CR-4350 V01: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 1 - PRA Fundamentals.
NUREG/CR-4350 V02: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 2-Probability And Statistics For PRA Applications.
NUREG/CR-4377: EVALUATIONS AND UTILIZATIONS OF RISK IMPORTANCES.
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP).A Methodology For Estimating Risk Impacts Of Safeguards Measures.

PTS
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4109: TRAC-PF1 ANALYSES OF POTENTIAL PRESSURIZED-THERMAL SHOCK TRANSIENTS AT CALVERT CLIFFS/UNIT 1.A Combustion Engineering PWR.
NUREG/CR-4253: REVIEW OF TRAC CALCULATIONS FOR CALVERT CLIFFS PTS STUDY.
NUREG/CR-4376: HEAT TRANSFER,CARRYOVER AND FALL BACK IN PWR STEAM GENERATORS DURING TRANSIENTS.

PVC
NUREG/CR-4008: GENERAL EXTRAPOLATION MODEL FOR AN IMPORTANT CHEMICAL DOSE-RATE EFFECT.

PWR-GALE
NUREG-0017 R01: CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM PRESSURIZED WATER REACTORS (PWR-GALE CODE).

Packaging
NUREG-0383 V01 R08: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Packages.
NUREG-0383 V02 R08: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Certificates of Compliance.
NUREG-0383 V03 R05: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Quality Assurance Programs For Radioactive Material Packages.

Packaging Material
NUREG/CR-2482 V06: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program.October 1983 - March 1984.
NUREG/CR-3091 V04: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1983 - March 1984.
NUREG/CR-3091 V05: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period April 1984 - September 1984.

Panel Session
NUREG/CP-0065: TRANSACTIONS OF THE 8TH INTERNATIONAL CONFERENCE ON STRUCTURE MECHANICS IN REACTOR TECHNOLOGY.Panel Session J-K. Status of Research In Structural And Mechanical Engineering For Nuclear Power Plants.

Pellet-Cladding
NUREG/CR-3810 V03: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,July-September 1984.
NUREG/CR-3999: ELECTRICALLY HEATED EX-REACTOR PELLET-CLADDING INTERACTION (PCI) SIMULATIONS UTILIZING IRRADIATED ZIRCALOY CLADDING.

Penetration
NUREG/CR-3855: CHARACTERIZATION OF NUCLEAR REACTOR CONTAINMENT PENETRATION - FINAL REPORT.
NUREG/CR-4064: STRUCTURAL RESPONSE OF LARGE PENETRATIONS AND CLOSURES FOR CONTAINMENT VESSELS SUBJECTED TO LOADINGS BEYOND DESIGN BASIS.
NUREG/CR-4119: INTEGRITY OF CONTAINMENT PENETRATIONS UNDER SEVERE ACCIDENT CONDITIONS FY84 ANNUAL REPORT.

Perforated Plate
NUREG/CR-4415: COUNTER-CURRENT STEAM/WATER FLOW ABOVE A PERFORATED PLATE-VERTICAL INJECTION OF WATER.

Performance Evaluation
NUREG/CR-4344: INSTRUCTIONAL SKILLS EVALUATION IN NUCLEAR INDUSTRY TRAINING.

Performance Indicators
NUREG/CR-4281: AN EMPIRICAL ANALYSIS OF SELECTED NUCLEAR POWER PLANT MAINTENANCE FACTORS AND PLANT SAFETY.

Performance Simulation
NUREG/CR-4104: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL. Field Evaluation/Validation.

Performance-Based Training
NUREG/CR-4439: TRAINING REVIEW CRITERIA AND PROCEDURES.

Permeameter Testing
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

Personal Air Sampling
NUREG/CR-4033: THE ROLE OF PERSONAL AIR SAMPLING IN RADIATION SAFETY PROGRAMS AND RESULTS OF A LABORATORY EVALUATION OF PERSONAL AIR-SAMPLING EQUIPMENT.

Personnel Monitoring
NUREG/CR-4180: HISTORICAL SUMMARY OF OCCUPATIONAL RADIATION EXPOSURE EXPERIENCE IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.

Personnel Performance
NUREG/CR-4103: USES OF HUMAN RELIABILITY ANALYSIS PROBABILISTIC RISK ASSESSMENT RESULTS TO RESOLVE PERSONNEL PERFORMANCE ISSUES THAT COULD AFFECT SAFETY.

Petitions For Rulemaking
NUREG-0936 V03 N04: NRC REGULATORY AGENDA.Quarterly Report,October-December 1984.
NUREG-0936 V04 N01: NRC REGULATORY AGENDA.Quarterly Report,January-March 1985.
NUREG-0936 V04 N03: NRC REGULATORY AGENDA.Quarterly Report,July-September 1985.

Phenix
NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.

Phosphorus-32
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.

Physical Benchmark
NUREG/CR-4291: CONCLUSION AND SUMMARY REPORT ON PHYSICAL BENCHMARKING OF PIPING SYSTEMS.

Physical Protection
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP).A Methodology For Estimating Risk Impacts Of Safeguards Measures.

Physical Security
NUREG/CR-4298: DESIGN AND INSTALLATION OF COMPUTER SYSTEMS TO MEET THE REQUIREMENTS OF 10 CFR 73.55.

Physics-Dosimetry
NUREG/CR-3319: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM:LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium.

Pipe Crack
NUREG-1061 V05: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 5:Summary - Piping Review Committee Conclusions and Recommendations.

Pipe Failure
NUREG/CR-3660 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 1:Summary Report.
NUREG/CR-3660 V04: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 4:Pipe Failure Induced By Crack Growth In West Coast Plants.
NUREG/CR-3663 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 1:Summary Report.
NUREG/CR-4290 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF BABCOCK AND WILCOX PWR PLANTS.Volume 2:Guillotine Break Indirectly Induced By Earthquakes.

Pipe Support
NUREG/CR-4263: RELIABILITY ANALYSIS OF STIFF VERSUS FLEXIBLE PIPING.FINAL PROJECT REPORT.

Pipe Whip Restraint
NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-4263: RELIABILITY ANALYSIS OF STIFF VERSUS FLEXIBLE PIPING.FINAL PROJECT REPORT.

Piping
NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984.
NUREG-1061 V02: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2:Evaluation Of Seismic Designs - A Review Of Seismic Design Requirements For Nuclear Power Plant Piping.
NUREG-1061 V02 ADD: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2 Addendum:Summary And Evaluation Of Historical Strong-Motion Earthquake Seismic Response And Damage To Aboveground Industrial Piping.
NUREG-1061 V05: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 5:Summary - Piping Review Committee Conclusions and Recommendations.
NUREG-1155 V03: RESEARCH PROGRAM PLAN.Piping.
NUREG/CR-1677 V02: PIPING BENCHMARK PROBLEMS.VOLUME II DYNAMIC ANALYSIS INDEPENDENT SUPPORT MOTION RESPONSE SPECTRUM METHOD.
NUREG/CR-3613 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Annual Report For 1984.
NUREG/CR-3613 V03 N1: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Semiannual Report For October 1984 Through March 1985.
NUREG/CR-3663 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 3:Double Ended Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3945: FATIGUE CRACK GROWTH RATES OF LOW-CARBON AND STAINLESS PIPING STEELS IN PWR ENVIRONMENT.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.
NUREG/CR-4291: CONCLUSION AND SUMMARY REPORT ON PHYSICAL BENCHMARKING OF PIPING SYSTEMS.
NUREG/CR-4291: CONCLUSION AND SUMMARY REPORT ON PHYSICAL BENCHMARKING OF PIPING SYSTEMS.
NUREG/CR-4305: COMMENTS ON THE LEAK-BEFORE-BREAK CONCEPT FOR NUCLEAR POWER PLANT PIPING SYSTEMS.
NUREG/CR-4365: DESIGN AND DEVELOPMENT OF A SPECIAL PURPOSE SAFT SYSTEM FOR NONDESTRUCTIVE EVALUATION OF NUCLEAR REACTOR VESSELS AND PIPING COMPONENTS.
NUREG/CR-4422: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN PWR ENVIRONMENTS.

Piping Reliability
NUREG-1147: SEISMIC SAFETY RESEARCH PROGRAM PLAN.

Piping System
NUREG/CR-4221: AN EVALUATION OF STRESS CORROSION CRACK GROWTH IN BWR PIPING SYSTEMS.

Planning Guidance
NUREG-0885 I04: U.S. NUCLEAR REGULATORY COMMISSION POLICY AND PLANNING GUIDANCE 1985.

Plant Analyzer
NUREG/CR-3943: THE BWR PLAN ANALYZER.

Plant Safety
NUREG/CR-3943: THE BWR PLAN ANALYZER.
NUREG/CR-4281: AN EMPIRICAL ANALYSIS OF SELECTED NUCLEAR POWER PLANT MAINTENANCE FACTORS AND PLANT SAFETY.
NUREG/CR-4400: THE IMPACT OF MECHANICAL-AND MAINTENANCE-INDUCED FAILURES OF MAIN REACTOR COOLANT PUMP SEALS ON PLANT SAFETY.

Plugging
NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

Plutonium
NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS.Application To Establishing Values Of f1 For Soluble Plutonium.
NUREG/CR-4355 V01: 238 PU(IV) IN MONKEYS.Overview Of Metabolism.

Policy
NUREG-1070: NRC POLICY ON FUTURE REACTOR DESIGNS.Decisions On Severe Accident Issues In Nuclear Power Plant Regulation.

Policy And Planning
NUREG-0885 I04: U.S. NUCLEAR REGULATORY COMMISSION POLICY AND PLANNING GUIDANCE 1985.

Polyvinylchloride
NUREG/CR-4008: GENERAL EXTRAPOLATION MODEL FOR AN IMPORTANT CHEMICAL DOSE-RATE EFFECT.

Pond Surface Heat Transfer
NUREG/CR-4120: MATHEMATICAL MODELING OF ULTIMATE HEAT SINK COOLING PONDS.

Population
NUREG/CR-2850 V03: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1981.

Post-Accident Cleanup
NUREG/CR-3293 V01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Main Report.
NUREG/CR-3293 V02: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Appendices.

Post-LOCA Condition
NUREG/CR-3803: THE EFFECTS OF POST-LOCA CONDITIONS ON A PROTECTIVE COATING (PAINT) FOR THE NUCLEAR POWER INDUSTRY.

Post-LOCA Hydrogen Generation
NUREG/CR-3361: THE EFFECT OF WATER CHEMISTRY ON THE RATES OF HYDROGEN GENERATION FROM GALVANIZED STEEL CORROSION AT POST-LOCA CONDITIONS.

Postcritical Heat Flux
NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT.

Postirradiation
NUREG/CR-3810 V03: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,July-September 1984.

Postirradiation Examination
NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report.October-December 1984.
NUREG/CR-4218: LOCA SIMULATION IN THE NATIONAL RESEARCH UNIVERSAL REACTOR PROGRAM.Postirradiation Examination Results For The Third Materials Test (MT-3) - Second Campaign.
NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report.January-March 1985.

Postprocessor Manual
NUREG/CR-4396: SIMMER POSTPROCESSOR MANUAL.

Postulated UF(6)
NUREG/CR-4360 V01: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.
NUREG/CR-4360 V02: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.

Power Supply
NUREG/CR-3991: FAILURE MODES AND EFFECTS ANALYSIS (FMEA) OF THE ICS/NNI ELECTRIC POWER DISTRIBUTION CIRCUITRY AT THE OCONEE-1 NUCLEAR PLANT.
NUREG/CR-4256: MEASUREMENT OF RESPONSE TIME AND DETECTION OF DEGRADATION IN PRESSURE SENSOR/SENSING LINE SYSTEMS.

Practice And Procedures Digest
NUREG-0386 D03: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.JULY 1972 - SEPTEMBER 1983.

Precipitation
NUREG/CR-4157: A SCIENTIFIC CRITIQUE OF AVAILABLE MODELS FOR REAL-TIME SIMULATIONS OF DISPERSION.

Precipitation Scavenging
NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING.

Preliminary Strategic Data Plan
NUREG/CR-4322 V04: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 4: Preliminary Strategic Data Plan.

Premixed Combustion Experiment
NUREG/CR-4138: DATA ANALYSES FOR NEVADA TEST SITE (NTS) PREMIXED COMBUSTION TESTS.

Pressure Boundary
NUREG/CR-3915: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL.
NUREG/CR-4305: COMMENTS ON THE LEAK-BEFORE-BREAK CONCEPT FOR NUCLEAR POWER PLANT PIPING SYSTEMS.

Pressure Boundary Component
NUREG/CR-3228 V03: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1984.

Pressure Capacity
NUREG/CR-4149: ULTIMATE PRESSURE CAPACITY OF REINFORCED AND PRESTRESSED CONCRETE CONTAINMENT.

Pressure Differential
NUREG/CR-4112 V02: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task B:Firestop Test Method.

Pressure Probability Theory
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.

Pressure Sensor Systems
NUREG/CR-4256: MEASUREMENT OF RESPONSE TIME AND DETECTION OF DEGRADATION IN PRESSURE SENSOR/SENSING LINE SYSTEMS.

Pressure Suppression Pool
NUREG/CR-3764: BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE.

Pressure Transient
NUREG/CR-4225: SUMMARY OF EFFICIENCY TESTING OF STANDARD AND HIGH-CAPACITY HIGH-EFFICIENCY PARTICULATE AIR FILTERS SUBJECTED TO SIMULATED TORNADO DEPRESSURIZATION AND EXPLOSIVE SHOCK WAVES.

Pressure Transmitter
NUREG/CR-3863: ASSESSMENT OF CLASS 1E PRESSURE TRANSMITTER RESPONSE WHEN SUBJECTED TO HARSH ENVIRONMENT-SCREENING TESTS.

Pressure Vessel
NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984.
NUREG-1155 V01: RESEARCH PROGRAM PLAN.Reactor Vessels.
NUREG/CR-3319: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium.
NUREG/CR-3647: DESIGN AND FABRICATION OF A 1/8-SCALE STEEL CONTAINMENT MODEL.
NUREG/CR-3723: STRESS-INTENSITY-FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS.
NUREG/CR-3744 V02: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR APRIL-SEPTEMBER 1984.
NUREG/CR-3746 V02: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.Semiannual Progress Report.April 1984 - September 1984.
NUREG/CR-3746 V03: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.1984 Annual Report.October 1,1983 - September 30, 1984.
NUREG/CR-3825 V03-4: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Quarterly Report, April 1984 - September 1984.Volumes 3 and 4.
NUREG/CR-3915: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL.
NUREG/CR-3978: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL PLATE AND WELDS FOR THE FOURTH HSST IRRADIATION SERIES.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4086: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL WELDS FOR THE THIRD HSST IRRADIATION SERIES.
NUREG/CR-4092: ORNL CHARACTERIZATION OF HEAVY-SECTION STEEL TECHNOLOGY PROGRAM PLATES 01,02,AND 03.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.
NUREG/CR-4212: IN-PLACE THERMAL ANNEALING OF NUCLEAR REACTOR PRESSURE VESSELS.
NUREG/CR-4219 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.
NUREG/CR-4249: PRESSURE VESSEL FRACTURE STUDIES PENETRATING TO THE PWR THERMAL-SHOCK ISSUE:EXPERIMENTS TSE-5,TSE-5A AND TSE-6.
NUREG/CR-4275: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM - FIVE-YEAR PLAN FY 1984-1988.
NUREG/CR-4284: NEUTRON EXPOSURE PARAMETERS FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY IRRADIATION SERIES.
NUREG/CR-4304: PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO THE PWR THERMAL-SHOCK ISSUE.Experiment TSE-7.
NUREG/CR-4325: A PARAMETRIC STUDY OF PWR PRESSURE VESSEL INTEGRITY DURING OVERCOOLING ACCIDENTS,CONSIDERING BOTH 2-D AND 3-D FLAWS.
NUREG/CR-4325: A PARAMETRIC STUDY OF PWR PRESSURE VESSEL INTEGRITY DURING OVERCOOLING ACCIDENTS,CONSIDERING BOTH 2-D AND 3-D FLAWS.
NUREG/CR-4422: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN PWR ENVIRONMENTS.
NUREG/CR-4437: EXPLORATORY STUDIES OF ELEMENT INTERACTIONS AND COMPOSITION DEPENDENCIES IN RADIATION SENSITIVITY DEVELOPMENT.

Pressuremeter
NUREG/CR-4430: CURRENT METHODOLOGIES FOR ASSESSING THE POTENTIAL FOR EARTHQUARKE-INDUCED LIQUEFACTION IN SOILS.

Pressurized Thermal Shock
NUREG-1155 V01: RESEARCH PROGRAM PLAN.Reactor Vessels.
NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report.April 1-June 30,1984.

NUREG/CR-2331 V04 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July 1 - September 30,1984.
NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-3723: STRESS-INTENSITY-FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS.
NUREG/CR-3935: THERMAL-HYDRAULIC ANALYSES OF OVERCOOLING SEQUENCES FOR THE H.B. ROBINSON UNIT 2 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper Shelf Arrest.
NUREG/CR-4109: TRAC-PF1 ANALYSES OF POTENTIAL PRESSURIZED-THERMAL SHOCK TRANSIENTS AT CALVERT CLIFFS/UNIT 1.A Combustion Engineering PWR.
NUREG/CR-4267: VESSEL INTEGRITY SIMULATION (VISA) CODE SENSITIVITY STUDY.
NUREG/CR-4325: A PARAMETRIC STUDY OF PWR PRESSURE VESSEL INTEGRITY DURING OVERCOOLING ACCIDENTS,CONSIDERING BOTH 2-D AND 3-D FLAWS.
NUREG/CR-4376: HEAT TRANSFER,CARRYOVER AND FALL BACK IN PWR STEAM GENERATORS DURING TRANSIENTS.

Preventive Maintenance
NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE.

Price-Anderson Retrospective Premium
NUREG-1131: FINANCIAL ANALYSIS OF POTENTIAL RETROSPECTIVE PREMIUM ASSESSMENTS UNDER THE PRICE-ANDERSON SYSTEM.

Primary Coolant Circuit
NUREG/CR-4294: LEAK RATE ANALYSIS OF THE WESTINGHOUSE REACTOR COOLANT PUMP.

Primary Coolant System
NUREG/CR-4151: INTEGRATION OF EMERGENCY ACTION LEVELS WITH COMBUSTION ENGINEERING EMERGENCY OPERATING PROCEDURES.By Use Of Combustion Engineering Owners Group Emergency Operating Procedure Technical Guidelines.

Primates
NUREG/CR-4355 V01: 238 PU(IV) IN MONKEYS.Overview Of Metabolism.

Prioritization
NUREG-1115: CATEGORIZATION OF REACTOR SAFETY ISSUES FROM A RISK PERSPECTIVE.
NUREG/CR-2800 S03: GUIDELINES FOR NUCLEAR POWER PLANT SAFETY ISSUE PRIORITIZATION INFORMATION DEVELOPMENT.

Probabilistic Fracture Mechanics
NUREG/CR-3660 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 1:Summary Report.

Probabilistic Risk Analysis
NUREG-1115: CATEGORIZATION OF REACTOR SAFETY ISSUES FROM A RISK PERSPECTIVE.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-2815 V02 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 8-12.
NUREG/CR-4140: DOMINANT ACCIDENT SEQUENCES IN OCONEE-1 PRESSURIZED WATER REACTOR.
NUREG/CR-4213: SETS REFERENCE MANUAL.
NUREG/CR-4377: EVALUATIONS AND UTILIZATIONS OF RISK IMPORTANCES.
NUREG/CR-4386: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A BABCOCK AND WILCOX PRESSURIZED WATER REACTOR.
NUREG/CR-4387: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A GENERAL ELECTRIC PRESSURIZED WATER REACTOR.

Probabilistic Risk Assessment
NUREG-1148: NUCLEAR POWER PLANT FIRE PROTECTION RESEARCH PROGRAM.
NUREG/CR-3301: CATALOG OF PRA DOMINANT ACCIDENT SEQUENCE INFORMATION.
NUREG/CR-3485: PRA REVIEW MANUAL.
NUREG/CR-3519: HUMAN ERROR PROBABILITY ESTIMATION USING LICENSEE EVENT REPORTS.
NUREG/CR-3634: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: USER'S MANUAL.
NUREG/CR-3837: MULTIPLE-SEQUENTIAL FAILURE MODEL.Evaluation Of And Procedures For Human Error Dependency.
NUREG/CR-3862: DEVELOPMENT OF TRANSIENT INITIATING EVENT FREQUENCIES FOR USE IN PROBABILISTIC RISK ASSESSMENTS.
NUREG/CR-4009: HUMAN RELIABILITY DATA BANK.Evaluation Results.
NUREG/CR-4010: SPECIFICATION OF A HUMAN RELIABILITY DATA BANK FOR CONDUCTING HRA SEGMENTS OF PRAS FOR NUCLEAR POWER PLANTS.
NUREG/CR-4050: A REVIEW OF THE SHOREHAM NUCLEAR POWER STATION PROBABILISTIC RISK ASSESSMENT.Internal Events And Core Damage Frequency.
NUREG/CR-4103: USES OF HUMAN RELIABILITY ANALYSIS PROBABILISTIC RISK ASSESSMENT RESULTS TO RESOLVE PERSONNEL PERFORMANCE ISSUES THAT COULD AFFECT SAFETY.
NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences.
NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.
NUREG/CR-4153: APPLICATIONS OF FOREIGN PROBABILISTIC SAFETY ASSESSMENT EXPERIENCE TO THE U.S. NUCLEAR REGULATORY PROCESS.
NUREG/CR-4229: EVALUATION OF CURRENT METHODOLOGY EMPLOYED IN PROBABILISTIC RISK ASSESSMENT (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.
NUREG/CR-4231: EVALUATION OF AVAILABLE DATA FOR PROBABILISTIC RISK ASSESSMENTS (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.
NUREG/CR-4331: SIMPLIFIED SEISMIC PROBABILISTIC RISK ASSESSMENT.Procedures And Limitations.
NUREG/CR-4334: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS.
NUREG/CR-4350 V01: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 1 - PRA Fundamentals.
NUREG/CR-4350 V02: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 2-Probability And Statistics For PRA Applications.
NUREG/CR-4350 V03: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 3 - Systems Reliability And Analysis Techniques.Session A - Reliability.
NUREG/CR-4350 V04: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 4 - System Reliability And Analysis Techniques.Sessions B/C - Event Trees/Fault Trees.
NUREG/CR-4350 V05: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 5 - Systems Reliability And Analysis Techniques.Session D - Quantification.
NUREG/CR-4350 V06: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 6 - Data Development.
NUREG/CR-4350 V07: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 7 - Environmental Transport And Consequence Analysis.
NUREG/CR-4383: HIGH PRESSURE INJECTION OF MELT FROM A REACTOR PRESSURE VESSEL - THE DISCHARGE PHASE.
NUREG/CR-4385: EFFECTS OF CONTROL SYSTEM FAILURES IN TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A WESTINGHOUSE PRESSURIZED WATER REACTOR.

Probabilistic Safety Analysis
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-2815 V02 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 8-12.

Probability Assessment
NUREG/CR-3688 V01: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 1:Main Report.
NUREG/CR-3688 V02: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 2:Appendices.

Procedure
NUREG/CR-3817: DEVELOPMENT,USE AND CONTROL OF MAINTENANCE PROCEDURES IN NUCLEAR POWER PLANTS.Problems And Recommendations.

Progression
NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report.January-March 1985.

Project Review
NUREG-1125 V01: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 1,Part 1:ACRS Reports On Project Reviews (A-F).
NUREG-1125 V02: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 2,Part 1:ACRS Reports On Project Reviews (G-P).

Protective Cover
NUREG/CR-4075: DESIGNING PROTECTIVE COVERS FOR URANIUM MILL TAILINGS PILES: A Review.

Prototype Fast Reactor
NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.

Protracted Materials Losses
NUREG/CR-4107: SEQUENTIAL TEST PROCEDURES FOR DETECTING PROTRACTED MATERIALS LOSSES.

Psychological Stress
NUREG/CR-4040: OPERATIONAL DECISIONMAKING AND ACTION SELECTION UNDER PSYCHOLOGICAL STRESS IN NUCLEAR POWER PLANTS.

Public Safety
NUREG/CR-4153: APPLICATIONS OF FOREIGN PROBABILISTIC SAFETY ASSESSMENT EXPERIENCE TO THE U.S. NUCLEAR REGULATORY PROCESS.

Puff Model
NUREG/CR-4159: COMPARISON OF THE 1981 INEL DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS.

Pump
NUREG/CR-3914: PUMP AND VALVE QUALIFICATION REVIEW GUIDE.
NUREG/CR-4228: REVIEW OF THE VOGTLE UNITS 1 AND 2 AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS.

Purge
NUREG/CR-4141: CONTAINMENT PURGE AND VENT VALVE TEST PROGRAM FINAL REPORT.

QUICKM
NUREG/CR-4388: AEROSOL BEHAVIOR MODELING (TASK 3) - SUPPORT SERVICES FOR RESEARCH AND EVALUATION OF SEVERE ACCIDENT PHENOMENA AND MITIGATION.

Qualifications
NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.

Quality Assurance
NUREG-0940 V04 N02: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report.April-June,1985.
NUREG/CR-4271: RECOMMENDED SAFETY,RELIABILITY,QUALITY ASSURANCE AND MANAGEMENT AEROSPACE TECHNIQUES WITH POSSIBLE APPLICATION BY THE DOE TO THE HIGH LEVEL RADIOACTIVE WASTE REPOSITORY PROGRAM.

Quantification
NUREG/CR-4350 V05: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 5 - Systems Reliability And Analysis Techniques.Session D - Quantification.

RADTWO
NUREG/CR-3442: RADTWO:A COMPUTER CODE FOR SIMULATING FAST-TRANSIENT, TWO-DIMENSIONAL,TWO-LAYER RADIONUCLIDE CONCENTRATION CONDITIONS IN LAKES,RESERVOIRS,RIVERS,ESTUARIES,AND COASTAL REGIONS.

RAMONA-3B
NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report.April 1-June 30,1984.
NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report.January 1-March 31,1985.

RCL
NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes.

RELAP5
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-3772: RELAP5 ASSESSMENT:SEMISCALE SMALL BREAK TESTS S-UT-1,S-UT-2, S-UT-6,S-UT-7 AND S-UT-8.
NUREG/CR-3802: RELAP5 ASSESSMENT:QUANTITATIVE KEY PARAMETERS AND RUN TIME STATISTICS.
NUREG/CR-3936: RELAP5 ASSESSMENT:CONCLUSIONS AND USER GUIDELINES.
NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR.
NUREG/CR-4046: DETERMINING CRITICAL FLOW VALVE CHARACTERISTICS USING EXTRAPOLATION TECHNIQUES.

RELAP5/MOD1
NUREG/CR-3936: RELAP5 ASSESSMENT:CONCLUSIONS AND USER GUIDELINES.
NUREG/CR-4292: A COMPARATIVE ANALYSIS OF CONSTITUTIVE RELATIONS IN TRAC-PFL AND RELAP5/MOD1.

REMIX
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

RPA
NUREG-1125 V06: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 6,Part 2:ACRS Reports On Generic Subjects (RPA - Appendix C).

Radiation
NUREG-0837 V05 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, January-March 1985.
NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report.April 1983 - March 1984.
NUREG/CR-4033: THE ROLE OF PERSONAL AIR SAMPLING IN RADIATION SAFETY PROGRAMS AND RESULTS OF A LABORATORY EVALUATION OF PERSONAL AIR-SAMPLING EQUIPMENT.
NUREG/CR-4039: GAMMA-RAY CHARACTERIZATION OF THE TWO-YEAR IRRADIATION EXPERIMENT PERFORMED AT THE POOLSIDE FACILITY.
NUREG/CR-4147: THE EFFECT OF ENVIRONMENTAL STRESS ON SYLGARD 70 SILICONE ELASTOMER.

Radiation Effect
NUREG/CR-3514 V02: THE CHEMICAL BEHAVIOR OF IODINE IN AQUEOUS SOLUTIONS UP TO 150 C.II.Radiation-Redox Conditions.

Radiation Exposure
NUREG/CR-4090: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Annual Summary Report - Fiscal Year 1984.

Radiation Monitoring
NUREG-0837 V05 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, April-June 1985.
NUREG/CR-4160: HISTORICAL SUMMARY OF OCCUPATIONAL RADIATION EXPOSURE EXPERIENCE IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.

Radiation Protection
NUREG/CP-0066: PROCEEDINGS OF AN INTERNATIONAL WORKSHOP ON HISTORIC DOSE EXPERIENCE AND DOSE REDUCTION (ALARA) AT NUCLEAR POWER PLANTS,MAY 29-JUNE 1,1984.
NUREG/CR-3469 V02: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS. Annotated Bibliography Of Selected Readings In Radiation Protection And ALARA.
NUREG/CR-4033: THE ROLE OF PERSONAL AIR SAMPLING IN RADIATION SAFETY PROGRAMS AND RESULTS OF A LABORATORY EVALUATION OF PERSONAL AIR-SAMPLING EQUIPMENT.

NUREG/CR-4254: OCCUPATIONAL DOSE REDUCTION AND ALARA AT NUCLEAR POWER PLANTS.Study On High-Dose Jobs,Radwaste Handling,And ALARA Incentives.
NUREG/CR-4373: COMPENDIUM OF COST-EFFECTIVENESS EVALUATIONS OF MODIFICATIONS FOR DOSE REDUCTION AT NUCLEAR POWER PLANTS.

Radiation Protection Training
NUREG-1127: RADIATION PROTECTION TRAINING AT URANIUM HEXAFLUORIDE AND FUEL FABRICATION PLANTS.
NUREG-1134: RADIATION PROTECTION TRAINING FOR PERSONNEL EMPLOYED IN MEDICAL FACILITIES.

Radiation Safety Program
NUREG-0940 V04 N02: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June,1985.

Radiation Safety Survey
NUREG/CR-4190: CALIFORNIA OFFSHORE SURVEY OF LICENSEES USING RADIOACTIVE MATERIAL.

Radiation Sensitivity
NUREG/CR-3228 V03: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1984.
NUREG/CR-4437: EXPLORATORY STUDIES OF ELEMENT INTERACTIONS AND COMPOSITION DEPENDENCIES IN RADIATION SENSITIVITY DEVELOPMENT.

Radiation Stability
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.

Radioactive Emission
NUREG/CR-4068: SUMMARY OF HISTORICAL EXPERIENCE WITH RELEASES OF RADIOACTIVE MATERIALS FROM COMMERCIAL NUCLEAR POWER PLANTS IN THE UNITED STATES.

Radioactive Gas
NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA.

Radioactive Iodine
NUREG/CR-3990: CHARCOAL PERFORMANCE UNDER ACCIDENT CONDITIONS IN LIGHT-WATER REACTORS.

Radioactive Isotopes
NUREG/CR-3865: EVALUATION OF THE RADIOACTIVE INVENTORY IN,AND ESTIMATION OF ISOTOPIC RELEASE FROM,THE WASTE IN EIGHT TRENCHES AT THE SHEFFIELD LOW-LEVEL WASTE BURIAL SITE.

Radioactive Material
NUREG-0383 V01 R08: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Packages.
NUREG-0383 V02 R08: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Certificates of Compliance.
NUREG-0383 V03 R05: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Quality Assurance Programs For Radioactive Material Packages.
NUREG/CR-3019: RECOMMENDED WELDED CRITERIA FOR USE IN THE FABRICATION OF SHIPPING CONTAINERS FOR RADIOACTIVE MATERIALS.
NUREG/CR-3518: A SURVEY OF THE USES OF RADIOACTIVE MATERIALS IN LOUISIANA'S OFFSHORE WATERS.
NUREG/CR-3611: RADIOACTIVE MATERIAL (RAM) ACCIDENT/INCIDENT DATA ANALYSIS PROGRAM.
NUREG/CR-3657: PRELIMINARY SCREENING OF FUEL CYCLE AND BY-PRODUCT MATERIAL LICENSES FOR EMERGENCY PLANNING.
NUREG/CR-3854: FABRICATION CRITERIA FOR SHIPPING CONTAINERS.
NUREG/CR-4357: THE FEASIBILITY OF DETECTING THE IMPORT OF UNAUTHORIZED RADIOACTIVE MATERIALS INTO THE UNITED STATES.

Radioactive Release
NUREG/CR-2850 V03: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1981.

Radioactive Waste
NUREG-0017 R01: CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM PRESSURIZED WATER REACTORS (PWR-GALE CODE).
NUREG-0946: AN EVALUATION OF RADIONUCLIDE CONCENTRATIONS IN HIGH-LEVEL RADIOACTIVE WASTES.
NUREG-1164: INFORMATION ON THE CONFINEMENT CAPABILITY OF THE FACILITY DISPOSAL AREA AT WEST VALLEY,NEW YORK.
NUREG/CR-1755 ADD01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING NUCLEAR REACTORS AT MULTIPLE-REACTOR STATIONS.Effects On Decommissioning Of Interim Inability To Dispose Of Wastes Offsite.
NUREG/CR-3413: OFF-SITE CONSEQUENCES OF RADIOLOGICAL ACCIDENTS:METHODS, COSTS AND SCHEDULES FOR DECONTAMINATION.
NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT.
NUREG/CR-3774 V02: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2A:Technical Requirements For Belowground Vault Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V03: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2B:Technical Requirements For Aboveground Vault Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V04: ALTERNATIVE METHOD FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2C:Technical Requirements For Earth Mounded Concrete Bunker Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V05: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2E:Technical Requirements For Shaft Disposal Of Low Level Radioactive Waste.
NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984.
NUREG/CR-4271: RECOMMENDED SAFETY,RELIABILITY,QUALITY ASSURANCE AND MANAGEMENT AEROSPACE TECHNIQUES WITH POSSIBLE APPLICATION BY THE DOE TO THE HIGH LEVEL RADIOACTIVE WASTE REPOSITORY PROGRAM.

Radioelement Transport
NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT.

Radiographer
NUREG-0940 V04 N02: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June,1985.

Radioisotope
NUREG/CR-4094: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN SULFATE LAKE ENVIRONMENTS.

Radiological Assessment
NUREG/CR-3738: ENVIRONMENTAL EFFECTS OF THE URANIUM FUEL CYCLE.A Review Of Data For Technetium.
NUREG/CR-4057: RADIOLOGICAL ASSESSMENT OF THE TOWN OF EDGEMONT.

Radionuclide
NUREG-0946: AN EVALUATION OF RADIONUCLIDE CONCENTRATIONS IN HIGH-LEVEL RADIOACTIVE WASTES.
NUREG-1164: INFORMATION ON THE CONFINEMENT CAPABILITY OF THE FACILITY DISPOSAL AREA AT WEST VALLEY,NEW YORK.
NUREG-1165: ESRP 7.1.1 "ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RELEASES OF RADIOACTIVE MATERIALS TO GROUNDWATER."
NUREG/CR-2850 V03: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1981.
NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER (Final Report).
NUREG/CR-4101: ASSAY OF LONG-LIVED RADIONUCLIDES IN LOW-LEVEL WASTES FROM POWER REACTORS.
NUREG/CR-4169: AN APPROACH TO TREATING RADIONUCLIDE DECAY HEATING FOR USE IN THE MELCOR CODE SYSTEM.

NUREG/CR-4181: LEACHABILITY OF RADIONUCLIDES FROM CEMENT SOLIDIFIED WASTE FORMS PRODUCED AT OPERATING NUCLEAR POWER REACTORS.
NUREG/CR-4185: AN ASSESSMENT OF DOSIMETRY DATA FOR ACCIDENTAL RADIONUCLIDE RELEASES FROM NUCLEAR REACTORS.
NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS.
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984.
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.
NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.
NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION.
NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions.
NUREG/CR-4251 V02: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 2:Case Study Analysis Of Hydrologic Characterization And Mitigative Schemes.
NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.
NUREG/CR-4397: IN-PLANT SOURCE TERM MEASUREMENTS AT PRAIRIE ISLAND NUCLEAR GENERATING STATION.

Radionuclide Transport
NUREG/CR-3442: RADTWO:A COMPUTER CODE FOR SIMULATING FAST-TRANSIENT, TWO-DIMENSIONAL,TWO-LAYER RADIONUCLIDE CONCENTRATION CONDITIONS IN LAKES,RESERVOIRS,RIVERS,ESTUARIES,AND COASTAL REGIONS.

Radiopharmaceutical Package
NUREG/CR-4035: A HIGHWAY ACCIDENT INVOLVING RADIOPHARMACEUTICALS NEAR BROOKHAVEN,MISSISSIPPI ON DECEMBER 3,1983.

Radium Treatment
NUREG/CR-4259: TAILINGS NEUTRALIZATION AND OTHER ALTERNATIVES FOR IMMOBILIZING TOXIC MATERIALS IN TAILINGS.Final Report.

Radwaste
NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION.
NUREG/CR-4254: OCCUPATIONAL DOSE REDUCTION AND ALARA AT NUCLEAR POWER PLANTS.Study On High-Dose Jobs,Radwaste Handling,And ALARA Incentives.
NUREG/CR-4397: IN-PLANT SOURCE TERM MEASUREMENTS AT PRAIRIE ISLAND NUCLEAR GENERATING STATION.

Reaction Kinetics
NUREG/CP-0062: PROCEEDINGS OF THE CONFERENCE ON THE APPLICATION OF GEOCHEMICAL MODELS TO HIGH-LEVEL NUCLEAR WASTE REPOSITORY ASSESSMENT.

Reactor
NUREG/CR-3943: THE BWR PLAN ANALYZER.
NUREG/CR-3954: HECTR ANALYSIS OF EQUIPMENT TEMPERATURE RESPONSES TO SELECTED HYDROGEN BURNS IN AN ICE CONDENSER CONTAINMENT.

Reactor Coolant Loop
NUREG/CR-3660 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 1:Summary Report.
NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3660 V04: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 4:Pipe Failure Induced By Crack Growth In West Coast Plants.
NUREG/CR-3663 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 1:Summary Report.
NUREG/CR-3663 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 3:Double Ended Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-4290 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF BABCOCK AND WILCOX PWR PLANTS.Volume 2:Guillotine Break Indirectly Induced By Earthquakes.

Reactor Coolant Pressure Boundary
NUREG-1095: EVALUATION OF RESPONSES TO IE BULLETIN 82-02.Degradation Of Threaded Fasteners In Reactor Coolant Pressure Boundary Of Pressurized Water-Reactor Plants.

Reactor Coolant Pump
NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT.

Reactor Coolant Pump Seal
NUREG/CR-4400: THE IMPACT OF MECHANICAL-AND MAINTENANCE-INDUCED FAILURES OF MAIN REACTOR COOLANT PUMP SEALS ON PLANT SAFETY.

Reactor Coolant System
NUREG-1167: TPDWR2:THERMAL POWER DETERMINATION FOR WESTINGHOUSE REACTORS,VERSION2.User's Guide.
NUREG/CR-4205: TRAP-MELT2 USER'S MANUAL.

Reactor Core Thermal Hydraulic
NUREG/CR-4240 V01: PHYSICS OF REACTOR SAFETY.Quarterly Report,January-March 1985.

Reactor Design
NUREG-1070: NRC POLICY ON FUTURE REACTOR DESIGNS.Decisions On Severe Accident Issues In Nuclear Power Plant Regulation.

Reactor Operator
NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.

Reactor Operators Licensing
NUREG/CR-4280: THE EFFECTS OF SUPERVISOR EXPERIENCE AND ASSISTANCE OF A SHIFT TECHNICAL ADVISOR (STA) ON CREW PERFORMANCE IN CONTROL ROOM SIMULATORS.

Reactor Pressure Boundary
NUREG/CR-3825 V03-4: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Quarterly Report, April 1984 - September 1984.Volumes 3 and 4.
NUREG/CR-4300 V01: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Progress Report, October-March 1985.

Reactor Protection System
NUREG/CR-4146: SIMULATION OF AN EPRI-NEVADA TEST SITE (NTS) HYDROGEN BURN TEST AT THE CENTRAL RECEIVER TEST FACILITY.

Reactor Safeguards
NUREG-1125 V01: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 1,Part 1:ACRS Reports On Project Reviews (A-F).
NUREG-1125 V02: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 2,Part 1:ACRS Reports On Project Reviews (G-P).
NUREG-1125 V03: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 3,Part 1:ACRS Reports On Project Reviews (Q-Z).
NUREG-1125 V04: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 4,Part 2:ACRS Reports On Generic Subjects (Accident Analysis - Generic Items).
NUREG-1125 V05: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 5,Part 2:ACRS Reports On Generic Subjects (HTGR - Regulatory Guides).
NUREG-1125 V06: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 6,Part 2:ACRS Reports On Generic Subjects (RPA - Appendix C).

Reactor Safety
NUREG/CP-0058 V01: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CP-0071: TRANSACTIONS OF THE THIRTEENTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984.
NUREG/CR-3361: THE EFFECT OF WATER CHEMISTRY ON THE RATES OF HYDROGEN GENERATION FROM GALVANIZED STEEL CORROSION AT POST-LOCA CONDITIONS.
NUREG/CR-3485: PRA REVIEW MANUAL.
NUREG/CR-3651: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES.
NUREG/CR-3804 V03: PHYSICS OF REACTOR SAFETY.Quarterly Report,July-September 1984.
NUREG/CR-3804 V04: PHYSICS OF REACTOR SAFETY.Quarterly Report,October-December 1984.
NUREG/CR-3810 V03: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,July-September 1984.
NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,October-December 1984.
NUREG/CR-3816 V01: REACTOR SAFETY RESEARCH.Quarterly Report,January-March 1984.
NUREG/CR-3816 V02: REACTOR SAFETY RESEARCH.Quarterly Report,April-June 1984.
NUREG/CR-3816 V03: REACTOR SAFETY RESEARCH.Quarterly Report,July-September 1984.
NUREG/CR-3816 V04: REACTOR SAFETY RESEARCH.Quarterly Report,October-December 1984.
NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1-September 30,1984.
NUREG/CR-3885 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, October 1-December 31,1984.
NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences.
NUREG/CR-4177 V01: MANAGEMENT OF SEVERE ACCIDENTS.Perspectives On Managing Severe Accidents In Commercial Nuclear Power Plants.
NUREG/CR-4177 V02: MANAGEMENT OF SEVERE ACCIDENTS.Extending Plant Operating Procedures Into The Severe Accident Regime.
NUREG/CR-4240 V01: PHYSICS OF REACTOR SAFETY.Quarterly Report,January-March 1985.
NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,January-March 1985.
NUREG/CR-4318 V02: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,April-June 1985.

Reactor Scram
NUREG/CR-3948: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY.

Reactor Trip
NUREG-1154: LOSS OF MAIN AND AUXILIARY FEEDWATER EVENT AT THE DAVIS-BESSE PLANT ON JUNE 9, 1985.

Reactor Vessel
NUREG-1155 V01: RESEARCH PROGRAM PLAN.Reactor Vessels.
NUREG/CR-4365: DESIGN AND DEVELOPMENT OF A SPECIAL PURPOSE SAFT SYSTEM FOR NONDESTRUCTIVE EVALUATION OF NUCLEAR REACTOR VESSELS AND PIPING COMPONENTS.

Real-Time Data Acquisition
NUREG/CR-3872: DATA ACQUISITION AND CONTROL OF THE HSST SERIES V IRRADIATION EXPERIMENT AT THE ORR.

Redox
NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.

Redox Process
NUREG/CR-4114: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS.II.

Redox Reaction
NUREG/CR-3514 V02: THE CHEMICAL BEHAVIOR OF IODINE IN AQUEOUS SOLUTIONS UP TO 150 C.II.Radiation-Redox Conditions.

Reelfoot Rift
NUREG/CR-4333: STE. GENEVIEVE FAULT ZONE,MISSOURI AND ILLINOIS.

Reference Manual
NUREG/CR-4213: SETS REFERENCE MANUAL.

Refill-Reflood
NUREG/CR-3651: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES.

Reflood
NUREG/CR-4166: ANALYSIS OF FLECHT-SEASET 163-ROD BLOCKED BUNDLE DATA USING COBRA-TF.
NUREG/CR-4167: FLECHT SEASET PROGRAM.Final Report.NRC/EPRI Westinghouse Report Number 16.

Reform Amendment
NUREG-1065 R01: ACCEPTANCE CRITERIA FOR THE LOW ENRICHED URANIUM REFORM AMENDMENTS.
NUREG-1065 R01: ACCEPTANCE CRITERIA FOR THE LOW ENRICHED URANIUM REFORM AMENDMENTS.

Regulation
NUREG-1070: NRC POLICY ON FUTURE REACTOR DESIGNS.Decisions On Severe Accident Issues In Nuclear Power Plant Regulation.
NUREG/CR-3817: DEVELOPMENT,USE AND CONTROL OF MAINTENANCE PROCEDURES IN NUCLEAR POWER PLANTS.Problems And Recommendations.

Regulatory Agenda
NUREG-0936 V04 N01: NRC REGULATORY AGENDA.Quarterly Report,January-March 1985.

Regulatory Analysis
NUREG-0869 R01: USI A-43 REGULATORY ANALYSIS.

Regulatory And Technical Report
NUREG-0304 V09 N04: REGULATORY AND TECHNICAL REPORTS.Annual Compilation For 1984.
NUREG-0304 V10 N01: REGULATORY AND TECHNICAL REPORTS.Compilation For First Quarter 1985,January-March.
NUREG-0304 V10 N02: REGULATORY AND TECHNICAL REPORTS.Compilation For Second Quarter 1985,April-June.
NUREG-0304 V10 N03: REGULATORY AND TECHNICAL REPORTS.Compilation For Third Quarter 1985,July - September.

Regulatory Approach
NUREG-0885 I04: U.S. NUCLEAR REGULATORY COMMISSION POLICY AND PLANNING GUIDANCE 1985.

Regulatory Guides
NUREG-1125 V05: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS.1957-1984.Volume 5,Part 2:ACRS Reports On Generic Subjects (HTGR - Regulatory Guides).

Regulatory Issues
NUREG/CR-4103: USES OF HUMAN RELIABILITY ANALYSIS PROBABILISTIC RISK ASSESSMENT RESULTS TO RESOLVE PERSONNEL PERFORMANCE ISSUES THAT COULD AFFECT SAFETY.

Reinforced-Concrete Shear Wall
NUREG/CR-4274: ANALYSIS AND TESTS ON SMALL-SCALE SHEAR WALLS FY-82 FINAL REPORT.

Release
NUREG/CR-4159: COMPARISON OF THE 1981 INEL DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS.
NUREG/CR-4360 V01: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.
NUREG/CR-4360 V02: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.

Reliability
NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.
NUREG-1155 V04: RESEARCH PROGRAM PLAN.Non-Destructive Examination.
NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.
NUREG/CR-3831: THE IN-PLANT RELIABILITY DATA BASE FOR NUCLEAR PLANT COMPONENTS.Interim Report - Diesel Generators,Batteries, Chargers And Inverters.
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.
NUREG/CR-4010: SPECIFICATION OF A HUMAN RELIABILITY DATA BANK FOR CONDUCTING HRA SEGMENTS OF PRAS FOR NUCLEAR POWER PLANTS.

NUREG/CR-4153: APPLICATIONS OF FOREIGN PROBABILISTIC SAFETY ASSESSMENT EXPERIENCE TO THE U.S. NUCLEAR REGULATORY PROCESS.
NUREG/CR-4180: STATE-OF-THE-ART OF SOLID-STATE MOTOR CONTROLLERS.
NUREG/CR-4271: RECOMMENDED SAFETY,RELIABILITY,QUALITY ASSURANCE AND MANAGEMENT AEROSPACE TECHNIQUES WITH POSSIBLE APPLICATION BY THE DOE TO THE HIGH LEVEL RADIOACTIVE WASTE REPOSITORY PROGRAM.
NUREG/CR-4347: EMERGENCY DIESEL GENERATOR OPERATING EXPERIENCE,1981-1983.
NUREG/CR-4350 V03: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 3 - Systems Reliability And Analysis Techniques,Session A -Reliability.
NUREG/CR-4373: COMPENDIUM OF COST-EFFECTIVENESS EVALUATIONS OF MODIFICATIONS FOR DOSE REDUCTION AT NUCLEAR POWER PLANTS.

Reliability Analysis
NUREG/CR-4220: RELIABILITY ANALYSIS OF CONTAINMENT ISOLATION SYSTEMS.
NUREG/CR-4228: REVIEW OF THE VOGTLE UNITS 1 AND 2 AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS.

Reliability Model
NUREG/CR-4350 V06: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 6 - Data Development.

Repair-Welded Stainless Steel
NUREG/CR-3613 V03 N1: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Semiannual Report For October 1984 Through March 1985.

Repeatability
NUREG/CR-4112 V01: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task A:IEEE Flame Test.

Report To Congress
NUREG-0090 V07 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1984.
NUREG-0090 V07 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1984.
NUREG-0090 V08 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.Janurary-March 1985.
NUREG-0090 V08 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1985.

Reporting System
NUREG/CR-4133: NUCLEAR POWER SAFETY REPORTING SYSTEM IMPLEMENTATION AND OPERATIONAL SPECIFICATIONS.

Repository
NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT.
NUREG/CR-4110: REPOSITORY SITE DATA REPORT FOR UNSATURATED TUFF,YUCCA MOUNTAIN,NEVADA.
NUREG/CR-4114: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS.II.
NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.
NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984.
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.

Repository Condition
NUREG/CR-3091 V04: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1983 - March 1984.
NUREG/CR-3091 V05: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period April 1984 - September 1984.

Repository Site
NUREG/CR-2663 V01: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Main Report.
NUREG/CR-2663 V02: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Appendices.

Research
NUREG/CP-0065: TRANSACTIONS OF THE 8TH INTERNATIONAL CONFERENCE ON STRUCTURE MECHANICS IN REACTOR TECHNOLOGY.Panel Session J.K: Status of Research In Structural And Mechanical Engineering For Nuclear Power Plants.
NUREG/CR-3005: SUMMARY OF THE NUCLEAR REGULATORY COMMISSION'S LOFT PROGRAM RESEARCH FINDINGS.

Research Utilization Report
NUREG-1080 V02: LONG-RANGE RESEARCH PLAN FY 1986-FY 1990.

Residual Life
NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.

Residual Stress Measurement
NUREG/CR-4287: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Annual Report,October 1983 - September 1984.

Residue Number System
NUREG/CR-4170: AN ULTRA-HIGH SPEED RESIDUE PROCESSOR FOR SAFT INSPECTION SYSTEM IMAGE ENHANCEMENT.

Residue Processor
NUREG/CR-4170: AN ULTRA-HIGH SPEED RESIDUE PROCESSOR FOR SAFT INSPECTION SYSTEM IMAGE ENHANCEMENT.

Resistance Factor
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.

Respirator
NUREG/CR-3537: EXPEDIENT METHODS OF RESPIRATORY PROTECTION:III. SUBMICRON PARTICLE TESTS AND SUMMARY OF QUALITY FACTORS.
NUREG/CR-3953: THE USE OF MAG-1 SPECTACLES WITH POSITIVE- AND NEGATIVE-PRESSURE RESPIRATORS.
NUREG/CR-4045: LITERATURE REVIEW ON AEROSOL-SAMPLING DEVICES FOR RESPIRATORY FIELD STUDIES.

Respirator Quality Assurance Testing
NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET-GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE,AND SODIUM CHLORIDE.

Response Tree
NUREG/CR-4272: RESPONSE TREE EVALUATION:EXPERIMENTAL ASSESSMENT OF AN EXPERT SYSTEM FOR NUCLEAR REACTOR OPERATORS.

Retention Efficiency
NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

Rg Waves
NUREG/CR-4354: A STUDY OF SEISMICITY AND TECTONICS IN NEW ENGLAND.Final Report.

Rift
NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983.
NUREG/CR-4226: NEW MADRID SEISMOTECTONIC STUDY.Activities During Fiscal Year 1983.

Riprap
NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS.

Risk
NUREG/CR-3816 V04: REACTOR SAFETY RESEARCH.Quarterly Report,October-December 1984.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4050: A REVIEW OF THE SHOREHAM NUCLEAR POWER STATION PROBABILISTIC RISK ASSESSMENT.Internal Events And Core Damage Frequency.
NUREG/CR-4377: EVALUATIONS AND UTILIZATIONS OF RISK IMPORTANCES.

NURFG/CR-4385: EFFECTS OF CONTROL SYSTEM FAILURES IN TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A WESTINGHOUSE PRESSURIZED WATER REACTOR.
NUREG/CR-4386: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A BABCOCK AND WILCOX PRESSURIZED WATER REACTOR.
NUREG/CR-4387: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS AND CORE-MELT FREQUENCIES AT A GENERAL ELECTRIC PRESSURIZED WATER REACTOR.

Risk Analysis
NUREG-1115: CATEGORIZATION OF REACTOR SAFETY ISSUES FROM A RISK PERSPECTIVE.
NUREG/CR-4342: UNCERTAINTY AND SENSITIVITY ANALYSIS OF A MODEL FOR MULTICOMPONENT AEROSOL DYNAMICS.

Risk Assessment
NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.
NUREG/CR-4133: NUCLEAR POWER SAFETY REPORTING SYSTEM IMPLEMENTATION AND OPERATIONAL SPECIFICATIONS.
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP).A Methodology For Estimating Risk Impacts Of Safeguards Measures.

Rock Armor
NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS.

Rock Fracture System
NUREG/CR-4042: A 3-DIMENSIONAL COMPUTER MODEL TO SIMULATE FLUID FLOW AND CONTAINMENT TRANSPORT THROUGH A ROCK FRACTURE SYSTEM.

Rock Mass Sealing
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

Rock Riprap
NUREG/CR-3752: EFFECTS OF HYDROLOGIC VARIABLES ON ROCK RIPRAP DESIGN FOR URANIUM TAILINGS IMPOUNDMENTS.

Rules
NUREG-0936 V04 N01: NRC REGULATORY AGENDA.Quarterly Report,January-March 1985.

Rules Of Practice
NUREG-0386 D03: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.JULY 1972 - SEPTEMBER 1983.

SAFSTOR
NUREG/CR-1755 ADD01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING NUCLEAR REACTORS AT MULTIPLE-REACTOR STATIONS.Effects On Decommissioning Of Interim Inability To Dispose Of Wastes Offsite.

SAFT
NUREG/CR-4365: DESIGN AND DEVELOPMENT OF A SPECIAL PURPOSE SAFT SYSTEM FOR NONDESTRUCTIVE EVALUATION OF NUCLEAR REACTOR VESSELS AND PIPING COMPONENTS.

SAFT-UT
NUREG/CR-4170: AN ULTRA-HIGH SPEED RESIDUE PROCESSOR FOR SAFT INSPECTION SYSTEM IMAGE ENHANCEMENT.

SASA
NUREG/CR-3764: BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE.

SEASET
NUREG/CR-4167: FLECHT SEASET PROGRAM.Final Report.NRC/EPRI Westinghouse Report Number 16.

SEFOR
NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.

SETS
NUREG/CR-4213: SETS REFERENCE MANUAL.

SGTR
NUREG-0844 DRFT FC: NRC INTEGRATED PROGRAM FOR RESOLUTION OF UNRESOLVED SAFETY ISSUES A-3,A-4 AND A-5 REGARDING STEAM GENERATOR TUBE INTEGRITY.Draft Report For Comment.

SIMMER
NUREG/CR-4396: SIMMER POSTPROCESSOR MANUAL.

SIMMER-II
NUREG/CR-4396: SIMMER POSTPROCESSOR MANUAL.

SIMQUAKE
NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS.

SIT-SG
NUREG/CR-4376: HEAT TRANSFER,CARRYOVER AND FALL BACK IN PWR STEAM GENERATORS DURING TRANSIENTS.

SLIM-MAUD
NUREG/CR-4016 V01: APPLICATION OF SLIM-MAUD:A TEST OF AN INTERACTIVE COMPUTER-BASED METHOD FOR ORGANIZING EXPERT ASSESSMENT OF HUMAN PERFORMANCE AND RELIABILITY.Volume I:Main Report.

SMOKE
NUREG/CR-4136: SMOKE.A Data Reduction Package For Analysis Of Combustion Experiments.

SOLA-PTS
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

SPARC
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).

SPD
NUREG/CR-3767: INTERACTIVE SIMULATOR EVALUATION FOR CRT-GENERATED DISPLAYS.

SPDS
NUREG-0800 18.2 R0: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 0 To SRP Section 18.2, "Safety Parameter Display System (SPDS)."
NUREG-0800 18.2A1 R0: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 0 To Appendix A To SRP Section 18.2, "Human Factors Review Guidelines For The Safety Parameter Display System (SPDS)."

SROA
NUREG/CR-4271: RECOMMENDED SAFETY,RELIABILITY,QUALITY ASSURANCE AND MANAGEMENT AEROSPACE TECHNIQUES WITH POSSIBLE APPLICATION BY THE DOE TO THE HIGH LEVEL RADIOACTIVE WASTE REPOSITORY PROGRAM.

SSNM
NUREG/CR-4107: SEQUENTIAL TEST PROCEDURES FOR DETECTING PROTRACTED MATERIALS LOSSES.
NUREG/CR-4108: DEVELOPMENT OF MC&A ALARM RESOLUTION PROCEDURES.

Sabotage
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP).A Methodology For Estimating Risk Impacts Of Safeguards Measures.

Safe Storage
NUREG/CR-1755 ADD01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING NUCLEAR REACTORS AT MULTIPLE-REACTOR STATIONS.Effects On Decommissioning Of Interim Inability To Dispose Of Wastes Offsite.

Safeguards
NUREG-0525 R10: SAFEGUARDS SUMMARY EVENT LIST (SSEL),REVISION 10.
NUREG/CR-3817: DEVELOPMENT,USE AND CONTROL OF MAINTENANCE PROCEDURES IN NUCLEAR POWER PLANTS.Problems And Recommendations.

NUREG/CR-4093: SAFETY/SAFEGUARDS INTERACTIONS DURING SAFETY-RELATED EMERGENCIES AT NUCLEAR POWER REACTOR FACILITIES.
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP).A Methodology For Estimating Risk Impacts Of Safeguards Measures.

Safety
NUREG/CR-4093: SAFETY/SAFEGUARDS INTERACTIONS DURING SAFETY-RELATED EMERGENCIES AT NUCLEAR POWER REACTOR FACILITIES.
NUREG/CR-4271: RECOMMENDED SAFETY,RELIABILITY,QUALITY ASSURANCE AND MANAGEMENT AEROSPACE TECHNIQUES WITH POSSIBLE APPLICATION BY THE DOE TO THE HIGH LEVEL RADIOACTIVE WASTE REPOSITORY PROGRAM.
NUREG/CR-4281: AN EMPIRICAL ANALYSIS OF SELECTED NUCLEAR POWER PLANT MAINTENANCE FACTORS AND PLANT SAFETY.
NUREG/CR-4388: AEROSOL BEHAVIOR MODELING (TASK 3) - SUPPORT SERVICES FOR RESEARCH AND EVALUATION OF SEVERE ACCIDENT PHENOMENA AND MITIGATION.

Safety Assurance
NUREG/CR-4377: EVALUATIONS AND UTILIZATIONS OF RISK IMPORTANCES.

Safety Evaluation Report
NUREG-0420 S09: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SHOREHAM NUCLEAR POWER STATION,UNIT 1.Docket No. 50-322. (Long Island Lighting Company)
NUREG-0675 S28: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company)
NUREG-0675 S28: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company)
NUREG-0675 S29: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company)
NUREG-0675 S30: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company)
NUREG-0675 S31: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company)
NUREG-0787 S10: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATERFORD STEAM ELECTRIC STATION,UNIT 3.Docket No. 50-382. (Louisiana Power And Light Company)
NUREG-0797 S07: 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)
NUREG-0797 S08: 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)
NUREG-0797 S09: 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)
NUREG-0797 S10: 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 Electric Company, et al)
NUREG-0797 S11: 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)
NUREG-0797 S12: 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)
NUREG-0798 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF ENRICO FERMI ATOMIC POWER PLANT,UNIT NO. 2.Docket No. 50-341. (Detroit Edison Company)
NUREG-0798 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF FERMI-2.Docket No. 50-341.(Detroit Edison Company)
NUREG-0847 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT,UNITS 1 AND 2.Docket Nos. 50-390 And 50-391.(Tennessee Valley Authority)
NUREG-0847 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT,UNITS 1 AND 2.Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)
NUREG-0853 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF CLINTON POWER STATION,UNIT 1.Docket No. 50-461.(Illinois Power Company,et al)
NUREG-0857 S08: 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)
NUREG-0857 S09: 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)
NUREG-0876 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BYRON STATION,UNITS 1 AND 2.Docket Nos. 50-454 And 50-455. (Commonwealth Edison Company)
NUREG-0881 S05: 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)
NUREG-0881 S06: 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)
NUREG-0887 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF THE PERRY NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-440 And 50-441.(Cleveland Electric Illuminating Company)
NUREG-0887 S06: 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)
NUREG-0887 S07: 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)
NUREG-0896 S03: SAFETY EVALUATION REPORT 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)
NUREG-0940 V04 N02: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June,1985.
NUREG-0979 S03: SAFETY EVALUATION REPORT RELATED TO THE FINAL DESIGN APPROVAL OF THE GESSAR II BWR/6 NUCLEAR ISLAND DESIGN. Docket No. 50-447. (General Electric Company)
NUREG-0979 S04: SAFETY EVALUATION REPORT RELATED TO FINAL DESIGN APPROVAL OF THE GESSAR II BWR/6 NUCLEAR ISLAND DESIGN.Docket No. 50-447.(General Electric Company)
NUREG-0989 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF RIVER BEND STATION.Docket No. 50-458.(Gulf States Utilities Company,Cajun Electric Power Cooperative)
NUREG-0989 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF RIVER BEND STATION.Docket No. 50-458.(Gulf States Utilities Company,Cajun Electric Power Cooperative)
NUREG-0989 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF RIVER BEND STATION.Docket No. 50-458.(Gulf States Utilities Company,Cajun Electric Power Cooperative)
NUREG-0989 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF RIVER BEND STATION.Docket No. 50-458.(Gulf States Utilities Company,Cajun Electric Power Cooperative)
NUREG-0991 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF LIMERICK GENERATING STATION,UNITS 1 AND 2.Docket Nos. 50-352 And 50-353.(Philadelphia Electric Company)
NUREG-0991 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF LIMERICK GENERATING STATION,UNITS 1 AND 2.Docket Nos. 50-352 And 50-353. (Philadelphia Electric Company)
NUREG-0991 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF LIMERICK GENERATING STATION,UNITS 1 AND 2.Docket Nos. 50-352 And 50-353.(Philadelphia Electric Company)
NUREG-1031 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION,UNIT 3.Docket No. 50-423. (Northeast Nuclear Energy Company)
NUREG-1031 S02: SAFETY EVALUATION RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION,UNIT 3.Docket No.50-423.(Northeast Nuclear Energy Company)
NUREG-1031 S04: SAFETY EVALUATION RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION,UNIT 3.Docket No. 50-423. (Northeast Nuclear Energy Company)
NUREG-1038 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SHEARON HARRIS NUCLEAR POWER PLANT,UNIT 1.Docket No. 50-400. (Carolina Power And Light Company And North Carolina Eastern Municipal Power Agency)
NUREG-1047: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION,UNIT NO. 2.Docket No. 50-410. (Niagara Mohawk Power Corporation,et al)

NUREG-1047 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION,UNIT NO. 2.Docket No. 50-410. (Niagara Mohawk Power Corporation,et al)
NUREG-1047 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION,UNIT NO.2.Docket No. 50-410. (Niagara Mohawk Power Corporation)
NUREG-1048 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF HOPE CREEK GENERATING STATION.Docket No. 50-354.(Public Service Electric And Gas Company,Atlantic City Electric Company)
NUREG-1048 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF HOPE CREEK GENERATING STATION.Docket No. 50-354.(Public Service Electric And Gas Company,Atlantic City Electric Company)
NUREG-1048 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF HOPE CREEK GENERATING STATION.Docket No. 50-354. (Public Service Electric And Gas Company,Atlantic City Electric Company)
NUREG-1048 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF HOPE CREEK GENERATING STATION.Docket No. 50-354.(Public Service Electric And Gas Company,Atlantic City Electric Company)
NUREG-1057: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BEAVER VALLEY POWER STATION,UNIT 2.Docket No. 50-412.(Duquesne Light Company,et al)
NUREG-1096: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE TRIGA TRAINING AND RESEARCH REACTOR AT THE UNIVERSITY OF UTAH.Docket No. 50-407. (University of Utah)
NUREG-1098: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF OPERATING LICENSE FOR THE RESEARCH REACTOR AT MANHATTAN COLLEGE. Docket No. 50-199. (Manhattan College)
NUREG-1119: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE CAVALIER TRAINING REACTOR AT THE UNIVERSITY OF VIRGINIA.Docket No. 50-396.(University Of Virginia)
NUREG-1135: SAFETY EVALUATION REPORT RELATED TO THE CONSTRUCTION PERMIT AND OPERATING LICENSE FOR THE RESEARCH REACTOR AT THE UNIVERSITY OF TEXAS.Docket No. 50-602. (University of Texas)
NUREG-1137: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF VOGTLE ELECTRIC GENERATING PLANT,UNITS 1 AND 2.Docket Nos. 50-424 And 50-425.(Georgia Power Company,et al)
NUREG-1138: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE TRAINING AND RESEARCH REACTOR AT THE UNIVERSITY OF MICHIGAN.Docket No. 50-2.(University of Michigan)
NUREG-1143: SAFETY EVALUATION REPORT RELATED TO THE FULL-TERM OPERATING LICENSE FOR MILLSTONE NUCLEAR POWER STATION,UNIT NO. 1.Docket No. 50-245.(Northeast Nuclear Energy Company)

Safety Fuel Systems Research
NUREG/CR-3980 V02: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,April-June 1984.

Safety Goals
NUREG-1128: TRIAL EVALUATIONS IN COMPARISON WITH THE 1983 SAFETY GOALS.
NUREG/CR-4067: SUMMARY OF BARRIER DEGRADATION EVENTS AND SMALL ACCIDENTS IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.
NUREG/CR-4068: SUMMARY OF HISTORICAL EXPERIENCE WITH RELEASES OF RADIOACTIVE MATERIALS FROM COMMERCIAL NUCLEAR POWER PLANTS IN THE UNITED STATES.
NUREG/CR-4177 V01: MANAGEMENT OF SEVERE ACCIDENTS.Perspectives On Managing Severe Accidents In Commercial Nuclear Power Plants.
NUREG/CR-4177 V02: MANAGEMENT OF SEVERE ACCIDENTS.Extending Plant Operating Procedures Into The Severe Accident Regime.
NUREG/CR-4197: SAFETY GOAL SENSITIVITY STUDIES.

Safety Injection Pump
NUREG-0090 V08 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1985.

Safety Issue A-47
NUREG/CR-4326 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Main Report.

Safety Margin
NUREG/CR-3558: HANDBOOK OF NUCLEAR POWER PLANT SEISMIC FRAGILITIES. Seismic Safety Margins Research Program.

Safety Materials
NUREG/CR-3998 V03: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,October-December 1984.

Safety Parameter Display System
NUREG-0800 18.2 R0: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 0 To SRP Section 18.2, "Safety Parameter Display System (SPDS)."
NUREG-0800 18.2A1 R0: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 0 To Appendix A To SRP Section 18.2, "Human Factors Review Guidelines For The Safety Parameter Display System (SPDS)."
NUREG/CR-3767: INTERACTIVE SIMULATOR EVALUATION FOR CRT-GENERATED DISPLAYS.

Safety Relief Valve
NUREG/CR-3794: CLOSEOUT OF IE BULLETIN 80-25:OPERATING PROBLEMS WITH TARGET ROCK SAFETY-RELIEF VALVES AT BWRS.
NUREG/CR-4046: DETERMINING CRITICAL FLOW VALVE CHARACTERISTICS USING EXTRAPOLATION TECHNIQUES.

Safety Research
NUREG-1105: REVIEW AND EVALUATION OF THE NUCLEAR REGULATORY COMMISSION SAFETY RESEARCH PROGRAM FOR FISCAL YEARS 1986 AND 1987.
NUREG-1125 V01: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 1,Part 1:ACRS Reports On Project Reviews (A-F).
NUREG-1125 V02: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 2,Part 1:ACRS Reports On Project Reviews (G-P).
NUREG-1125 V03: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 3,Part 1.ACRS Reports On Project Reviews (Q-Z).
NUREG-1125 V04: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 4,Part 2:ACRS Reports On Generic Subjects (Accident Analysis - Generic Items).
NUREG-1125 V05: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 5,Part 2:ACRS Reports On Generic Subjects (HTGR - Regulatory Guides).
NUREG-1125 V06: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 6,Part 2:ACRS Reports On Generic Subjects (RPA - Appendix C).
NUREG-1147: SEISMIC SAFETY RESEARCH PROGRAM PLAN.
NUREG/CP-0071: TRANSACTIONS OF THE THIRTEENTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984.
NUREG/CR-2331 V04 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July 1 -September 30,1984.
NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.
NUREG/CR-2531 R03: INTRODUCTORY USER'S MANUAL FOR THE U.S. NUCLEAR REGULATORY COMMISSION REACTOR SAFETY RESEARCH DATA BANK.
NUREG/CR-3816 V03: REACTOR SAFETY RESEARCH.Quarterly Report,July-September 1984.
NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,January-March 1985.
NUREG/CR-4040 V01: REACTOR SAFETY RESEARCH SEMIANNUAL REPORT.January - June 1985.
NUREG/CR-4428: OVERVIEW OF TRAC-BD1/MOD1 ASSESSMENT STUDIES.

Safety Training
NUREG-1127: RADIATION PROTECTION TRAINING AT URANIUM HEXAFLUORIDE AND FUEL FABRICATION PLANTS.

Safety-Related Equipment
NUREG-1148: NUCLEAR POWER PLANT FIRE PROTECTION RESEARCH PROGRAM.

Safety-Related Personnel
NUREG/CR-4248: RECOMMENDATIONS FOR NRC POLICY ON SHIFT SCHEDULING AND OVERTIME AT NUCLEAR POWER PLANTS.

Safety-Related Systems
NUREG/CR-3791: CLOSEOUT OF IE BULLETIN 79-09:FAILURE OF GE TYPE AK-2 CIRCUIT BREAKERS IN SAFETY-RELATED SYSTEMS.
NUREG/CR-4004: CLOSEOUT OF IE BULLETIN 79-25:FAILURES OF WESTINGHOUSE BFD RELAYS IN SAFETY-RELATED SYSTEMS.

Salt
NUREG/CR-2663 V01: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Main Report.
NUREG/CR-2663 V02: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Appendices.

Saturated Fractured Rocks
NUREG/CR-3736: FIELD AND THEORETICAL INVESTIGATIONS OF FRACTURED CRYSTALLINE ROCK NEAR ORACLE,ARIZONA.

Scenario
NUREG/CR-4360 V01: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.
NUREG/CR-4360 V02: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.

Scram
NUREG/CR-4262 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR.Main Report.

Seal
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

Seal Assembly
NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT.

Search System
NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.

Security
NUREG/CR-4093: SAFETY/SAFEGUARDS INTERACTIONS DURING SAFETY-RELATED EMERGENCIES AT NUCLEAR POWER REACTOR FACILITIES.

Security System
NUREG/CR-4298: DESIGN AND INSTALLATION OF COMPUTER SYSTEMS TO MEET THE REQUIREMENTS OF 10 CFR 73.55.

Seismic Design
NUREG-1061 V02 ADD: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2 Addendum:Summary And Evaluation Of Historical Strong-Motion Earthquake Seismic Response And Damage To Aboveground Industrial Piping.
NUREG-1061 V05: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 5:Summary - Piping Review Committee Conclusions and Recommendations.
NUREG/CR-3558: HANDBOOK OF NUCLEAR POWER PLANT SEISMIC FRAGILITIES. Seismic Safety Margins Research Program.

Seismic Design Criteria
NUREG-1061 V02: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2:Evaluation Of Seismic Designs - A Review Of Seismic Design Requirements For Nuclear Power Plant Piping.

Seismic Hazard
NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes.

Seismic History
NUREG/CR-4430: CURRENT METHODOLOGIES FOR ASSESSING THE POTENTIAL FOR EARTHQUARKE-INDUCED LIQUEFACTION IN SOILS.

Seismic Margin
NUREG/CP-0070: PROCEEDINGS OF THE WORKSHOP ON SEISMIC AND DYNAMIC FRAGILITY OF NUCLEAR POWER PLANT COMPONENTS.
NUREG/CR-4334: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS.

Seismic Qualification

Seismic Reflection
NUREG/CR-4339: A REVIEW OF RECENT RESEARCH ON THE SEISMOTECTONICS OF THE SOUTHEASTERN SEABOARD AND AN EVALUATION OF HYPOTHESES ON THE SOURCE OF THE 1886 CHARLESTON,SOUTH CAROLINA EARTHQUAKE.

Seismic Research
NUREG-1147: SEISMIC SAFETY RESEARCH PROGRAM PLAN.
NUREG/CP-0059 V01: PROCEEDINGS OF THE MITI-NRC SEISMIC INFORMATION EXCHANGE MEETING,VOLUME I.
NUREG/CR-4095: TEST SERIES 2:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E EXIDE FHC-19 BATTERY CELLS.
NUREG/CR-4096: TEST SERIES 3:SEISMIC FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E C&D LCU-13 BATTERY CELLS.
NUREG/CR-4097: TEST SERIES 4:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED EXIDE EMP-13 BATTERY CELLS.

Seismic Risk Assessment
NUREG/CR-4123: SEISMIC FRAGILITY OF REINFORCED CONCRETE STRUCTURES AND COMPONENTS FOR APPLICATION TO NUCLEAR FACILITIES.
NUREG/CR-4331: SIMPLIFIED SEISMIC PROBABILISTIC RISK ASSESSMENT.Procedures And Limitations.

Seismic-Fragility Test
NUREG/CR-4096: TEST SERIES 3:SEISMIC FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E C&D LCU-13 BATTERY CELLS.
NUREG/CR-4097: TEST SERIES 4:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED EXIDE EMP-13 BATTERY CELLS.

Seismicity
NUREG/CR-3145 V03: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION - ANNUAL REPORT.(October 1982 - September 1983, Volume 3).
NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983.
NUREG/CR-4288: FOCAL MECHANISM ANALYSES FOR VIRGINIA AND EASTERN TENNESSEE EARTHQUAKES (1978-1984).
NUREG/CR-4317 V01: CANADIAN SEISMIC AGREEMENT.Technical Report Covering 1979-1985.
NUREG/CR-4339: A REVIEW OF RECENT RESEARCH ON THE SEISMOTECTONICS OF THE SOUTHEASTERN SEABOARD AND AN EVALUATION OF HYPOTHESES ON THE SOURCE OF THE 1886 CHARLESTON,SOUTH CAROLINA EARTHQUAKE.
NUREG/CR-4354: A STUDY OF SEISMICITY AND TECTONICS IN NEW ENGLAND.Final Report.

Seismograph
NUREG/CR-4317 V01: CANADIAN SEISMIC AGREEMENT.Technical Report Covering 1979-1985.

Seismotectonics
NUREG/CR-3178: STRUCTURAL AND TECTONIC STUDIES IN NEW YORK STATE.Final Report,July 1981 - June 1982.
NUREG/CR-4226: NEW MADRID SEISMOTECTONIC STUDY.Activities During Fiscal Year 1983.

Self-Contained Breathing Apparatus
NUREG/CR-3953: THE USE OF MAG-1 SPECTACLES WITH POSITIVE- AND NEGATIVE-PRESSURE RESPIRATORS.

Semiscale
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.
NUREG/CR-4278: TRAC-PF1/MOD1 DEVELOPMENT ASSESSMENT.

Semiscale MOD-2A Feedwater-Line Break
NUREG/CR-4189: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT.Semiscale MOD-2A Feedwater-Line Break (S-SF-3) And Steam-Line Break (S-SF-5) Tests.

Semiscale Mod-2B
NUREG/CR-4073: RESULTS OF THE SEMISCALE MOD-2B STEAM GENERATOR TUBE RUPTURE TEST SERIES.

Semiscale Small Break Test
NUREG/CR-3772: RELAP5 ASSESSMENT:SEMISCALE SMALL BREAK TESTS S-UT-1,S-UT-2, S-UT-6,S-UT-7 AND S-UT-8.

Senior Reactor Operator
NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.

Sensitivity Analysis
NUREG/CR-3904: A COMPARISON OF UNCERTAINTY AND SENSITIVITY ANALYSIS TECHNIQUES FOR COMPUTER MODELS.
NUREG/CR-4199: A DEMONSTRATION UNCERTAINTY/SENSITIVITY ANALYSIS USING THE HEALTH AND ECONOMIC CONSEQUENCE MODEL CRAC2.

Sensitivity Study
NUREG/CR-4197: SAFETY GOAL SENSITIVITY STUDIES.

Sensitization
NUREG/CR-4287: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Annual Report,October 1983 - September 1984.

Sequence Coding
NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.

Sequence Coding And Search System
NUREG/CR-4071: EXPLORATORY TREND AND PATTERN ANALYSIS FOR 1981 LICENSEE EVENT REPORT DATA.

Sequential Failures
NUREG/CR-3837: MULTIPLE-SEQUENTIAL FAILURE MODEL.Evaluation Of And Procedures For Human Error Dependency.

Service Water
NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.

Service Water System
NUREG/CR-4233: DISTRIBUTION OF CORBICULA FLUMINEA AT NUCLEAR FACILITIES.

Service Wear
NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.
NUREG/CR-3819: SURVEY OF AGED POWER PLANT FACILITIES.
NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.
NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.

Service-Water
NUREG/CR-4070 V02: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Volume 2:Current Status Of Biofouling Surveillance And Control Techniques.
NUREG/CR-4070 V03: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Factors That May Intensify The Safety Consequences Of Biofouling.

Settlement
NUREG/CR-3972: SETTLEMENT OF URANIUM MILL TAILINGS PILES: A COMPARISON OF ANALYSIS TECHNIQUES.

Severe Accident
NUREG-1070: NRC POLICY ON FUTURE REACTOR DESIGNS.Decisions On Severe Accident Issues In Nuclear Power Plant Regulation.
NUREG-1079 DRFT FC: ESTIMATES OF EARLY CONTAINMENT FROM CORE MELT ACCIDENTS. Draft Report for Comment.
NUREG-1080 V02: LONG-RANGE RESEARCH PLAN FY 1986-FY 1990.
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.
NUREG/CR-3647: DESIGN AND FABRICATION OF A 1/8-SCALE STEEL CONTAINMENT MODEL.
NUREG/CR-3764: BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE.
NUREG/CR-3855: CHARACTERIZATION OF NUCLEAR REACTOR CONTAINMENT PENETRATION - FINAL REPORT.
NUREG/CR-3887: HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS.
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.
NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT.
NUREG/CR-3930: OBSERVED BEHAVIOR OF CESIUM,IODINE,AND TELLURIUM IN THE ORNL FISSION PRODUCT RELEASE PROGRAM.
NUREG/CR-3952: SEQUOYAH EQUIPMENT HATCH SEAL LEAKAGE.
NUREG/CR-4055: THE D10 EXPERIMENT:COOLABILITY OF UO2 DEBRIS IN SODIUM WITH DOWNWARD HEAT REMOVAL.
NUREG/CR-4064: STRUCTURAL RESPONSE OF LARGE PENETRATIONS AND CLOSURES FOR CONTAINMENT VESSELS SUBJECTED TO LOADINGS BEYOND DESIGN BASIS.
NUREG/CR-4080: DETERMINATION OF THE AVAILABILITY OF CORE EXIT THERMOCOUPLES DURING SEVERE ACCIDENT SITUATIONS.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.
NUREG/CR-4119: INTEGRITY OF CONTAINMENT PENETRATIONS UNDER SEVERE ACCIDENT CONDITIONS FY84 ANNUAL REPORT.
NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.
NUREG/CR-4137: PRETEST PREDICTIONS FOR THE RESPONSE OF A 1:8-SCALE STEEL LWR CONTAINMENT BUILDING MODEL TO STATIC OVERPRESSURIZATION.
NUREG/CR-4173: CORSOR USER'S MANUAL.
NUREG/CR-4177 V01: MANAGEMENT OF SEVERE ACCIDENTS.Perspectives On Managing Severe Accidents In Commercial Nuclear Power Plants.
NUREG/CR-4177 V02: MANAGEMENT OF SEVERE ACCIDENTS.Extending Plant Operating Procedures Into The Severe Accident Regime.
NUREG/CR-4209: COMPARISON OF ANALYTICAL PREDICTIONS AND EXPERIMENTAL RESULTS FOR A 1:8-SCALE STEEL CONTAINMENT MODEL PRESSURIZED TO FAILURE.
NUREG/CR-4388: AEROSOL BEHAVIOR MODELING (TASK 3) - SUPPORT SERVICES FOR RESEARCH AND EVALUATION OF SEVERE ACCIDENT PHENOMENA AND MITIGATION.

Severe Fuel Damage
NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,January-March 1985.

Shaft Disposal
NUREG/CR-3774 V05: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2E:Technical Requirements For Shaft Disposal Of Low Level Radioactive Waste.

Shale
NUREG/CR-2663 V01: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Main Report.
NUREG/CR-2663 V02: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Appendices.

Shear
NUREG/CR-4414: DIRECT-CONTACT CONDENSATION OF STEAM ON COLD WATER IN STRATIFIED COUNTERCURRENT FLOW.

Shear Failure
NUREG/CR-4149: ULTIMATE PRESSURE CAPACITY OF REINFORCED AND PRESTRESSED CONCRETE CONTAINMENT.

Shear Wall
NUREG/CR-4274: ANALYSIS AND TESTS ON SMALL-SCALE SHEAR WALLS FY-82 FINAL REPORT.

Shift Scheduling
NUREG/CR-4248: RECOMMENDATIONS FOR NRC POLICY ON SHIFT SCHEDULING AND OVERTIME AT NUCLEAR POWER PLANTS.

Shift Supervisor
NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.

Shift Technical Advisor
NUREG/CR-4280: THE EFFECTS OF SUPERVISOR EXPERIENCE AND ASSISTANCE OF A SHIFT TECHNICAL ADVISOR (STA) ON CREW PERFORMANCE IN CONTROL ROOM SIMULATORS.

Shipment Route
NUREG-0725 R05: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL.

Shipping Container
NUREG/CR-3019: RECOMMENDED WELDED CRITERIA FOR USE IN THE FABRICATION OF SHIPPING CONTAINERS FOR RADIOACTIVE MATERIALS.
NUREG/CR-3854: FABRICATION CRITERIA FOR SHIPPING CONTAINERS.

Short-Period Scram
NUREG-0905: CLOSEOUT OF IE BULLETIN 79-12:SHORT-PERIOD SCRAMS AT BOILING-WATER REACTORS.

Silicone Elastomer
NUREG/CR-4147: THE EFFECT OF ENVIRONMENTAL STRESS ON SYLGARD 70 SILICONE ELASTOMER.

Silver Alumina
NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

Silver Silica Gel
NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

Silver Zeolite
NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

Silver-Indium-Cadmium
NUREG/CR-4401: BEHAVIOR OF CONTROL RODS DURING CORE DEGRADATION: PRESSURIZATION OF SILVER-INDIUM-CADMIUM CONTROL RODS.

Simulation
NUREG/CR-4091: THE EFFECT OF ALTERNATIVE AGING AND ACCIDENT SIMULATIONS ON POLYMER PROPERTIES.

Simulator
NUREG/CR-4206: A SELECT REVIEW OF THE RECENT (1979-1983) BEHAVIORAL RESEARCH LITERATURE ON TRAINING SIMULATORS.

Site Characterization
NUREG/CR-2663 V01: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Main Report.
NUREG/CR-2663 V02: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Appendices.

Smoke Transport
NUREG/CR-4321: FULL SCALE MEASUREMENTS OF SMOKE TRANSPORT AND DEPOSITION IN VENTILATION SYSTEM DUCTWORK.

Snubber
NUREG-1061 V02: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2:Evaluation Of Seismic Designs - A Review Of Seismic Design Requirements For Nuclear Power Plant Piping.
NUREG/CR-4263: RELIABILITY ANALYSIS OF STIFF VERSUS FLEXIBLE PIPING FINAL PROJECT REPORT.

Soil
NUREG/CR-4069: ANALYSES OF SOILS FROM AN AREA ADJACENT TO THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE AT SHEFFIELD,ILLINOIS.
NUREG/CR-4083: ANALYSES OF SOILS FROM THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITES AT BARNWELL,SC AND RICHLAND,WA.

Soil Analysis
NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.

Soil Response
NUREG-1147: SEISMIC SAFETY RESEARCH PROGRAM PLAN.

Soil-Structure Interaction
NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS.
NUREG/CR-4329: RELIABILITY EVALUATION OF CONTAINMENTS INCLUDING SOIL-STRUCTURE INTERACTION.

Solenoid Valve
NUREG/CR-4146: SIMULATION OF AN EPRI-NEVADA TEST SITE (NTS) HYDROGEN BURN TEST AT THE CENTRAL RECEIVER TEST FACILITY.

Solid-State Motor Controller
NUREG/CR-4180: STATE-OF-THE-ART OF SOLID-STATE MOTOR CONTROLLERS.

Solidification
NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.

Soluble Plutonium
NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS.Application To Establishing Values Of f1 For Soluble Plutonium.

Solute Transport
NUREG/CR-4042: A 3-DIMENSIONAL COMPUTER MODEL TO SIMULATE FLUID FLOW AND CONTAINMENT TRANSPORT THROUGH A ROCK FRACTURE SYSTEM.

Sorption
NUREG/CR-4114: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS.II.

Source Term
NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences.

Space-Time Correlation
NUREG/CR-4072: THE ESTIMATION OF ATMOSPHERIC DISPERSION AT NUCLEAR POWER PLANTS UTILIZING REAL TIME ANEMOMETER STATISTICS.

Spacer Grid Heat Transfer
NUREG/CR-4166: ANALYSIS OF FLECHT-SEASET 163-ROD BLOCKED BUNDLE DATA USING COBRA-TF.

Speciation
NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER.(Final Report)

Spectrometry
NUREG/CR-3609: EVALUATION OF NEUTRON DOSIMETRY TECHNIQUES FOR WELL-LOGGING OPERATIONS.

Spent Fuel
NUREG-0725 R05: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL.
NUREG/CR-1755 ADD01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING NUCLEAR REACTORS AT MULTIPLE-REACTOR STATIONS.Effects On Decommissioning Of Interim Inability To Dispose Of Wastes Offsite.

Spent Fuel Rod
NUREG/CR-4074: THE PERFORMANCE OF DEFECTED SPENT LWR FUEL RODS IN INERT GAS AND DRY AIR STORAGE ATMOSPHERES.
NUREG/CR-4084: DRY SPENT FUEL STORAGE TEST PLAN FOR DESTRUCTIVE FUEL ROD EXAMINATIONS.
NUREG/CR-4345: INVESTIGATION OF THE STABILITY OF LWR SPENT FUEL RODS BELOW 250 C.

Spent-Fuel
NUREG/CR-4379 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report,Year Four April-June 1985.

Stability Analysis
NUREG/CR-4116: NUFEGO-NP:A DIGITAL COMPUTER CODE FOR THE LINEAR STABILITY ANALYSIS OF BOILING WATER NUCLEAR REACTORS.

Stability Test
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.

Stainless Steel
NUREG/CR-3911 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Quarterly Report,April-June 1984.
NUREG/CR-3945: FATIGUE CRACK GROWTH RATES OF LOW-CARBON AND STAINLESS PIPING STEELS IN PWR ENVIRONMENT.
NUREG/CR-4060: THE DC-1 AND DC-2 DEBRIS COOLABILITY AND MELT DYNAMICS EXPERIMENTS.

Stainless Steel Piping System
NUREG/CR-4221: AN EVALUATION OF STRESS CORROSION CRACK GROWTH IN BWR PIPING SYSTEMS.

Stainless Steel Weld
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.

Standard Penetration Test
NUREG/CR-4430: CURRENT METHODOLOGIES FOR ASSESSING THE POTENTIAL FOR EARTHQUARKE-INDUCED LIQUEFACTION IN SOILS.

Standard Review Plan
NUREG-0800 06.2.2 R4: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 4 To Section 6.2.2, "Containment Heat Removal System."
NUREG-0800 18.2 R0: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 0 To SRP Section 18.2, "Safety Parameter Display System (SPDS)."
NUREG-0800 18.2A1 R0: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 0 To Appendix A To SRP Section 18.2, "Human Factors Review Guidelines For The Safety Parameter Display System (SPDS)."
NUREG-0800 R05: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 5 To SRP Table Of Contents.

Starting Temperature
NUREG/CR-4112 V01: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task A:IEEE Flame Test.

State Regulation
NUREG/CR-4352: SUGGESTED STATE REQUIREMENTS AND CRITERIA FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE REGULATORY PROGRAM.

Station Blackout
NUREG-1032 DRFT FC: EVALUATION OF STATION BLACKOUT ACCIDENTS AT NUCLEAR POWER PLANTS.Technical Findings Related To Unresolved Safety issue A-44.Draft Report For Comment.
NUREG/CR-3764: BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE.
NUREG/CR-3992: COLLECTION AND EVALUATION OF COMPLETE AND PARTIAL LOSSES OF OFF-SITE POWER AT NUCLEAR POWER PLANTS.
NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT.
NUREG/CR-4347: EMERGENCY DIESEL GENERATOR OPERATING EXPERIENCE,1981-1983.

Statistical Symposium
NUREG/CP-0063: PROCEEDINGS OF THE 1984 STATISTICAL SYMPOSIUM ON NATIONAL ENERGY ISSUES.

Ste Genevieve
NUREG/CR-4333: STE. GENEVIEVE FAULT ZONE,MISSOURI AND ILLINOIS.

Steam Explosion
NUREG-1116: A REVIEW OF THE CURRENT UNDERSTANDING OF THE POTENTIAL FOR CONTAINMENT FAILURE FROM IN-VESSEL STEAM EXPLOSIONS.
NUREG/CR-2718: STEAM EXPLOSION EXPERIMENTS WITH SINGLE DROPS OF IRON OXIDE MELTED WITH A CO2 LASER.Part II:Parametric Studies.

Steam Generation
NUREG/CR-4041: SYSTEM ANALYSIS HANDBOOK.

Steam Generator
NUREG-0844 DRFT FC: NRC INTEGRATED PROGRAM FOR RESOLUTION OF UNRESOLVED SAFETY ISSUES A-3,A-4 AND A-5 REGARDING STEAM GENERATOR TUBE INTEGRITY.Draft Report For Comment.
NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984.
NUREG-1108: RADIOACTIVITY TRANSPORT FOLLOWING STEAM GENERATOR TUBE RUPTURE.
NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.
NUREG/CR-3937: STEAM GENERATOR TUBE RUPTURE IODINE TRANSPORT MECHANISMS. Task 1:Experimental Studies.
NUREG/CR-3949 V01: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Semiannual Progress Report For Period Ending June 30,1984.
NUREG/CR-3949 V02: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Annual Progress Report For Period Ending December 31,1984.
NUREG/CR-4073: RESULTS OF THE SEMISCALE MOD-2B STEAM GENERATOR TUBE RUPTURE TEST SERIES.
NUREG/CR-4276: VIBRATION AND WEAR IN STEAM GENERATOR TUBES FOLLOWING CHEMICAL CLEANING - SEMIANNUAL REPORT.
NUREG/CR-4361: STEAM GENERATOR GROUP PROJECT.Annual Report - 1983.
NUREG/CR-4376: HEAT TRANSFER,CARRYOVER AND FALL BACK IN PWR STEAM GENERATORS DURING TRANSIENTS.

Steam Generator Tube Rupture
NUREG/CR-4079: ANALYTIC STUDIES PERTAINING TO STEAM GENERATOR TUBE RUPTURE ACCIDENTS.

Steam-Line Break
NUREG/CR-4189: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT.Semiscale MOD-2A Feedwater-Line Break (S-SF-3) And Steam-Line Break (S-SF-5) Tests.

Steam-Water Countercurrent Flow
NUREG/CR-4416: STABILITY OF STEAM-WATER COUNTERCURRENT STRATIFIED FLOW.

Steel
NUREG/CR-4283: STUDY OF THE EFFECTS OF ELASTIC UNLOADINGS ON THE JI-R CURVES FROM COMPACT SPECIMENS.

Steel Containment
NUREG/CR-3647: DESIGN AND FABRICATION OF A 1/8-SCALE STEEL CONTAINMENT MODEL.
NUREG/CR-4209: COMPARISON OF ANALYTICAL PREDICTIONS AND EXPERIMENTAL RESULTS FOR A 1:8-SCALE STEEL CONTAINMENT MODEL PRESSURIZED TO FAILURE.

Strategic Information Planning
NUREG/CR-4322 V01: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 1: Enterprise Model.
NUREG/CR-4322 V02: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 2: Data Entity Dictionary.
NUREG/CR-4322 V03: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 3: Data Model.
NUREG/CR-4322 V04: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 4: Preliminary Strategic Data Plan.

Strategic Special Nuclear Material
NUREG/CR-4107: SEQUENTIAL TEST PROCEDURES FOR DETECTING PROTRACTED MATERIALS LOSSES.
NUREG/CR-4108: DEVELOPMENT OF MC&A ALARM RESOLUTION PROCEDURES.

Stratified Debris Bed
NUREG/CR-2951: THE D9 EXPERIMENT.Heat Removal From Stratified UO2 Debris.

Stratified Steam-Water
NUREG/CR-4417: LOCAL PROPERTIES OF COUNTERCURRENT STRATIFIED STEAM-WATER FLOW.

Stratified Two-Phase Flow
NUREG/CR-4416: STABILITY OF STEAM-WATER COUNTERCURRENT STRATIFIED FLOW.

Stress Analysis
NUREG/CR-4367: ORVIRT.PC:A 2-D FINITE-ELEMENT FRACTURE ANALYSIS PROGRAM FOR A MICROCOMPUTER.

Stress Corrosion Cracking
NUREG-1155 V03: RESEARCH PROGRAM PLAN.Piping.
NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984.
NUREG/CR-2331 V04 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July 1 -September 30,1984.
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.
NUREG/CR-3911 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Quarterly Report,April-June 1984.
NUREG/CR-3998 V02: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,April-June 1984.
NUREG/CR-3998 V03: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,October-December 1984.
NUREG/CR-4221: AN EVALUATION OF STRESS CORROSION CRACK GROWTH IN BWR PIPING SYSTEMS.

Stress Field
NUREG/CR-4117: FAULTING AND JOINTING IN AND NEAR SURFACE MINES OF SOUTHWESTERN INDIANA.

Stress-Corrosion Cracking
NUREG/CR-4422: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN PWR ENVIRONMENTS.

Stress-State Variable
NUREG/CR-4087: MEASUREMENTS OF URANIUM MILL TAILINGS CONSOLIDATION CHARACTERISTICS.

Structural Geology
NUREG/CR-3178: STRUCTURAL AND TECTONIC STUDIES IN NEW YORK STATE.Final Report,July 1981 - June 1982.

Structural Reliability
NUREG/CR-4329: RELIABILITY EVALUATION OF CONTAINMENTS INCLUDING SOIL-STRUCTURE INTERACTION.

Structural Steel
NUREG/CR-3228 V03: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1984.

Structure Mechanics
NUREG/CP-0065: TRANSACTIONS OF THE 8TH INTERNATIONAL CONFERENCE ON STRUCTURE MECHANICS IN REACTOR TECHNOLOGY.Panel Session J-K: Status of Research In Structural And Mechanical Engineering For Nuclear Power Plants.

Subcooling
NUREG/CR-4415: COUNTER-CURRENT STEAM/WATER FLOW ABOVE A PERFORATED PLATE-VERTICAL INJECTION OF WATER.

Submicron Particle Tests
NUREG/CR-3537: EXPEDIENT METHODS OF RESPIRATORY PROTECTION:III. SUBMICRON PARTICLE TESTS AND SUMMARY OF QUALITY FACTORS.

Substructure Method
NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS.

Sulfate Lake
NUREG/CR-4094: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN SULFATE LAKE ENVIRONMENTS.

Sulfur
NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS.

Summary Information Report
NUREG-0871 V04 N01: SUMMARY INFORMATION REPORT.Data As Of June 30,1985. (Brown Book)

Sump Design
NUREG-0800 06.2.2 R4: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 4 To Section 6.2.2, "Containment Heat Removal System."

Super System
NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984.

Super System Code
NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.

Superposition Technique
NUREG/CR-3723: STRESS-INTENSITY-FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS.

Suppression Pool Aerosol
NUREG/CR-3317: TECHNICAL BASES AND USER'S MANUAL FOR THE PROTOTYPE OF SPARC - A SUPPRESSION POOL AEROSOL REMOVAL CODE.

Suppression System
NUREG/CR-4230: PROBABILITY-BASED EVALUATION OF SELECTED FIRE PROTECTION FEATURES IN NUCLEAR POWER PLANTS.

Surveillance
NUREG/CR-3319: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium.
NUREG/CR-4070 V02: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Volume 2:Current Status Of Biofouling Surveillance And Control Techniques.
NUREG/CR-4070 V03: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Factors That May Intensify The Safety Consequences Of Biofouling.
NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.

Surveillance Dosimetry Improvement Program
NUREG/CR-3746 V02: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.Semiannual Progress Report,April 1984 - September 1984.
NUREG/CR-3746 V03: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.1984 Annual Report,October 1,1983 - September 30, 1984.

Survey
NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.
NUREG/CR-4139: THE MAILED SURVEY:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL.
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.
NUREG/CR-4268: RATIO METHODS FOR COST-EFFECTIVE FIELD SAMPLING OF COMMERCIAL RADIOACTIVE LOW-LEVEL WASTES.
NUREG/CR-4406: AN ANALYSIS OF LOW-LEVEL WASTES.Review of Hazardous Waste Regulations And Identification of Radioactive Mixed Wastes. Final Report.
NUREG/CR-3516: A SURVEY OF THE USES OF RADIOACTIVE MATERIALS IN LOUISIANA'S OFFSHORE WATERS.

Sylgard 170
NUREG/CR-4147: THE EFFECT OF ENVIRONMENTAL STRESS ON SYLGARD 70 SILICONE ELASTOMER.

Symbolic Manipulation
NUREG/CR-4213: SETS REFERENCE MANUAL.

Synthetic Aperture Focusing Technique
NUREG/CR-4365: DESIGN AND DEVELOPMENT OF A SPECIAL PURPOSE SAFT SYSTEM FOR NONDESTRUCTIVE EVALUATION OF NUCLEAR REACTOR VESSELS AND PIPING COMPONENTS.

Synthetic Liner
NUREG/CR-4023: FIELD PERFORMANCE ASSESSMENT OF SYNTHETIC LINERS FOR URANIUM TAILINGS POND.A Status Report.

System Analysis
NUREG/CR-4041 R01: SYSTEM ANALYSIS HANDBOOK.
NUREG/CR-4350 V04: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 4 - System Reliability And Analysis Techniques.Sessions B/C - Event Trees/Fault Trees.

Systematic Evaluation Program
NUREG-0824 S01: INTEGRATED PLANT SAFETY ASSESSMENT REPORT.SYSTEMATIC EVALUATION PROGRAM.MILLSTONE NUCLEAR POWER STATION.UNIT 1. Docket No. 50-245. (Northeast Nuclear Energy Company)
NUREG-0829 DRFT: INTEGRATED PLANT SAFETY ASSESSMENT REPORT.SYSTEMATIC EVALUATION PROGRAM - SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1.Docket No. 50-206.(Southern California Edison Company)

Systems Analysis
NUREG/CR-3922 V01: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Main Report And Appendices A And B.
NUREG/CR-3922 V02: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Appendices C And D.
NUREG/CR-4350 V01: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 1 - PRA Fundamentals.
NUREG/CR-4350 V03: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 3 - Systems Reliability And Analysis Techniques.Session A -Reliability.
NUREG/CR-4350 V05: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 5 - Systems Reliability And Analysis Techniques.Session D - Quantification.

Systems Approach To Training
NUREG/CR-4439: TRAINING REVIEW CRITERIA AND PROCEDURES.

TAPS
NUREG/CR-3481 V02: NUCLEAR POWER PLANT PERSONNEL QUALIFICATIONS AND TRAINING: TAPS -- The Task Analysis Profiling System.

TLD
NUREG-0837 V04 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, July-September 1984.
NUREG-0837 V04 N04: NRC TLD DIRECT RADIATION MONITORING REPORT.Progress Report. October-December 1984.
NUREG-0837 V05 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, January-March 1985.
NUREG-0837 V05 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, April-June 1985.

TMI-2 Basement
NUREG/CR-4399: POSSIBLE OPTIONS FOR REDUCING OCCUPATIONAL DOSE FROM THE TMI-2 BASEMENT.

TORAC
NUREG/CR-4260: TORAC USER'S MANUAL.A Computer Code For Analyzing Tornado-Induced Flow And Material Transport In Nuclear Facilities.

TPDWR2
NUREG-1167: TPDWR2:THERMAL POWER DETERMINATION FOR WESTINGHOUSE REACTORS.VERSION2.User's Guide.

TRAC
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-3706: TRAC ANALYSES OF SEVERE OVERCOOLING TRANSIENTS FOR THE OCONEE-1 PWR.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4127 V01: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 1.Numerical Methods.
NUREG/CR-4140: DOMINANT ACCIDENT SEQUENCES IN OCONEE-1 PRESSURIZED WATER REACTOR.
NUREG/CR-4253: REVIEW OF TRAC CALCULATIONS FOR CALVERT CLIFFS PTS STUDY.
NUREG/CR-4278: TRAC-PF1/MOD1 DEVELOPMENT ASSESSMENT.

TRAC PF1
NUREG/CR-4044: TRAC-PF1 LOCA CALCULATIONS USING FINE-NODE AND COARSE-NODE INPUT MODELS.

TRAC-BD1 (Version 12)
NUREG/CR-4196: OVERVIEW OF TRAC-BD1 (VERSION 12) ASSESSMENT STUDIES.

TRAC-BD1
NUREG/CR-4428: OVERVIEW OF TRAC-BD1/MOD1 ASSESSMENT STUDIES.

TRAC-BD1/MOD1
NUREG/CR-3633 V01 S1: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.
NUREG/CR-3633 V04: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.Volume 4. Developmental Assessment.
NUREG/CR-4429: TRAC-BD1/MOD1 USERS GUIDELINE.

TRAC-PD2
NUREG/CR-3208: TRAC-PD2 DEVELOPMENTAL ASSESSMENT.
NUREG/CR-4195: OVERVIEW OF TRAC-PD2 ASSESSMENT CALCULATIONS.
NUREG/CR-3866: TRAC-PD2 INDEPENDENT ASSESSMENT.

TRAC-PD2/MOD1
NUREG/CR-4252: INDEPENDENT ASSESSMENT OF TRAC-PD2/MOD1 CODE WITH BCL ECC BYPASS TESTS.

TRAC-PF1
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.
NUREG/CR-3706: TRAC ANALYSES OF SEVERE OVERCOOLING TRANSIENTS FOR THE OCONEE-1 PWR.
NUREG/CR-4109: TRAC-PF1 ANALYSES OF POTENTIAL PRESSURIZED-THERMAL SHOCK TRANSIENTS AT CALVERT CLIFFS/UNIT 1.A Combustion Engineering PWR.
NUREG/CR-4189: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT.Semiscale MOD-2A Feedwater-Line Break (S-SF-3) And Steam-Line Break (S-SF-5) Tests.
NUREG/CR-4253: REVIEW OF TRAC CALCULATIONS FOR CALVERT CLIFFS PTS STUDY.
NUREG/CR-4292: A COMPARATIVE ANALYSIS OF CONSTITUTIVE RELATIONS IN TRAC-PFL AND RELAP5/MOD1.

TRAC-PF1/MOD1
NUREG/CR-3919: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT.NEPTUNUS PRESSURIZER TEST Y05.
NUREG/CR-4155: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT.NORTHWESTERN UNIVERSITY PERFORATED-PLATE CCFL TESTS.
NUREG/CR-4278: TRAC-PF1/MOD1 DEVELOPMENT ASSESSMENT.

TRAN
NUREG/CR-3757: TRAN B-2:THE EFFECT OF LOW STEEL CONTENT ON FUEL PENETRATION IN A NON-MELTING CYLINDRICAL FLOW CHANNEL.

TRAN B Experiment
NUREG/CR-3944: TRAN B-3:EXPERIMENTAL INVESTIGATION OF FUEL CRUST STABILITY ON MELTING SURFACES OF AN ANNULAR FLOW CHANNEL.

TRAN B-3
NUREG/CR-3944: TRAN B-3:EXPERIMENTAL INVESTIGATION OF FUEL CRUST STABILITY ON MELTING SURFACES OF AN ANNULAR FLOW CHANNEL.

TRAP-MELT
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment)
NUREG/CR-4172: A USER'S GUIDE FOR MERGE.

TRAP-MELT 2
NUREG/CR-4205: TRAP-MELT2 USER'S MANUAL.

TRUNC
NUREG/CR-3972: SETTLEMENT OF URANIUM MILL TAILINGS PILES: A COMPARISON OF ANALYSIS TECHNIQUES.

NUREG/CR-4192: THE ANALYSIS OF DRAINAGE AND CONSOLIDATION AT TYPICAL URANIUM MILL TAILINGS SITES.

TRUST
NUREG/CR-3972: SETTLEMENT OF URANIUM MILL TAILINGS PILES: A COMPARISON OF ANALYSIS TECHNIQUES.

TWIST
NUREG/CR-3498: TWO-DIMENSIONAL MODELING OF INTRA-SUBASSEMBLY HEAT TRANSFER AND BUOYANCY-INDUCED FLOW REDISTRIBUTION IN LMFBRS.

Tailings
NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS.
NUREG/CR-3906: URANIUM MILL TAILINGS NEUTRALIZATION:CONTAMINANT COMPLEXATION AND TAILINGS LEACHING STUDY.
NUREG/CR-4075: DESIGNING PROTECTIVE COVERS FOR URANIUM MILL TAILINGS PILES. A Review.
NUREG/CR-4192: THE ANALYSIS OF DRAINAGE AND CONSOLIDATION AT TYPICAL URANIUM MILL TAILINGS SITES.

Tailings Isolation
NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION.

Tailings Neutralization
NUREG/CR-4259: TAILINGS NEUTRALIZATION AND OTHER ALTERNATIVES FOR IMMOBILIZING TOXIC MATERIALS IN TAILINGS.Final Report.

Team Skills
NUREG/CR-4258: AN APPROACH TO TEAM SKILLS TRAINING OF NUCLEAR POWER PLANT CONTROL ROOM CREWS.

Technetium
NUREG/CR-3738: ENVIRONMENTAL EFFECTS OF THE URANIUM FUEL CYCLE.A Review Of Data For Technetium.

Technical Specifications
NUREG-0940 V04 N02: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June,1985.
NUREG-1089: TECHNICAL SPECIFICATIONS FOR FERMI-2.Docket No. 50-341. (Detroit Edison Company)
NUREG-1104: TECHNICAL SPECIFICATIONS FOR WOLF CREEK GENERATING STATION, UNIT 1.Docket No. 50-482.(Kansas Gas And Electric Company)
NUREG-1106: TECHNICAL SPECIFICATIONS FOR CATAWBA NUCLEAR STATION,UNIT 1. Docket No. 50-413.(Duke Power Company)
NUREG-1113: TECHNICAL SPECIFICATIONS FOR BYRON STATION UNITS 1 AND 2. Docket Nos. 50-454 And 50-455. (Commonwealth Edison Company)
NUREG-1117: TECHNICAL SPECIFICATIONS FOR WATERFORD STEAM ELECTRIC STATION UNIT 3.Docket No. 50-382.(Louisiana Power And Light Company)
NUREG-1126: TECHNICAL SPECIFICATIONS FOR SHOREHAM NUCLEAR POWER STATION, UNIT NO. 1.Docket No. 50-322.(Long Island Lighting Company)
NUREG-1126: TECHNICAL SPECIFICATIONS FOR SHOREHAM NUCLEAR POWER STATION, UNIT NO. 1.Docket No. 50-322.(Long Island Lighting Company)
NUREG-1132: TECHNICAL SPECIFICATIONS FOR DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2.Docket No. 50-323.(Pacific Gas and Electric Company)
NUREG-1133: TECHNICAL SPECIFICATIONS FOR PALO VERDE NUCLEAR GENERATING STATION,UNIT 1.Docket No. 50-528. (Arizona Public Service Company)
NUREG-1136: TECHNICAL SPECIFICATIONS FOR WOLF CREEK GENERATING STATION, UNIT 1.Docket No. 50-482.(Kansas Gas And Electric Company)
NUREG-1141: TECHNICAL SPECIFICATIONS FOR FERMI-2 FACILITY.Docket No. 50-341. (Detroit Edison Company)
NUREG-1142: TECHNICAL SPECIFICATIONS FOR RIVER BEND STATION.Docket No. 50-458. (Gulf States Utilities Company)
NUREG-1149: TECHNICAL SPECIFICATIONS FOR LIMERICK GENERATING STATION, UNIT 1.Docket No. 50-352. (Philadelphia Electric Company)
NUREG-1151: TECHNICAL SPECIFICATIONS FOR DIABLO CANYON NUCLEAR POWER PLANT UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company)
NUREG-1161: TECHNICAL SPECIFICATIONS FOR MILLSTONE NUCLEAR POWER STATION UNIT 3.Docket No. 50-423.(Northeast Nuclear Energy Company)
NUREG-1172: TECHNICAL SPECIFICATIONS FOR RIVER BEND STATION. Docket No. 50-458.(Gulf States Utilities Company)
NUREG-1173: TECHNICAL SPECIFICATIONS FOR PALO VERDE NUCLEAR GENERATING STATION,UNIT 2.Docket No.50-529.(Arizona Public Service Company)

Tectonics
NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983.

Teletherapy Calibration
NUREG/CR-4131: INVESTIGATION OF ALTERNATIVE MEANS TO ACCOMPLISH THE GOALS OF BIENNIAL ION CHAMBER CALIBRATION.

Temperature
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-3872: DATA ACQUISITION AND CONTROL OF THE HSST SERIES V IRRADIATION EXPERIMENT AT THE ORR.
NUREG/CR-3945: FATIGUE CRACK GROWTH RATES OF LOW-CARBON AND STAINLESS PIPING STEELS IN PWR ENVIRONMENT.
NUREG/CR-3954: HECTR ANALYSIS OF EQUIPMENT TEMPERATURE RESPONSES TO SELECTED HYDROGEN BURNS IN AN ICE CONDENSER CONTAINMENT.
NUREG/CR-4112 V02: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task B:Firestop Test Method.
NUREG/CR-4387: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS AND CORE-MELT FREQUENCIES AT A GENERAL ELECTRIC PRESSURIZED WATER REACTOR.

Tensile Strength
NUREG/CR-4092: ORNL CHARACTERIZATION OF HEAVY-SECTION STEEL TECHNOLOGY PROGRAM PLATES 01,02,AND 03.

Test
NUREG/CP-0070: PROCEEDINGS OF THE WORKSHOP ON SEISMIC AND DYNAMIC FRAGILITY OF NUCLEAR POWER PLANT COMPONENTS.
NUREG/CR-2482 V06: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program.October 1983 - March 1984.
NUREG/CR-3091 V04: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1983 - March 1984.
NUREG/CR-3091 V05: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period April 1984 - September 1984.
NUREG/CR-3091 V06: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1984 - March 1985.
NUREG/CR-3426 V01: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY. Facility And Test Design Report.
NUREG/CR-3426 V02: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY.Data Report.
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.
NUREG/CR-3736: FIELD AND THEORETICAL INVESTIGATIONS OF FRACTURED CRYSTALLINE ROCK NEAR ORACLE,ARIZONA.
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.
NUREG/CR-3837: MULTIPLE-SEQUENTIAL FAILURE MODEL.Evaluation Of And Procedures For Human Error Dependency.
NUREG/CR-3948: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY.
NUREG/CR-4016 V01: APPLICATION OF SLIM-MAUD:A TEST OF AN INTERACTIVE COMPUTER-BASED METHOD FOR ORGANIZING EXPERT ASSESSMENT OF HUMAN PERFORMANCE AND RELIABILITY.Volume I:Main Report.
NUREG/CR-4073: RESULTS OF THE SEMISCALE MOD-2B STEAM GENERATOR TUBE RUPTURE TEST SERIES.
NUREG/CR-4107: SEQUENTIAL TEST PROCEDURES FOR DETECTING PROTRACTED MATERIALS LOSSES.
NUREG/CR-4136: SMOKE:A Data Reduction Package For Analysis Of Combustion Experiments.
NUREG/CR-4141: CONTAINMENT PURGE AND VENT VALVE TEST PROGRAM FINAL REPORT.
NUREG/CR-4146: SIMULATION OF AN EPRI-NEVADA TEST SITE (NTS) HYDROGEN BURN TEST AT THE CENTRAL RECEIVER TEST FACILITY.
NUREG/CR-4189: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT.Semiscale MOD-2A Feedwater-Line Break (S-SF-3) And Steam-Line Break (S-SF-5) Tests.

NUREG/CR-4274: ANALYSIS AND TESTS ON SMALL-SCALE SHEAR WALLS.FY-82 FINAL REPORT.
NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,January-March 1985.
NUREG/CR-4345: INVESTIGATION OF THE STABILITY OF LWR SPENT FUEL RODS BELOW 250 C.
NUREG/CR-4432: COMPARISON OF DYNAMIC CHARACTERISTICS OF FUKUSHIMA NUCLEAR POWER PLANT CONTAINMENT BUILDING DETERMINED FROM TESTS AND EARTHQUAKES.

Tetroon
NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.

Theft
NUREG-0525 R10: SAFEGUARDS SUMMARY EVENT LIST (SSEL).REVISION 10.

Thermal Aging
NUREG-1155 V03: RESEARCH PROGRAM PLAN.Piping.
NUREG/CR-4147: THE EFFECT OF ENVIRONMENTAL STRESS ON SYLGARD 70 SILICONE ELASTOMER.

Thermal Analysis
NUREG/CR-4064: STRUCTURAL RESPONSE OF LARGE PENETRATIONS AND CLOSURES FOR CONTAINMENT VESSELS SUBJECTED TO LOADINGS BEYOND DESIGN BASIS.

Thermal Annealing
NUREG-1155 V01: RESEARCH PROGRAM PLAN.Reactor Vessels.
NUREG/CR-4212: IN-PLACE THERMAL ANNEALING OF NUCLEAR REACTOR PRESSURE VESSELS.

Thermal Conductivity
NUREG/CR-4060: THE DC-1 AND DC-2 DEBRIS COOLABILITY AND MELT DYNAMICS EXPERIMENTS.

Thermal Cycling
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.

Thermal Fluid Mixing
NUREG/CR-3426 V01: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY. Facility And Test Design Report.
NUREG/CR-3426 V02: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY.Data Report.

Thermal Gradient Tube
NUREG/CR-4105: AN ASSESSMENT OF THERMAL GRADIENT TUBE RESULTS FROM THE HI SERIES OF FISSION PRODUCT RELEASE TESTS.

Thermal Power Determination
NUREG-1167: TPOWR2:THERMAL POWER DETERMINATION FOR WESTINGHOUSE REACTORS,VERSION2.User's Guide.

Thermal Shock
NUREG-1155 V01: RESEARCH PROGRAM PLAN.Reactor Vessels.
NUREG/CR-3744 V02: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR APRIL-SEPTEMBER 1984.
NUREG/CR-4249: PRESSURE VESSEL FRACTURE STUDIES PENETRATING TO THE PWR THERMAL-SHOCK ISSUE.EXPERIMENTS TSE-5,TSE-5A AND TSE-6.
NUREG/CR-4275: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM - FIVE-YEAR PLAN FY 1984-1988.

Thermal Stability
NUREG/CR-4201: THERMAL STABILITY TESTING OF LOW-LEVEL WASTE FORMS.

Thermal-Hydraulic
NUREG-1079 DRFT FC: ESTIMATES OF EARLY CONTAINMENT FROM CORE MELT ACCIDENTS. Draft Report for Comment.
NUREG-1108: RADIOACTIVITY TRANSPORT FOLLOWING STEAM GENERATOR TUBE RUPTURE.
NUREG/CP-0058 V01: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984.
NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.
NUREG/CR-3802: RELAP5 ASSESSMENT:QUANTITATIVE KEY PARAMETERS AND RUN TIME STATISTICS.
NUREG/CR-3804 V03: PHYSICS OF REACTOR SAFETY.Quarterly Report,July-September 1984.
NUREG/CR-3804 V04: PHYSICS OF REACTOR SAFETY.Quarterly Report,October-December 1984.
NUREG/CR-3820 V03: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM.Quarterly Report.July-September 1984.
NUREG/CR-3919: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:NEPTUNUS PRESSURIZER TEST Y05.
NUREG/CR-3935: THERMAL-HYDRAULIC ANALYSES OF OVERCOOLING SEQUENCES FOR THE H.B. ROBINSON UNIT 2 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-3936: RELAP5 ASSESSMENT:CONCLUSIONS AND USER GUIDELINES.
NUREG/CR-4079: ANALYTIC STUDIES PERTAINING TO STEAM GENERATOR TUBE RUPTURE ACCIDENTS.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.
NUREG/CR-4120: MATHEMATICAL MODELING OF ULTIMATE HEAT SINK COOLING PONDS.
NUREG/CR-4155: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:NORTHWESTERN UNIVERSITY PERFORATED-PLATE CCFL TESTS.
NUREG/CR-4172: A USER'S GUIDE FOR MERGE.
NUREG/CR-4189: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT.Semiscale MOD-2A Feedwater-Line Break (S-SF-3) And Steam-Line Break (S-SF-5) Tests.
NUREG/CR-4292: A COMPARATIVE ANALYSIS OF CONSTITUTIVE RELATIONS IN TRAC-PFL AND RELAP5/MOD1.

Thermal-Hydraulic Analyses
NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR.

Thermal-Hydraulic Reactor Safety
NUREG/CR-2331 V04 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July 1 -September 30,1984.

Thermal-Shock
NUREG/CR-4304: PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO THE PWR THERMAL-SHOCK ISSUE.Experiment TSE-7.

Thermocouples
NUREG/CR-4080: DETERMINATION OF THE AVAILABILITY OF CORE EXIT THERMOCOUPLES DURING SEVERE ACCIDENT SITUATIONS.

Thermodynamic Database
NUREG/CP-0062: PROCEEDINGS OF THE CONFERENCE ON THE APPLICATION OF GEOCHEMICAL MODELS TO HIGH-LEVEL NUCLEAR WASTE REPOSITORY ASSESSMENT.

Thermoluminescent Dosimeter
NUREG-0837 V04 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, July-September 1984.
NUREG-0837 V04 N04: NRC TLD DIRECT RADIATION MONITORING REPORT.Progress Report, October-December 1984.
NUREG-0837 V05 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, April-June 1985.
NUREG/CR-4131: INVESTIGATION OF ALTERNATIVE MEANS TO ACCOMPLISH THE GOALS OF BIENNIAL ION CHAMBER CALIBRATION.

Thermomechanical History
NUREG/CR-3613 V03 N1: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Semiannual Report For October 1984 Through March 1985.

Thorium
NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.

Threaded Fastener
NUREG-1095: EVALUATION OF RESPONSES TO IE BULLETIN 82-02.Degradation Of Threaded Fasteners In Reactor Coolant Pressure Boundary Of Pressurized Water-Reactor Plants.

Time Averaging
NUREG/CR-3989: TIME- AND VOLUME-AVERAGED CONSERVATION EQUATIONS FOR MULTIPHASE FLOW.Part One:System Without Internal Solid Structures.

Title List
NUREG-0540 V06 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30,1984.
NUREG-0540 V06 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.December 1-31,1984.
NUREG-0540 V07 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.January 1-31,1985.
NUREG-0540 V07 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.February 1-28,1985.
NUREG-0540 V07 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.March 1-31,1985.
NUREG-0540 V07 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30, 1985.
NUREG-0540 V07 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.May 1-31, 1985.
NUREG-0540 V07 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.June 1-30, 1985.
NUREG-0540 V07 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.July 1-31, 1985.
NUREG-0540 V07 N08: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.August 1-31, 1985.
NUREG-0540 V07 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.October 1-31,1985.

Tornado Model
NUREG/CR-4260: TORAC USER'S MANUAL.A Computer Code For Analyzing Tornado-Induced Flow And Material Transport In Nuclear Facilities.

Tornado Transient
NUREG/CR-4232: THE RESPONSE OF VENTILATION DAMPERS TO LARGE AIRFLOW PULSES.

Toxic Element Migration
NUREG/CR-4435: ORGANIC COMPLEXANT-ENHANCED MOBILITY OF TOXIC ELEMENTS IN LOW-LEVEL WASTES.Annual Report,July 1984 - June 1985.

Training
NUREG-1122: KNOWLEDGES AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS.Pressurized Water Reactors.
NUREG-1134: RADIATION PROTECTION TRAINING FOR PERSONNEL EMPLOYED IN MEDICAL FACILITIES.
NUREG/CR-3481 V02: NUCLEAR POWER PLANT PERSONNEL QUALIFICATIONS AND TRAINING: TAPS -- The Task Analysis Profiling System.
NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.
NUREG/CR-4206: A SELECT REVIEW OF THE RECENT (1979-1983) BEHAVIORAL RESEARCH LITERATURE ON TRAINING SIMULATORS.
NUREG/CR-4258: AN APPROACH TO TEAM SKILLS TRAINING OF NUCLEAR POWER PLANT CONTROL ROOM CREWS.
NUREG/CR-4344: INSTRUCTIONAL SKILLS EVALUATION IN NUCLEAR INDUSTRY TRAINING.
NUREG/CR-4439: TRAINING REVIEW CRITERIA AND PROCEDURES.

Transient
NUREG/CR-3862: DEVELOPMENT OF TRANSIENT INITIATING EVENT FREQUENCIES FOR USE IN PROBABILISTIC RISK ASSESSMENTS.
NUREG/CR-3948: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY.
NUREG/CR-4262 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR.Main Report.
NUREG/CR-4262 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR. Appendices.
NUREG/CR-4342: UNCERTAINTY AND SENSITIVITY ANALYSIS OF A MODEL FOR MULTICOMPONENT AEROSOL DYNAMICS.
NUREG/CR-4386: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A BABCOCK AND WILCOX PRESSURIZED WATER REACTOR.
NUREG/CR-4387: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS AND CORE-MELT FREQUENCIES AT A GENERAL ELECTRIC PRESSURIZED WATER REACTOR.

Transient Analysis
NUREG/CR-3633 V04: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.Volume 4: Developmental Assessment.

Transient Reactor Analysis Code
NUREG/CR-3208: TRAC-PD2 DEVELOPMENTAL ASSESSMENT.
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.
NUREG/CR-3706: TRAC ANALYSES OF SEVERE OVERCOOLING TRANSIENTS FOR THE OCONEE-1 PWR.
NUREG/CR-4127 V01: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 1:Numerical Methods.
NUREG/CR-4140: DOMINANT ACCIDENT SEQUENCES IN OCONEE-1 PRESSURIZED WATER REACTOR.
NUREG/CR-4155: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:NORTHWESTERN UNIVERSITY PERFORATED-PLATE CCFL TESTS.
NUREG/CR-4253: REVIEW OF TRAC CALCULATIONS FOR CALVERT CLIFFS PTS STUDY.
NUREG/CR-4278: TRAC-PF1/MOD1 DEVELOPMENT ASSESSMENT.

Transients
NUREG/CR-4326 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Appendices.
NUREG/CR-4385: EFFECTS OF CONTROL SYSTEM FAILURES IN TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A WESTINGHOUSE PRESSURIZED WATER REACTOR.

Transition-Phase
NUREG/CR-3757: TRAN B-2:THE EFFECT OF LOW STEEL CONTENT ON FUEL PENETRATION IN A NON-MELTING CYLINDRICAL FLOW CHANNEL.

Transmitter
NUREG/CR-4146: SIMULATION OF AN EPRI-NEVADA TEST SITE (NTS) HYDROGEN BURN TEST AT THE CENTRAL RECEIVER TEST FACILITY.

Transport
NUREG/CR-3830 V02: AEROSOL RELEASE AND TRANSPORT PROGRAM.Semiannual Progress Report For April 1984-September 1984.
NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER.(Final Report)
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.
NUREG/CR-4255 V01: AEROSAL RELEASE AND TRANSPORT PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.

Transport Experiments
NUREG/CR-4402 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31,1985.

Transport Mechanism
NUREG/CR-3937: STEAM GENERATOR TUBE RUPTURE IODINE TRANSPORT MECHANISMS. Task 1:Experimental Studies.

Transportation
NUREG-0383 V01 R08: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Packages.
NUREG-0383 V02 R08: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Certificates of Compliance.
NUREG-0383 V03 R05: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Quality Assurance Programs For Radioactive Material Packages.

Transportation Accident
NUREG/CR-4035: A HIGHWAY ACCIDENT INVOLVING RADIOPHARMACEUTICALS NEAR BROOKHAVEN,MISSISSIPPI ON DECEMBER 3,1983.

Transportation Of Radioactive Materials
NUREG/CR-4035: A HIGHWAY ACCIDENT INVOLVING RADIOPHARMACEUTICALS NEAR BROOKHAVEN,MISSISSIPPI ON DECEMBER 3,1983.

Trench
NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984.

Tube Integrity
NUREG-0844 DRFT FC: NRC INTEGRATED PROGRAM FOR RESOLUTION OF UNRESOLVED SAFETY ISSUES A-3,A-4 AND A-5 REGARDING STEAM GENERATOR TUBE INTEGRITY.Draft Report For Comment.
NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.

Tube Rupture
NUREG-1108: RADIOACTIVITY TRANSPORT FOLLOWING STEAM GENERATOR TUBE RUPTURE.
NUREG/CR-3937: STEAM GENERATOR TUBE RUPTURE IODINE TRANSPORT MECHANISMS. Task 1 Experimental Studies.
NUREG/CR-4073: RESULTS OF THE SEMISCALE MOD-2B STEAM GENERATOR TUBE RUPTURE TEST SERIES.

Tubing
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.
NUREG/CR-3949 V02: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Annual Progress Report For Period Ending December 31,1984.
NUREG/CR-4276: VIBRATION AND WEAR IN STEAM GENERATOR TUBES FOLLOWING CHEMICAL CLEANING - SEMIANNUAL REPORT.

Tubing Inspection
NUREG/CR-3949 V01: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Semiannual Progress Report For Period Ending June 30,1984.

Tuff
NUREG/CR-2482 V06: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program.October 1983 - March 1984.
NUREG/CR-2663 V01: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Main Report.
NUREG/CR-2663 V02: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Appendices.
NUREG/CR-3091 V04: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1983 - March 1984.
NUREG/CR-3091 V05: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period April 1984 - September 1984.
NUREG/CR-4110: REPOSITORY SITE DATA REPORT FOR UNSATURATED TUFF,YUCCA MOUNTAIN,NEVADA.

Tumuli
NUREG/CR-3774 V04: ALTERNATIVE METHOD FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2C:Technical Requirements For Earth Mounded Concrete Bunker Disposal Of Low Level Radioactive Waste.

Turbulence
NUREG/CR-4072: THE ESTIMATION OF ATMOSPHERIC DISPERSION AT NUCLEAR POWER PLANTS UTILIZING REAL TIME ANEMOMETER STATISTICS.

Turbulent Mixing
NUREG/CR-3498: TWO-DIMENSIONAL MODELING OF INTRA-SUBASSEMBLY HEAT TRANSFER AND BUOYANCY-INDUCED FLOW REDISTRIBUTION IN LMFBRS.

Turbulent Transfer
NUREG/CR-4157: A SCIENTIFIC CRITIQUE OF AVAILABLE MODELS FOR REAL-TIME SIMULATIONS OF DISPERSION.
NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING.

Twelfth Water Reactor Safety Research
NUREG/CP-0058 V01: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CP-0058 V02: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CP-0058 V03: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CP-0058 V04: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CP-0058 V05: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.
NUREG/CP-0058 V06: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.

Two-Phase Flow
NUREG/CR-4277: INVERTED ANNUAL FLOW EXPERIMENTAL STUDY.

Two-Phase Flow Loop
NUREG/CR-4164: DATA REPORT FOR THE TPFL TEE/CRITICAL FLOW EXPERIMENTS.

Two-Phase Measurement
NUREG/CR-3651: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES.

Type A Package
NUREG/CR-4035: A HIGHWAY ACCIDENT INVOLVING RADIOPHARMACEUTICALS NEAR BROOKHAVEN,MISSISSIPPI ON DECEMBER 3,1983.

UHI
NUREG/CR-3772: RELAP5 ASSESSMENT:SEMISCALE SMALL BREAK TESTS S-UT-1,S-UT-2, S-UT-6,S-UT-7 AND S-UT-8.

USI A-43
NUREG-0800 06.2.2 R4: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS:LWR Edition.Revision 4 T: Section 6.2.2, "Containment Heat Removal System."
NUREG-0869 R01: USI A-43 REGULATORY ANALYSIS.
NUREG-0897 R01: CONTAINMENT EMERGENCY SUMP PERFORMANCE.(Technical Findings Related To Unresolved Safety Issues).

USI A-47
NUREG/CR-4262 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR. Appendices.

Ultimate Heat Sink
NUREG/CR-4120: MATHEMATICAL MODELING OF ULTIMATE HEAT SINK COOLING PONDS.

Ultrasonic Testing
NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984.

Ultrasonics
NUREG-1155 V04: RESEARCH PROGRAM PLAN.Non-Destructive Examination.

Unauthorized Possession
NUREG-1153: INSPECTION REPORT OF UNAUTHORIZED POSSESSION AND USE OF UNSEALED AMERICIUM 241 AND SUBSEQUENT CONFISCATION: J.C. Haynes Company,Newark,Ohio.

Unauthorized Radioactive Material
NUREG/CR-4357: THE FEASIBILITY OF DETECTING THE IMPORT OF UNAUTHORIZED RADIOACTIVE MATERIALS INTO THE UNITED STATES.

Unavailability
NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.

Uncertainties
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

Uncertainty
NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE.
NUREG/CR-4199: A DEMONSTRATION UNCERTAINTY/SENSITIVITY ANALYSIS USING THE HEALTH AND ECONOMIC CONSEQUENCE MODEL CRAC2.
NUREG/CR-4350 V06: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 6 - Data Development.

Uncertainty Analysis
NUREG/CR-3904: A COMPARISON OF UNCERTAINTY AND SENSITIVITY ANALYSIS TECHNIQUES FOR COMPUTER MODELS.

Uncertainty Bound
NUREG/CR-3688 V01: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 1.Main Report.
NUREG/CR-3688 V02: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 2.Appendices.

Uncertainty Estimate
NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING.

Underground Disposal
NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report.October 1981 - September 1984.

Undersodium Experiment
NUREG/CR-4346: AEROSOL RELEASE EXPERIMENTS IN THE FUEL AEROSOL SIMULANT TEST FACILITY:UNDERSODIUM EXPERIMENTS.

Unified Transport Approach
NUREG/CR-3442: RADTWO:A COMPUTER CODE FOR SIMULATING FAST-TRANSIENT, TWO-DIMENSIONAL,TWO-LAYER RADIONUCLIDE CONCENTRATION CONDITIONS IN LAKES,RESERVOIRS,RIVERS,ESTUARIES,AND COASTAL REGIONS.

Unintentional Release
NUREG/CR-4067: SUMMARY OF BARRIER DEGRADATION EVENTS AND SMALL ACCIDENTS IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.

Unresolved Safety Issue
NUREG-0606 V07 N03: UNRESOLVED SAFETY ISSUES SUMMARY.Data As Of August 16,1985. (Aqua Book)

Unresolved Safety Issue A-17
NUREG/CR-3922 V01: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Main Report And Appendices A And B.
NUREG/CR-3922 V02: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Appendices C And D.

Unresolved Safety Issue A-44
NUREG-1032 DRFT FC: EVALUATION OF STATION BLACKOUT ACCIDENTS AT NUCLEAR POWER PLANTS.Technical Findings Related To Unresolved Safety Issue A-44.Draft Report For Comment.
NUREG/CR-3992: COLLECTION AND EVALUATION OF COMPLETE AND PARTIAL LOSSES OF OFF-SITE POWER AT NUCLEAR POWER PLANTS.

Unresolved Safety Issue A-46

Unresolved Safety Issue A-47
NUREG/CR-3991: FAILURE MODES AND EFFECTS ANALYSIS (FMEA) OF THE ICS/NNI ELECTRIC POWER DISTRIBUTION CIRCUITRY AT THE OCONEE-1 NUCLEAR PLANT.

Unresolved Safety Issue A-49
NUREG/CR-3935: THERMAL-HYDRAULIC ANALYSES OF OVERCOOLING SEQUENCES FOR THE H.B. ROBINSON UNIT 2 PRESSURIZED THERMAL SHOCK STUDY.

Unresolved Safety Issues
NUREG-0606 V07 N01: UNRESOLVED SAFETY ISSUES SUMMARY.Data As Of February 15, 1985.(Aqua Book)
NUREG-0606 V07 N02: UNRESOLVED SAFETY ISSUES SUMMARY.Data As Of May 17, 1985. (Aqua Book)
NUREG-0844 DRFT FC: NRC INTEGRATED PROGRAM FOR RESOLUTION OF UNRESOLVED SAFETY ISSUES A-3,A-4 AND A-5 REGARDING STEAM GENERATOR TUBE INTEGRITY.Draft Report For Comment.

Unsaturated Zone
NUREG-1046: DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN UNSATURATED ZONE:TECHNICAL CONSIDERATIONS AND RESPONSE TO COMMENTS.
NUREG/CR-4110: REPOSITORY SITE DATA REPORT FOR UNSATURATED TUFF,YUCCA MOUNTAIN,NEVADA.

Upper Head Injection
NUREG/CR-3772: RELAP5 ASSESSMENT:SEMISCALE SMALL BREAK TESTS S-UT-1,S-UT-2, S-UT-6,S-UT-7 AND S-UT-8.

Upper Head Injection System
NUREG-0090 V07 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1984.

Uranium
NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984.
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.
NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.
NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.

Uranium Fuel Cycle
NUREG/CR-3738: ENVIRONMENTAL EFFECTS OF THE URANIUM FUEL CYCLE:A Review Of Data For Technetium.

Uranium Hexafluoride
NUREG/CR-4360 V01: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.
NUREG/CR-4360 V02: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.

Uranium Mill
NUREG/CR-4075: DESIGNING PROTECTIVE COVERS FOR URANIUM MILL TAILINGS PILES: A Review.

Uranium Mill Tailings
NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS.
NUREG/CR-3906: URANIUM MILL TAILINGS NEUTRALIZATION:CONTAMINANT COMPLEXATION AND TAILINGS LEACHING STUDY.
NUREG/CR-3972: SETTLEMENT OF URANIUM MILL TAILINGS PILES: A COMPARISON OF ANALYSIS TECHNIQUES.
NUREG/CR-4023: FIELD PERFORMANCE ASSESSMENT OF SYNTHETIC LINERS FOR URANIUM TAILINGS POND:A Status Report.
NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION.
NUREG/CR-4087: MEASUREMENTS OF URANIUM MILL TAILINGS CONSOLIDATION CHARACTERISTICS.
NUREG/CR-4089: EVALUATION OF FIELD-TESTED FUGITIVE DUST CONTROL TECHNIQUES FOR URANIUM MILL TAILINGS PILES.
NUREG/CR-4192: THE ANALYSIS OF DRAINAGE AND CONSOLIDATION AT TYPICAL URANIUM MILL TAILINGS SITES.
NUREG/CR-4259: TAILINGS NEUTRALIZATION AND OTHER ALTERNATIVES FOR IMMOBILIZING TOXIC MATERIALS IN TAILINGS.Final Report.

Uranium Milling
NUREG/CR-4176: EMISSION CONTROL TECHNOLOGY AND QUALITY ASSURANCE NEEDS AT URANIUM MILLING FACILITIES.Includes Supporting Methods For Testing,Operating,And Maintaining Air Pollution Control Devices.

Uranium Milling Operations
NUREG/CR-4088: METHODS FOR ESTIMATING RADIOACTIVE AND TOXIC AIRBORNE SOURCE TERMS FOR URANIUM MILLING OPERATIONS.

Uranium Mining
NUREG/CR-3709: METHODS OF MINIMIZING GROUND-WATER CONTAMINATION FROM IN SITU LEACH URANIUM MINING.Final Report.

Uranium Tailings Impoundment
NUREG/CR-3752: EFFECTS OF HYDROLOGIC VARIABLES ON ROCK RIPRAP DESIGN FOR URANIUM TAILINGS IMPOUNDMENTS.

User's Guide
NUREG-1167: TPDWR2:THERMAL POWER DETERMINATION FOR WESTINGHOUSE REACTORS,VERSION2.User's Guide.
NUREG/CR-3901: DOCUMENTATION AND USER'S GUIDE:GS2 & GS3 - VARIABLY SATURATED FLOW AND MASS TRANSPORT MODELS.
NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide.
NUREG/CR-3936: RELAP5 ASSESSMENT:CONCLUSIONS AND USER GUIDELINES.
NUREG/CR-4122: A FORTRAN 77 PROGRAM AND USER'S GUIDE FOR THE CALCULATION OF PARTIAL CORRELATION AND STANDARDIZED REGRESSION COEFFICIENTS.
NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.
NUREG/CR-4172: A USER'S GUIDE FOR MERGE.
NUREG/CR-4429: TRAC-BD1/MOD1 USERS GUIDELINE.

User's Manual
NUREG/CR-2531 R03: INTRODUCTORY USER'S MANUAL FOR THE U.S. NUCLEAR REGULATORY COMMISSION REACTOR SAFETY RESEARCH DATA BANK.
NUREG/CR-3913: HECTR VERSION 1.0 USER'S MANUAL.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.
NUREG/CR-4173: CORSOR USER'S MANUAL.
NUREG/CR-4205: TRAP-MELT2 USER'S MANUAL.
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.

VANESA
NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment).
NUREG/CR-4340 V01: REACTOR SAFETY RESEARCH SEMIANNUAL REPORT.January - June 1985.

VISA
NUREG/CR-4267: VESSEL INTEGRITY SIMULATION (VISA) CODE SENSITIVITY STUDY.

Valence
NUREG/CR-4114: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS II.

Valve
NUREG/CR-3914: PUMP AND VALVE QUALIFICATION REVIEW GUIDE.
NUREG/CR-4228: REVIEW OF THE VOGTLE UNITS 1 AND 2 AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS.

Valve Failure
NUREG/CR-4217: A STATISTICAL ANALYSIS OF NUCLEAR POWER PLANT VALVE FAILURE-RATE VARIABILITY-SOME PRELIMINARY RESULTS.

Valve Operator
NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.

Vapor Pressure
NUREG/CR-4401: BEHAVIOR OF CONTROL RODS DURING CORE DEGRADATION: PRESSURIZATION OF SILVER-INDIUM-CADMIUM CONTROL RODS.

Vapor Pull-Through
NUREG/CR-4164: DATA REPORT FOR THE TPFL TEE/CRITICAL FLOW EXPERIMENTS.

Vault Disposal
NUREG/CR-3774 V02: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2A:Technical Requirements For Belowground Vault Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V03: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2B:Technical Requirements For Aboveground Vault Disposal Of Low Level Radioactive Waste.

Vehicle Barriers
NUREG/CR-4250: VEHICLE BARRIERS:EMPHASIS ON NATURAL FEATURES.

Velocity Structure
NUREG/CR-3145 V03: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION - ANNUAL REPORT.(October 1982 - September 1983, Volume 3).

Vent Valve
NUREG/CR-4141: CONTAINMENT PURGE AND VENT VALVE TEST PROGRAM FINAL REPORT.

Vent Valve Flow
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.

Ventilation
NUREG/CR-4260: TORAC USER'S MANUAL.A Computer Code For Analyzing Tornado-Induced Flow And Material Transport In Nuclear Facilities.

Ventilation Damper
NUREG/CR-4232: THE RESPONSE OF VENTILATION DAMPERS TO LARGE AIRFLOW PULSES.

Ventilation System
NUREG-0090 V07 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1984.
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.
NUREG/CR-4225: SUMMARY OF EFFICIENCY TESTING OF STANDARD AND HIGH-CAPACITY HIGH-EFFICIENCY PARTICULATE AIR FILTERS SUBJECTED TO SIMULATED TORNADO DEPRESSURIZATION AND EXPLOSIVE SHOCK WAVES.
NUREG/CR-4321: FULL SCALE MEASUREMENTS OF SMOKE TRANSPORT AND DEPOSITION IN VENTILATION SYSTEM DUCTWORK.

Vertical Annular Flow
NUREG/CR-4424: DROPLET SIZES,DYNAMICS AND DEPOSITION IN VERTICAL ANNULAR FLOW.

Vessel Integrity Simulation
NUREG/CR-4267: VESSEL INTEGRITY SIMULATION (VISA) CODE SENSITIVITY STUDY.

Vibration
NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.

Visual Display Terminal
NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS.

Visual Display Units
NUREG/CR-4227: HUMAN ENGINEERING GUIDELINES FOR THE EVALUATION AND ASSESSMENT OF VIDEO DISPLAY UNITS.

Void Fraction
NUREG/CR-3651: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES.

Volume Averaging
NUREG/CR-3989: TIME- AND VOLUME-AVERAGED CONSERVATION EQUATIONS FOR MULTIPHASE FLOW.Part One:System Without Internal Solid Structures.

WAPPA
NUREG/CR-3091 V06: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1984 - March 1985.

Waste
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.
NUREG/CR-4200: BIODEGRADATION TESTING OF SOLIDIFIED LOW-LEVEL WASTE STREAMS.
NUREG/CR-4379 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report,Year Four April-June 1985.

Waste Container
NUREG/CR-4062: EXTENDED STORAGE OF LOW-LEVEL RADIOACTIVE WASTES:Potential Problem Areas.
NUREG/CR-4083: ANALYSES OF SOILS FROM THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITES AT BARNWELL,SC AND RICHLAND,WA.

Waste Disposal
NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.
NUREG/CR-3774 V02: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2A:Technical Requirements For Belowground Vault Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V03: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2B:Technical Requirements For Aboveground Vault Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V04: ALTERNATIVE METHOD FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2C:Technical Requirements For Earth Mounded Concrete Bunker Disposal Of Low Level Radioactive Waste.
NUREG/CR-3774 V05: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2E:Technical Requirements For Shaft Disposal Of Low Level Radioactive Waste.
NUREG/CR-4061: LEACHATE PLUME MIGRATION DOWNGRADIENT FROM URANIUM TAILINGS DISPOSAL IN MINE STOPES.
NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984.

Waste Form
NUREG/CR-4181: LEACHABILITY OF RADIONUCLIDES FROM CEMENT SOLIDIFIED WASTE FORMS PRODUCED AT OPERATING NUCLEAR POWER REACTORS.
NUREG/CR-4201: THERMAL STABILITY TESTING OF LOW-LEVEL WASTE FORMS.
NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA.

Waste Gas
NUREG/CR-3237: CONTROL OF EXPLOSIVE MIXTURES IN PWR WASTE GAS SYSTEMS.

Waste Isolation Project
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.

Waste Management
NUREG/CR-4101: ASSAY OF LONG-LIVED RADIONUCLIDES IN LOW-LEVEL WASTES FROM POWER REACTORS.
NUREG/CR-4403: SUMMARY OF THE WASTE MANAGEMENT PROGRAMS AT URANIUM RECOVERY FACILITIES AS THEY RELATE TO THE 40 CFR PART 192 STANDARDS.

Waste Package
NUREG/CR-2482 V06: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program.October 1983 - March 1984.
NUREG/CR-2482 V06: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program.October 1983 - March 1984.
NUREG/CR-2482 V06: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program.October 1983 - March 1984.
NUREG/CR-2482 V07: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program April 1984-September 1984.
NUREG/CR-3091 V04: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1983 - March 1984.
NUREG/CR-3091 V05: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period April 1984 - September 1984.
NUREG/CR-3091 V06: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1984 - March 1985.
NUREG/CR-3900 V02: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report.July-September 1984.
NUREG/CR-3900 V03: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report.October-December 1984.
NUREG/CR-3900 V04: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Annual Report.April 1984 - April 1985.
NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.
NUREG/CR-4379 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report,Year Four April-June 1985.

Waste Repository
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-4303: HIGH-LEVEL WASTE PRECLOSURE SYSTEMS SAFETY ANALYSIS.Phase 1. Final Report.

Water Chemistry
NUREG/CR-3361: THE EFFECT OF WATER CHEMISTRY ON THE RATES OF HYDROGEN GENERATION FROM GALVANIZED STEEL CORROSION AT POST-LOCA CONDITIONS.
NUREG/CR-3998 V02: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report.April-June 1984.
NUREG/CR-3998 V03: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report.October-December 1984.
NUREG/CR-4287: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Annual Report.October 1983 - September 1984.

Water Flow
NUREG/CR-4042: A 3-DIMENSIONAL COMPUTER MODEL TO SIMULATE FLUID FLOW AND CONTAINMENT TRANSPORT THROUGH A ROCK FRACTURE SYSTEM.
NUREG/CR-4415: COUNTER-CURRENT STEAM/WATER FLOW ABOVE A PERFORATED PLATE-VERTICAL INJECTION OF WATER.

Water Hammer
NUREG/CR-4416: STABILITY OF STEAM-WATER COUNTERCURRENT STRATIFIED FLOW.

Wear
NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.
NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.

Weathering
NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS.

Weld Overlay
NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report.October 1983 - September 1984.
NUREG/CR-4287: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Annual Report.October 1983 - September 1984.

Weld Repair
NUREG-1155 V03: RESEARCH PROGRAM PLAN.Piping.

Welded Stainless Steel
NUREG/CR-3613 V03 N1: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Semiannual Report For October 1984 Through March 1985.

Welding
NUREG/CR-3019: RECOMMENDED WELDED CRITERIA FOR USE IN THE FABRICATION OF SHIPPING CONTAINERS FOR RADIOACTIVE MATERIALS.

Weldment
NUREG/CR-3228 V03: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1984.
NUREG/CR-3613 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Annual Report For 1984.
NUREG/CR-4219 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.
NUREG/CR-4275: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM - FIVE-YEAR PLAN FY 1984-1988.

Well-Logging
NUREG/CR-3609: EVALUATION OF NEUTRON DOSIMETRY TECHNIQUES FOR WELL-LOGGING OPERATIONS.

Westinghouse BFD Relays
NUREG/CR-4004: CLOSEOUT OF IE BULLETIN 79-25:FAILURES OF WESTINGHOUSE BFD RELAYS IN SAFETY-RELATED SYSTEMS.

Wet Deposition
NUREG/CR-4157: A SCIENTIFIC CRITIQUE OF AVAILABLE MODELS FOR REAL-TIME SIMULATIONS OF DISPERSION.
NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING.

Wind Tunnel
NUREG/CR-4072: THE ESTIMATION OF ATMOSPHERIC DISPERSION AT NUCLEAR POWER PLANTS UTILIZING REAL TIME ANEMOMETER STATISTICS.

Windscreen
NUREG/CR-4089: EVALUATION OF FIELD-TESTED FUGITIVE DUST CONTROL TECHNIQUES FOR URANIUM MILL TAILINGS PILES.

Wipe Test
NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.

Workbook
NUREG/CR-4125 V02: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES

SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 2:Workbook For Assessment Of Organization And Management.

Workshop

NUREG/CP-0066: PROCEEDINGS OF AN INTERNATIONAL WORKSHOP ON HISTORIC DOSE EXPERIENCE AND DOSE REDUCTION (ALARA) AT NUCLEAR POWER PLANTS.MAY 29-JUNE 1,1984.
NUREG/CP-0070: PROCEEDINGS OF THE WORKSHOP ON SEISMIC AND DYNAMIC FRAGILITY OF NUCLEAR POWER PLANT COMPONENTS.
NUREG/CR-4125 V01: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 1:Guidelines For Utility Organization And Administration Plan.

Xenon

NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

Yellowcake

NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984.
NUREG/CR-4100: EVALUATION OF INSTRUMENTAL METHODS FOR THE MEASUREMENT OF YELLOWCAKE EMISSIONS.

Yucca Mountains

NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.REPORT FOR OCTOBER-DECEMBER 1984.

Zinc-Rich Paint

NUREG/CR-3803: THE EFFECTS OF POST-LOCA CONDITIONS ON A PROTECTIVE COATING (PAINT) FOR THE NUCLEAR POWER INDUSTRY.

Zircaloy

NUREG/CR-3980 V02: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,April-June 1984.
NUREG/CR-3980 V03: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,July-September 1984.
NUREG/CR-3999: ELECTRICALLY HEATED EX-REACTOR PELLET-CLADDING INTERACTION (PCI) SIMULATIONS UTILIZING IRRADIATED ZIRCALOY CLADDING.

Zircaloy Fracture

NUREG/CR-3980 V04: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,October-December 1984.

Zoogeographic Record

NUREG/CR-4233: DISTRIBUTION OF CORBICULA FLUMINEA AT NUCLEAR FACILITIES.

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.

ADVISORY COMMITTEE(S)

ACRS - ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NUREG-1105: REVIEW AND EVALUATION OF THE NUCLEAR REGULATORY COMMISSION SAFETY RESEARCH PROGRAM FOR FISCAL YEARS 1986 AND 1987.

NUREG-1125 V01: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 1,Part 1:ACRS Reports On Project Reviews (A-F).

NUREG-1125 V02: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 2,Part 1:ACRS Reports On Project Reviews (G-P).

NUREG-1125 V03: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 3,Part 1:ACRS Reports On Project Reviews (Q-Z).

NUREG-1125 V04: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 4,Part 2:ACRS Reports On Generic Subjects (Accident Analysis - Generic Items).

NUREG-1125 V05: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 5,Part 2:ACRS Reports On Generic Subjects (HTGR - Regulatory Guides).

NUREG-1125 V06: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 6,Part 2:ACRS Reports On Generic Subjects (RPA - Appendix C).

OFFICE OF EXECUTIVE DIRECTOR FOR OPERATIONS (EDO)

OFFICE OF THE EXECUTIVE DIRECTOR FOR OPERATIONS

NUREG-1154: LOSS OF MAIN AND AUXILIARY FEEDWATER EVENT AT THE DAVIS-BESSE PLANT ON JUNE 9, 1985.

REGION 1, OFFICE OF DIRECTOR

NUREG-0837 V04 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, July-September 1984.

NUREG-0837 V04 N04: NRC TLD DIRECT RADIATION MONITORING REPORT.Progress Report, October-December 1984.

NUREG-0837 V05 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, January-March 1985.

NUREG-0837 V05 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, April-June 1985.

REGION 3, OFFICE OF DIRECTOR

NUREG-1153: INSPECTION REPORT OF UNAUTHORIZED POSSESSION AND USE OF UNSEALED AMERICIUM-241 AND SUBSEQUENT CONFISCATION. J.C. Haynes Company,Newark,Ohio.

EDO - OFFICE OF ADMINISTRATION

DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL

NUREG-0304 V09 N04: REGULATORY AND TECHNICAL REPORTS.Annual Compilation For 1984.

NUREG-0304 V10 N01: REGULATORY AND TECHNICAL REPORTS.Compilation For First Quarter 1985,January-March.

NUREG-0304 V10 N02: REGULATORY AND TECHNICAL REPORTS.Compilation For Second Quarter 1985,April-June.

NUREG-0304 V10 N03: REGULATORY AND TECHNICAL REPORTS.Compilation For Third Quarter 1985,July - September.

NUREG-0540 V06 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30,1984.

NUREG-0540 V06 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.December 1-31,1984.

NUREG-0540 V07 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.January 1-31,1985.

NUREG-0540 V07 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.February 1-28,1985.

NUREG-0540 V07 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.March 1-31,1985.

NUREG-0540 V07 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30, 1985.

NUREG-0540 V07 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.May 1-31, 1985.

NUREG-0540 V07 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.June 1-30, 1985.

NUREG-0540 V07 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.July 1-31, 1985.

NUREG-0540 V07 N08: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.August 1-31, 1985.

NUREG-0540 V07 N09: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.September 1-30,1985.

NUREG-0540 V07 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.October 1-31,1985.

NUREG-0544 R02: A HANDBOOK OF ACRONYMS AND INITIALISMS.

NUREG-0750 V20 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY-SEPTEMBER 1984.

NUREG-0750 V20 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY-DECEMBER 1984.

NUREG-0750 V20 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1984. Pages 1,055-1,435.

NUREG-0750 V20 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1984. Pages 1,437-1,572.

NUREG-0750 V20 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1984. Pages 1,573-1,706.

NUREG-0750 V21 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY-MARCH 1985.

NUREG-0750 V21 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY-JUNE 1985.

NUREG-0750 V21 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1985. Pages 1-273.

NUREG-0750 V21 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1985. Pages 275-469.

NUREG-0750 V21 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1985. Pages 471-559.

NUREG-0750 V21 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1985. Pages 561-1,041.

NUREG-0750 V21 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1985. Pages 1,043-1,567.

NUREG-0750 V21 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1985. Pages 569-1,786.

NUREG-0750 V22 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1985. Pages 1-176.

NUREG-0750 V22 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1985. Pages 177-457.

NUREG-0750 V22 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1985. Pages 459-649.

NUREG-0750 V22 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1985. Pages 651-769.

NUREG-0910 R01 S01: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE.

NUREG-0910 R01 S02: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE.

NUREG-0910 R01 S03: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE.

NUREG-0910 R01 S04: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE.

DIVISION OF RULES AND RECORDS

NUREG-0936 V03 N04: NRC REGULATORY AGENDA.Quarterly Report,October-December 1984.

NUREG-0936 V04 N01: NRC REGULATORY AGENDA.Quarterly Report,January-March 1985.

NUREG-0936 V04 N02: NRC REGULATORY AGENDA.Quarterly Report,April-June 1985.

NUREG-0936 V04 N03: NRC REGULATORY AGENDA.Quarterly Report,July-September 1985.

EDO - OFFICE OF EXECUTIVE LEGAL DIRECTOR

OFFICE OF THE EXECUTIVE LEGAL DIRECTOR

NUREG-0386 D03: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.JULY 1972 - SEPTEMBER 1983.

EDO - OFFICE OF STATE PROGRAMS

OFFICE OF STATE PROGRAMS, DIRECTOR

NUREG-1131: FINANCIAL ANALYSIS OF POTENTIAL RETROSPECTIVE PREMIUM ASSESSMENTS UNDER THE PRICE-ANDERSON SYSTEM.

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA

AEOD, DIRECTOR'S OFFICE

NUREG-0090 V07 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1984.

NUREG-0090 V07 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1984.

NUREG-0090 V08 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.Janurary-March 1985.

NUREG-0090 V08 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1985.

NUREG-1022 S02: LICENSEE EVENT REPORT SYSTEM.Evaluation Of First Year Results And Recommendations For Improvements.

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

DIRECTOR'S OFFICE, OFFICE OF INSPECTION AND ENFORCEMENT

NUREG-0430 V05 N01: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.January 1984 - June 1984.(Gray Book II)

NUREG-0430 V05 N02: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.July 1984 - December 1984.(Gray Book II)

NUREG-0981 R01: NRC/FEMA OPERATIONAL RESPONSE PROCEDURES FOR RESPONSE TO A COMMERCIAL NUCLEAR REACTOR ACCIDENT.

ENFORCEMENT STAFF

NUREG-0940 V03 N04: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,October-December 1984.

NUREG-0940 V04 N01: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,January-March 1985.

NUREG-0940 V04 N02: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June,1985.

NUREG-0940 V04 N03: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,July-September 1985.

DIVISION OF EMERGENCY PREPAREDNESS & ENGINEERING RESPONSE (POST 830103)

NUREG-1095: EVALUATION OF RESPONSES TO IE BULLETIN 82-02.Degradation Of Threaded Fasteners In Reactor Coolant Pressure Boundary Of Pressurized Water-Reactor Plants.

NUREG-1167: TPDWR2:THERMAL POWER DETERMINATION FOR WESTINGHOUSE REACTORS,VERSION2.User's Guide.

EMERGENCY PREPAREDNESS BRANCH

NUREG-0905: CLOSEOUT OF IE BULLETIN 79-12:SHORT-PERIOD SCRAMS AT BOILING-WATER REACTORS.

DIVISION OF QA, SAFEGUARDS & INSP PROGRAMS (830103-850212)

NUREG-0040 V08 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1984. (White Book)

SAFEGUARDS & MATERIALS PROGRAM BRANCH

NUREG-1103: CONTAMINATED MEXICAN STEEL.Importation Of Steel Into The United States That Had Been Inadvertently Contaminated With Cobalt-60 As A Result Of Scrapping Of A Teletherapy Unit.

DIVISION OF QA, VENDOR & TECHNICAL TRAINING CENTER PROGRAMS (POST 85021

NUREG-0040 V09 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1985.(White Book)

NUREG-0040 V09 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,April-June 1985. (White Book)

NUREG-0040 V09 N03: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,July 1985-September 1985.(White Book)

OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS

DIVISION OF FUEL CYCLE & MATERIAL SAFETY

NUREG-0383 V01 R08: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Packages.

NUREG-0383 V02 R08: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Certificates of Compliance.

NUREG-0383 V03 R05: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Quality Assurance Programs For Radioactive Material Packages.

NUREG-1112: ENVIRONMENTAL ASSESSMENT FOR RENEWAL OF SPECIAL NUCLEAR MATERIAL LICENSE NO. SNM-368.(UNC Naval Products Division Of UNC Resources,Inc)

NUREG-1118: ENVIRONMENTAL ASSESSMENT FOR RENEWAL OF SPECIAL NUCLEAR MATERIAL LICENSE NO. SNM-1107.Docket No. 70-1151. (Westinghouse Electric Corporation)

NUREG-1130: ENVIRONMENTAL ASSESSMENT FOR RENEWAL AND CONSOLIDATION OF MATERIALS LICENSE NOS. SNM-362,SMB-405,08-00566-05, 08-00566-10,AND 08-00566-12.

NUREG-1157: ENVIRONMENTAL ASSESSMENT FOR RENEWAL OF SOURCE MATERIAL LICENSE NO. SUB-1010.Docket No. 40-8027. (Sequoyah Fuels Corporation)

DIVISION OF SAFEGUARDS

NUREG-0725 R05: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL.

NUREG-1065 R01: ACCEPTANCE CRITERIA FOR THE LOW ENRICHED URANIUM REFORM AMENDMENTS.

LICENSING POLICY & PROGRAMS BRANCH (PRE 850707)

NUREG-0525 R10: SAFEGUARDS SUMMARY EVENT LIST (SSEL).REVISION 10.

DIVISION OF WASTE MANAGEMENT

NUREG-0946: AN EVALUATION OF RADIONUCLIDE CONCENTRATIONS IN HIGH-LEVEL RADIOACTIVE WASTES.

U.S. NUCLEAR REGULATORY COMMISSION

COMMISSIONERS

NUREG-0750 V22 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-September 1985.

NUREG-0885 I04: U.S. NUCLEAR REGULATORY COMMISSION POLICY AND PLANNING GUIDANCE 1985.

NRC - NO DETAILED AFFILIATION GIVEN

NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences.

NUREG/CR-4173: CORSOR USER'S MANUAL.

OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 4/05/81)

OFFICE OF NUCLEAR REGULATORY RESEARCH, DIRECTOR

NUREG-1032 DRFT FC: EVALUATION OF STATION BLACKOUT ACCIDENTS AT NUCLEAR POWER PLANTS.Technical Findings Related To Unresolved Safety Issue A-44.Draft Report For Comment.

NUREG-1080 V02: LONG-RANGE RESEARCH PLAN FY 1986-FY 1990.

NUREG-1164: INFORMATION ON THE CONFINEMENT CAPABILITY OF THE FACILITY DISPOSAL AREA AT WEST VALLEY,NEW YORK.

NUREG/CP-0058 V01: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.

NUREG/CP-0058 V02: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.

NUREG/CP-0058 V03: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.

NUREG/CP-0058 V04: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.

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NUREG/CP-0058 V06: PROCEEDINGS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.

NUREG/CP-0071: TRANSACTIONS OF THE THIRTEENTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING.

ACCIDENT SOURCE TERM PROGRAM OFFICE

NUREG-0956 DRFT FC: REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS. (Draft Report For Comment)

DIVISION OF ACCIDENT EVALUATION

NUREG-1108: RADIOACTIVITY TRANSPORT FOLLOWING STEAM GENERATOR TUBE RUPTURE.

DIVISION OF RISK ANALYSIS & OPERATIONS (POST 840429)

NUREG-1115: CATEGORIZATION OF REACTOR SAFETY ISSUES FROM A RISK PERSPECTIVE.

NUREG-1140 DRFT FC: A REGULATORY ANALYSIS ON EMERGENCY PREPAREDNESS FOR FUEL CYCLE AND OTHER RADIOACTIVE MATERIAL LICENSEES.Draft Report For Comment.

DIVISION OF RADIATION PROGRAMS & EARTH SCIENCES (POST 840429)

NUREG-0713 V05: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS - 1983 ANNUAL REPORT.

NUREG-0714 V04-05: OCCUPATIONAL RADIATION EXPOSURE.Fifteenth And Sixteenth Annual Reports,1982 And 1983.

NUREG-1046: DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN UNSATURATED Z0NE:TECHNICAL CONSIDERATIONS AND RESPONSE TO COMMENTS.

NUREG-1127: RADIATION PROTECTION TRAINING AT URANIUM HEXAFLUORIDE AND FUEL FABRICATION PLANTS.

NUREG-1134: RADIATION PROTECTION TRAINING FOR PERSONNEL EMPLOYED IN MEDICAL FACILITIES.

DIVISION OF ENGINEERING TECHNOLOGY
NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984.
NUREG-1144: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.
NUREG-1147: SEISMIC SAFETY RESEARCH PROGRAM PLAN.
NUREG-1148: NUCLEAR POWER PLANT FIRE PROTECTION RESEARCH PROGRAM.
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NUREG-1155 V02: RESEARCH PROGRAM PLAN.Steam Generators.
NUREG-1155 V03: RESEARCH PROGRAM PLAN.Piping.
NUREG-1155 V04: RESEARCH PROGRAM PLAN.Non-Destructive Examination.
NUREG/CP-0065: TRANSACTIONS OF THE 8TH INTERNATIONAL CONFERENCE ON STRUCTURE MECHANICS IN REACTOR TECHNOLOGY.Panel Session J-K: Status of Research In Structural And Mechanical Engineering For Nuclear Power Plants.

INTRA-AGENCY COMMITTEES, REVIEW GROUPS, ETC.
COMANCHE PEAK PROJECT (TECHNICAL REVIEW TEAM)
NUREG-0737 S12: 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)
PIPING REVIEW COMMITTEE
NUREG-1061 V02: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2.Evaluation Of Seismic Designs - A Review Of Seismic Design Requirements For Nuclear Power Plant Piping.
NUREG-1061 V02 ADD: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2 Addendum.Summary And Evaluation Of Historical Strong-Motion Earthquake Seismic Response And Damage To Aboveground Industrial Piping.
NUREG-1061 V05: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 5.Summary - Piping Review Committee Conclusions and Recommendations.
STEAM EXPLOSION REVIEW GROUP
NUREG-1116: A REVIEW OF THE CURRENT UNDERSTANDING OF THE POTENTIAL FOR CONTAINMENT FAILURE FROM IN-VESSEL STEAM EXPLOSIONS.

EDO-RESOURCE MANAGEMENT
OFFICE OF RESOURCE MANAGEMENT, DIRECTOR
NUREG-0325 R07: U.S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS.
NUREG-1145 V01: U.S. NUCLEAR REGULATORY COMMISSION 1984 ANNUAL REPORT.
DIVISION OF BUDGET & ANALYSIS
NUREG-0020 V08 N12: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of November 30,1984.(Gray Book I)
NUREG-0020 V09 N01: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of December 31,1984.(Gray Book I)
NUREG-0020 V09 N02: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of January 31,1985.(Gray Book I)
NUREG-0020 V09 N03: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of Febuary 28,1985.(Gray Book I)
NUREG-0020 V09 N04: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of March 31,1985.(Gray Book I)
NUREG-0020 V09 N05: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of April 30, 1985.(Gray Book I)
NUREG-0020 V09 N06: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of May 31,1985.(Gray Book I)
NUREG-0020 V09 N07: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of June 30,1985.(Gray Book I)
NUREG-0020 V09 N08: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of July 31,1985.(Gray Book I)
NUREG-0020 V09 N09: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of August 31,1985.(Gray Book I)
NUREG-0020 V09 N10: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of September 30,1985.(Gray Book I)
NUREG-0020 V09 N11: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of October 31,1985.(Gray Book I)
NUREG-0871 V04 N01: SUMMARY INFORMATION REPORT.Data As Of June 30,1985. (Brown Book)
NUREG-1100 V01: FY 1986 BUDGET ESTIMATES.
MANAGEMENT SUPPORT BRANCH
NUREG-0748 V04 N12: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of December 31,1984.(Orange Book)
NUREG-0748 V05 N01: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of January 31,1985.(Orange Book)
NUREG-0748 V05 N02: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of February 28,1985.(Orange Book)
NUREG-0748 V05 N03: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of March 31,1985. (Orange Book)
NUREG-0748 V05 N04: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of April 30,1985.(Orange Book)
NUREG-0748 V05 N05: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of May 31,1985. (Orange Book)
NUREG-0748 V05 N06: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of June 30,1985. (Orange Book)
NUREG-0748 V05 N07: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of July 31,1985.(Orange Book)
NUREG-0748 V05 N08: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of August 31,1985. (Orange Book)
NUREG-0748 V05 N09: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of September 30,1985.(Orange Book)
NUREG-0748 V05 N10: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of October 31,1985. (Orange Book)

OFFICE OF NUCLEAR REACTOR REGULATION (POST 4/28/80)
OFFICE OF NUCLEAR REACTOR REGULATION, DIRECTOR (POST 851125)
NUREG-0420 S09: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SHOREHAM NUCLEAR POWER STATION,UNIT 1.Docket No. 50-322. (Long Island Lighting Company)
NUREG-0824 S01: INTEGRATED PLANT SAFETY ASSESSMENT REPORT.SYSTEMATIC EVALUATION PROGRAM-MILLSTONE NUCLEAR POWER STATION,UNIT 1. Docket No. 50-245. (Northeast Nuclear Energy Company)
NUREG-0887 S07: 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)
NUREG-0989 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF RIVER BEND STATION.Docket No. 50-458.(Gulf States Utilities Company,Cajun Electric Power Cooperative)
NUREG-1031 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION,UNIT NO. 3.Docket No. 50-423.(Northeast Nuclear Energy Company)
NUREG-1031 S04: SAFETY EVALUATION RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION,UNIT 3.Docket No. 50-423. (Northeast Nuclear Energy Company)
NUREG-1047 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION,UNIT NO.2.Docket No. 50-410. (Niagara Mohawk Power Corporation)
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EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA

AEOD, DIRECTOR'S OFFICE

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NUREG/CR-3551: SAFETY IMPLICATIONS ASSOCIATED WITH IN-PLANT PRESSURIZED GAS STORAGE AND DISTRIBUTION SYSTEMS IN NUCLEAR POWER PLANTS.

NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide.

NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.

NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.

NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.

NUREG/CR-4071: EXPLORATORY TREND AND PATTERN ANALYSIS FOR 1981 LICENSEE EVENT REPORT DATA.

NUREG/CR-4152: AN INDEPENDENT SAFETY ORGANIZATION.

NUREG/CR-4178 DRFT: AN EVALUATION OF SELECTED LICENSEE EVENT REPORTS PREPARED PURSUANT TO 10 CFR 50.73.Draft Report.

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OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS,DIRECTOR

NUREG/CR-4406: AN ANALYSIS OF LOW-LEVEL WASTES:Review of Hazardous Waste Regulations And Identification of Radioactive Mixed Wastes. Final Report.

DIVISION OF FUEL CYCLE & MATERIAL SAFETY

NUREG/CR-3019: RECOMMENDED WELDED CRITERIA FOR USE IN THE FABRICATION OF SHIPPING CONTAINERS FOR RADIOACTIVE MATERIALS.

NUREG/CR-3865: EVALUATION OF THE RADIOACTIVE INVENTORY IN,AND ESTIMATION OF ISOTOPIC RELEASE FROM,THE WASTE IN EIGHT TRENCHES AT THE SHEFFIELD LOW-LEVEL WASTE BURIAL SITE.

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NUREG/CR-2482 V07: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program April 1984-September 1984.

NUREG/CR-2482 V08: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period October 1984 - March 1985.

NUREG/CR-2482 V09: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period April 1985-September 1985.

NUREG/CR-3091 V04: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1983 - March 1984.

NUREG/CR-3091 V05: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period April 1984 - September 1984.

NUREG/CR-3091 V06: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1984 - March 1985.

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NUREG/CR-3774 V03: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2B:Technical Requirements For Aboveground Vault Disposal Of Low Level Radioactive Waste.

NUREG/CR-3774 V04: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2C:Technical Requirements For Earth Mounded Concrete Bunker Disposal Of Low Level Radioactive Waste.

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NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984.

NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.

NUREG/CR-3901: DOCUMENTATION AND USER'S GUIDE:GS2 & GS3 - VARIABLY SATURATED FLOW AND MASS TRANSPORT MODELS.

NUREG/CR-4023: FIELD PERFORMANCE ASSESSMENT OF SYNTHETIC LINERS FOR URANIUM TAILINGS POND.A Status Report.

NUREG/CR-4057: RADIOLOGICAL ASSESSMENT OF THE TOWN OF EDGEMONT.

NUREG/CR-4062: EXTENDED STORAGE OF LOW-LEVEL RADIOACTIVE WASTES.Potential Problem Areas.

NUREG/CR-4069: ANALYSES OF SOILS FROM AN AREA ADJACENT TO THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE AT SHEFFIELD,ILLINOIS.

NUREG/CR-4083: ANALYSES OF SOILS FROM THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITES AT BARNWELL,SC AND RICHLAND,WA.

NUREG/CR-4101: ASSAY OF LONG-LIVED RADIONUCLIDES IN LOW-LEVEL WASTES FROM POWER REACTORS.

NUREG/CR-4110: REPOSITORY SITE DATA REPORT FOR UNSATURATED TUFF,YUCCA MOUNTAIN,NEVADA.

NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.

NUREG/CR-4150: EPICOR-II RESIN DEGRADATION RESULTS FROM FIRST RESIN SAMPLES OF PF-8 AND PF-20.

NUREG/CR-4200: BIODEGRADATION TESTING OF SOLIDIFIED LOW-LEVEL WASTE STREAMS.

NUREG/CR-4201: THERMAL STABILITY TESTING OF LOW-LEVEL WASTE FORMS.

NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA.

NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984.

NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.

NUREG/CR-4271: RECOMMENDED SAFETY,RELIABILITY,QUALITY ASSURANCE AND MANAGEMENT AEROSPACE TECHNIQUES WITH POSSIBLE APPLICATION BY THE DOE TO THE HIGH-LEVEL RADIOACTIVE WASTE REPOSITORY PROGRAM.

NUREG/CR-4403: SUMMARY OF THE WASTE MANAGEMENT PROGRAMS AT URANIUM RECOVERY FACILITIES AS THEY RELATE TO THE 40 CFR PART 192 STANDARDS.

NUREG/CR-4406: AN ANALYSIS OF LOW-LEVEL WASTES.Review of Hazardous Waste Regulations And Identification of Radioactive Mixed Wastes. Final Report.

U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF POLICY EVALUATIONS

NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.

NRC - NO DETAILED AFFILIATION GIVEN

NUREG/CR-4050: A REVIEW OF THE SHOREHAM NUCLEAR POWER STATION PROBABILISTIC RISK ASSESSMENT.Internal Events And Core Damage Frequency.

OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 4/05/81)

OFFICE OF NUCLEAR REGULATORY RESEARCH, DIRECTOR

NUREG/CR-3519: HUMAN ERROR PROBABILITY ESTIMATION USING LICENSEE EVENT REPORTS.

NUREG/CR-4046: DETERMINING CRITICAL FLOW VALVE CHARACTERISTICS USING EXTRAPOLATION TECHNIQUES.

NUREG/CR-4146: SIMULATION OF AN EPRI-NEVADA TEST SITE (NTS) HYDROGEN BURN TEST AT THE CENTRAL RECEIVER TEST FACILITY.

DIVISION OF ACCIDENT EVALUATION

NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984.

NUREG/CR-2331 V04 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July 1 -September 30,1984.

NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.

NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.

NUREG/CR-2531 R03: INTRODUCTORY USER'S MANUAL FOR THE U.S. NUCLEAR REGULATORY COMMISSION REACTOR SAFETY RESEARCH DATA BANK.

NUREG/CR-2718: STEAM EXPLOSION EXPERIMENTS WITH SINGLE DROPS OF IRON OXIDE MELTED WITH A CO2 LASER.Part II:Parametric Studies.

NUREG/CR-2951: THE D9 EXPERIMENT.Heat Removal From Stratified UO2 Debris.

NUREG/CR-3005: SUMMARY OF THE NUCLEAR REGULATORY COMMISSION'S LOFT PROGRAM RESEARCH FINDINGS.

NUREG/CR-3197 V01: REACTION BETWEEN SOME CESIUM-IODINE COMPOUNDS AND THE REACTOR MATERIALS 304 STAINLESS STEEL,INCONEL 600 & SILVER. Volume I:Cesium Hydroxide Reactions.

NUREG/CR-3208: TRAC-PD2 DEVELOPMENTAL ASSESSMENT.

NUREG/CR-3361: THE EFFECT OF WATER CHEMISTRY ON THE RATES OF HYDROGEN GENERATION FROM GALVANIZED STEEL CORROSION AT POST-LOCA CONDITIONS.

NUREG/CR-3426 V01: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY. Facility And Test Design Report.

NUREG/CR-3426 V02: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY.Data Report.

NUREG/CR-3498: TWO-DIMENSIONAL MODELING OF INTRA-SUBASSEMBLY HEAT TRANSFER AND BUOYANCY-INDUCED FLOW REDISTRIBUTION IN LMFBRS.

NUREG/CR-3514 V02: THE CHEMICAL BEHAVIOR OF IODINE IN AQUEOUS SOLUTIONS UP TO 150 C.II.Radiation-Redox Conditions.

NUREG/CR-3633 V01 S1: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.

NUREG/CR-3633 V04: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.Volume 4: Developmental Assessment.

NUREG/CR-3638: HYDROGEN-STEAM JET-FLAME FACILITY AND EXPERIMENTS.

NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.

NUREG/CR-3651: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES.

NUREG/CR-3700: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.

NUREG/CR-3706: TRAC ANALYSES OF SEVERE OVERCOOLING TRANSIENTS FOR THE OCONEE-1 PWR.

NUREG/CR-3721 V01: PRESSURE MEASUREMENTS IN A HYDROGEN COMBUSTION ENVIRONMENT. Hydrogen-Air Combustion Test Series 1 And 2 In The FITS Tank.

NUREG/CR-3741 V02: EVALUATION OF POWER REACTOR FUEL ROD ANALYSIS CAPABILITIES. Phase 2 Topical Report.Volume 2:Code Evaluation.

NUREG/CR-3757: TRAN B-2:THE EFFECT OF LOW STEEL CONTENT ON FUEL PENETRATION IN A NON-MELTING CYLINDRICAL FLOW CHANNEL.

NUREG/CR-3764: BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE.

NUREG/CR-3772: RELAP5 ASSESSMENT:SEMISCALE SMALL BREAK TESTS S-UT-1,S-UT-2, S-UT-6,S-UT-7 AND S-UT-8.

NUREG/CR-3802: RELAP5 ASSESSMENT:QUANTITATIVE KEY PARAMETERS AND RUN TIME STATISTICS.

NUREG/CR-3803: THE EFFECTS OF POST-LOCA CONDITIONS ON A PROTECTIVE COATING (PAINT) FOR THE NUCLEAR POWER INDUSTRY.

NUREG/CR-3804 V03: PHYSICS OF REACTOR SAFETY.Quarterly Report,July-September 1984.

NUREG/CR-3804 V04: PHYSICS OF REACTOR SAFETY.Quarterly Report,October-December 1984.

NUREG/CR-3810 V03: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,July-September 1984.
NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,October-December 1984.
NUREG/CR-3816 V01: REACTOR SAFETY RESEARCH.Quarterly Report,January-March 1984.
NUREG/CR-3816 V02: REACTOR SAFETY RESEARCH.Quarterly Report,April-June 1984.
NUREG/CR-3816 V03: REACTOR SAFETY RESEARCH.Quarterly Report,July-September 1984.
NUREG/CR-3816 V04: REACTOR SAFETY RESEARCH.Quarterly Report,October-December 1984.
NUREG/CR-3820 V03: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM.Quarterly Report,July-September 1984.
NUREG/CR-3830 V02: AEROSOL RELEASE AND TRANSPORT PROGRAM.Semiannual Progress Report For April 1984-September 1984.
NUREG/CR-3866: TRAC-PD2 INDEPENDENT ASSESSMENT.
NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1-September 30,1984.
NUREG/CR-3885 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, October 1-December 31,1984.
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.
NUREG/CR-3912: MARCH-HECTR ANALYSIS OF SELECTED ACCIDENTS IN AN ICE-CONDENSER CONTAINMENT.
NUREG/CR-3913: HECTR VERSION 1.0 USER'S MANUAL.
NUREG/CR-3919: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT.NEPTUNUS PRESSURIZER TEST Y05.
NUREG/CR-3930: OBSERVED BEHAVIOR OF CESIUM,IODINE,AND TELLURIUM IN THE ORNL FISSION PRODUCT RELEASE PROGRAM.
NUREG/CR-3935: THERMAL-HYDRAULIC ANALYSES OF OVERCOOLING SEQUENCES FOR THE H.B. ROBINSON UNIT 2 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-3936: RELAP5 ASSESSMENT.CONCLUSIONS AND USER GUIDELINES.
NUREG/CR-3937: STEAM GENERATOR TUBE RUPTURE IODINE TRANSPORT MECHANISMS. Task 1:Experimental Studies.
NUREG/CR-3943: THE BWR PLAN ANALYZER.
NUREG/CR-3944: TRAN B-3.EXPERIMENTAL INVESTIGATION OF FUEL CRUST STABILITY ON MELTING SURFACES OF AN ANNULAR FLOW CHANNEL.
NUREG/CR-3948: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY.
NUREG/CR-3954: HECTR ANALYSIS OF EQUIPMENT TEMPERATURE RESPONSES TO SELECTED HYDROGEN BURNS IN AN ICE CONDENSER CONTAINMENT.
NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR.
NUREG/CR-3980 V02: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,April-June 1984.
NUREG/CR-3980 V03: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,July-September 1984.
NUREG/CR-3980 V04: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,October-December 1984.
NUREG/CR-3989: TIME- AND VOLUME-AVERAGED CONSERVATION EQUATIONS FOR MULTIPHASE FLOW.Part One:System Without Internal Solid Structures.
NUREG/CR-3990: CHARCOAL PERFORMANCE UNDER ACCIDENT CONDITIONS IN LIGHT-WATER REACTORS.
NUREG/CR-3999: ELECTRICALLY HEATED EX-REACTOR PELLET-CLADDING INTERACTION (PCI) SIMULATIONS UTILIZING IRRADIATED ZIRCALOY CLADDING.
NUREG/CR-4020: HMS:A COMPUTER PROGRAM FOR TRANSIENT,THREE-DIMENSIONAL MIXING GASES.
NUREG/CR-4037: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-5.
NUREG/CR-4041: SYSTEM ANALYSIS HANDBOOK.
NUREG/CR-4041 R01: SYSTEM ANALYSIS HANDBOOK.
NUREG/CR-4043: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-6.
NUREG/CR-4044: TRAC-PF1 LOCA CALCULATIONS USING FINE-NODE AND COARSE-NODE INPUT MODELS.
NUREG/CR-4055: THE D10 EXPERIMENT.COOLABILITY OF UO2 DEBRIS IN SODIUM WITH DOWNWARD HEAT REMOVAL.
NUREG/CR-4060: THE DC-1 AND DC-2 DEBRIS COOLABILITY AND MELT DYNAMICS EXPERIMENTS.
NUREG/CR-4073: RESULTS OF THE SEMISCALE MOD-2B STEAM GENERATOR TUBE RUPTURE TEST SERIES.
NUREG/CR-4079: ANALYTIC STUDIES PERTAINING TO STEAM GENERATOR TUBE RUPTURE ACCIDENTS.
NUREG/CR-4081: ABSORPTION OF GASEOUS IODINE BY WATER DROPLETS.
NUREG/CR-4085: USERS MANUAL FOR CONTAIN 1.0.A Computer Code for Severe Reactor Accident Containment Analysis.
NUREG/CR-4105: AN ASSESSMENT OF THERMAL GRADIENT TUBE RESULTS FROM THE HI SERIES OF FISSION PRODUCT RELEASE TESTS.
NUREG/CR-4109: TRAC-PF1 ANALYSES OF POTENTIAL PRESSURIZED-THERMAL SHOCK TRANSIENTS AT CALVERT CLIFFS/UNIT 1.A Combustion Engineering PWR.
NUREG/CR-4115: INTERNATIONAL STANDARD PROBLEM 13 (LOFT EXPERIMENT L2-5). Final Comparison Report.
NUREG/CR-4116: NUFEGQ-NP:A DIGITAL COMPUTER CODE FOR THE LINEAR STABILITY ANALYSIS OF BOILING WATER NUCLEAR REACTORS.
NUREG/CR-4127 V01: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 1:Numerical Methods.
NUREG/CR-4127 V02: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 2:Models.
NUREG/CR-4127 V03: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 3:Development Assessment For Plant Application.
NUREG/CR-4128: BWR FULL INTEGRAL SIMULATION TEST (FIST) PHASE II TEST RESULTS AND TRAC-BWR MODEL QUALIFICATION.
NUREG/CR-4138: DATA ANALYSES FOR NEVADA TEST SITE (NTS) PREMIXED COMBUSTION TESTS.
NUREG/CR-4140: DOMINANT ACCIDENT SEQUENCES IN OCONEE-1 PRESSURIZED WATER REACTOR.
NUREG/CR-4155: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT.NORTHWESTERN UNIVERSITY PERFORATED-PLATE CCFL TESTS.
NUREG/CR-4164: DATA REPORT FOR THE TPFL TEE/CRITICAL FLOW EXPERIMENTS.
NUREG/CR-4167: FLECHT SEASET PROGRAM.Final Report.NRC/EPRI Westinghouse Report Number 16.
NUREG/CR-4168: GT2F:A COMPUTER CODE FOR ESTIMATING LIGHT WATER REACTOR FUEL ROD FAILURES.
NUREG/CR-4172: A USER'S GUIDE FOR MERGE.
NUREG/CR-4173: CORSOR USER'S MANUAL.
NUREG/CR-4189: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT.Semiscale MOD-2A Feedwater-Line Break (S-SF-3) And Steam-Line Break (S-SF-5) Tests.
NUREG/CR-4195: OVERVIEW OF TRAC-PD2 ASSESSMENT CALCULATIONS.
NUREG/CR-4198: OVERVIEW OF TRAC-BD1 (VERSION 12) ASSESSMENT STUDIES.
NUREG/CR-4205: TRAP-MELT2 USER'S MANUAL.
NUREG/CR-4218: LOCA SIMULATION IN THE NATIONAL RESEARCH UNIVERSAL REACTOR PROGRAM.Postirradiation Examination Results For The Third Materials Test (MT-3) - Second Campaign.
NUREG/CR-4240 V01: PHYSICS OF REACTOR SAFETY.Quarterly Report,January-March 1985.
NUREG/CR-4252: INDEPENDENT ASSESSMENT OF TRAC-PD2/MOD1 CODE WITH BCL ECC BYPASS TESTS.
NUREG/CR-4253: REVIEW OF TRAC CALCULATIONS FOR CALVERT CLIFFS PTS STUDY.
NUREG/CR-4255 V01: AEROSAL RELEASE AND TRANSPORT PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.
NUREG/CR-4277: INVERTED ANNUAL FLOW EXPERIMENTAL STUDY.
NUREG/CR-4278: TRAC-PF1/MOD1 DEVELOPMENT ASSESSMENT.
NUREG/CR-4292: A COMPARATIVE ANALYSIS OF CONSTITUTIVE RELATIONS IN TRAC-PFL AND RELAP5/MOD1.
NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,January-March 1985.
NUREG/CR-4318 V02: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,April-June 1985.
NUREG/CR-4340 V01: REACTOR SAFETY RESEARCH SEMIANNUAL REPORT.January - June 1985.
NUREG/CR-4346: AEROSOL RELEASE EXPERIMENTS IN THE FUEL AEROSOL SIMULANT TEST FACILITY:UNDERSODIUM EXPERIMENTS.
NUREG/CR-4351: SUMMARY REPORT FOR LOFT ANTICIPATED TRANSIENT EXPERIMENT SERIES L6-8.

NUREG/CR-4376: HEAT TRANSFER,CARRYOVER AND FALL BACK IN PWR STEAM GENERATORS DURING TRANSIENTS.
NUREG/CR-4383: HIGH PRESSURE INJECTION OF MELT FROM A REACTOR PRESSURE VESSEL - THE DISCHARGE PHASE.
NUREG/CR-4388: AEROSOL BEHAVIOR MODELING (TASK 3) - SUPPORT SERVICES FOR RESEARCH AND EVALUATION OF SEVERE ACCIDENT PHENOMENA AND MITIGATION.
NUREG/CR-4396: SIMMER POSTPROCESSOR MANUAL.
NUREG/CR-4401: BEHAVIOR OF CONTROL RODS DURING CORE DEGRADATION: PRESSURIZATION OF SILVER-INDIUM-CADMIUM CONTROL RODS.
NUREG/CR-4402 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31,1985.
NUREG/CR-4414: DIRECT-CONTACT CONDENSATION OF STEAM ON COLD WATER IN STRATIFIED COUNTERCURRENT FLOW.
NUREG/CR-4415: COUNTER-CURRENT STEAM/WATER FLOW ABOVE A PERFORATED PLATE-VERTICAL INJECTION OF WATER.
NUREG/CR-4416: STABILITY OF STEAM-WATER COUNTERCURRENT STRATIFIED FLOW.
NUREG/CR-4417: LOCAL PROPERTIES OF COUNTERCURRENT STRATIFIED STEAM-WATER FLOW.
NUREG/CR-4424: DROPLET SIZES,DYNAMICS AND DEPOSITION IN VERTICAL ANNULAR FLOW.
NUREG/CR-4428: OVERVIEW OF TRAC-BD1/MOD1 ASSESSMENT STUDIES.
NUREG/CR-4429: TRAC-BD1/MOD1 USERS GUIDELINE.

DIVISION OF RISK ANALYSIS & OPERATIONS (POST 840429)

NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT.
NUREG/CR-3301: CATALOG OF PRA DOMINANT ACCIDENT SEQUENCE INFORMATION.
NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.
NUREG/CR-3469 V02: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS. Annotated Bibliography Of Selected Readings In Radiation Protection And ALARA.
NUREG/CR-3481 V02: NUCLEAR POWER PLANT PERSONNEL QUALIFICATIONS AND TRAINING: TAPS -- The Task Analysis Profiling System.
NUREG/CR-3537: EXPEDIENT METHODS OF RESPIRATORY PROTECTION:III. SUBMICRON PARTICLE TESTS AND SUMMARY OF QUALITY FACTORS.
NUREG/CR-3611: RADIOACTIVE MATERIAL (RAM) ACCIDENT/INCIDENT DATA ANALYSIS PROGRAM.
NUREG/CR-3626 V02: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: DESCRIPTION OF MODEL CONTENT,STRUCTURE,AND SENSITIVITY TESTING.
NUREG/CR-3634: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: USER'S MANUAL.
NUREG/CR-3688 V01: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 1:Main Report.
NUREG/CR-3688 V02: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 2:Appendices.
NUREG/CR-3767: INTERACTIVE SIMULATOR EVALUATION FOR CRT-GENERATED DISPLAYS.
NUREG/CR-3831: THE IN-PLANT RELIABILITY DATA BASE FOR NUCLEAR PLANT COMPONENTS.Interim Report - Diesel Generators,Batteries, Chargers And Inverters.
NUREG/CR-3837: MULTIPLE-SEQUENTIAL FAILURE MODEL.Evaluation Of And Procedures For Human Error Dependency.
NUREG/CR-3862: DEVELOPMENT OF TRANSIENT INITIATING EVENT FREQUENCIES FOR USE IN PROBABILISTIC RISK ASSESSMENTS.
NUREG/CR-3887: HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS.
NUREG/CR-3904: A COMPARISON OF UNCERTAINTY AND SENSITIVITY ANALYSIS TECHNIQUES FOR COMPUTER MODELS.
NUREG/CR-4009: HUMAN RELIABILITY DATA BANK.Evaluation Results.
NUREG/CR-4010: SPECIFICATION OF A HUMAN RELIABILITY DATA BANK FOR CONDUCTING HRA SEGMENTS OF PRAS FOR NUCLEAR POWER PLANTS.
NUREG/CR-4016 V01: APPLICATION OF SLIM-MAUD:A TEST OF AN INTERACTIVE COMPUTER-BASED METHOD FOR ORGANIZING EXPERT ASSESSMENT OF HUMAN PERFORMANCE AND RELIABILITY.Volume I:Main Report.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.
NUREG/CR-4093: SAFETY/SAFEGUARDS INTERACTIONS DURING SAFETY-RELATED EMERGENCIES AT NUCLEAR POWER REACTOR FACILITIES.
NUREG/CR-4103: USES OF HUMAN RELIABILITY ANALYSIS PROBABILISTIC RISK ASSESSMENT RESULTS TO RESOLVE PERSONNEL PERFORMANCE ISSUES THAT COULD AFFECT SAFETY.
NUREG/CR-4104: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL. Field Evaluation/Validation.
NUREG/CR-4122: A FORTRAN 77 PROGRAM AND USER'S GUIDE FOR THE CALCULATION OF PARTIAL CORRELATION AND STANDARDIZED REGRESSION COEFFICIENTS.
NUREG/CR-4133: NUCLEAR POWER SAFETY REPORTING SYSTEM IMPLEMENTATION AND OPERATIONAL SPECIFICATIONS.
NUREG/CR-4153: APPLICATIONS OF FOREIGN PROBABILISTIC SAFETY ASSESSMENT EXPERIENCE TO THE U.S. NUCLEAR REGULATORY PROCESS.
NUREG/CR-4169: AN APPROACH TO TREATING RADIONUCLIDE DECAY HEATING FOR USE IN THE MELCOR CODE SYSTEM.
NUREG/CR-4177 V01: MANAGEMENT OF SEVERE ACCIDENTS.Perspectives On Managing Severe Accidents In Commercial Nuclear Power Plants.
NUREG/CR-4177 V02: MANAGEMENT OF SEVERE ACCIDENTS.Extending Plant Operating Procedures Into The Severe Accident Regime.
NUREG/CR-4185: AN ASSESSMENT OF DOSIMETRY DATA FOR ACCIDENTAL RADIONUCLIDE RELEASES FROM NUCLEAR REACTORS.
NUREG/CR-4197: SAFETY GOAL SENSITIVITY STUDIES.
NUREG/CR-4199: A DEMONSTRATION UNCERTAINTY/SENSITIVITY ANALYSIS USING THE HEALTH AND ECONOMIC CONSEQUENCE MODEL CRAC2.
NUREG/CR-4206: A SELECT REVIEW OF THE RECENT (1979-1983) BEHAVIORAL RESEARCH LITERATURE ON TRAINING SIMULATORS.
NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS.
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.
NUREG/CR-4213: SETS REFERENCE MANUAL.
NUREG/CR-4214: HEALTH EFFECTS MODEL FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Part I:Introduction,Integration & Summary.Part II:Scientific Basis For Health Effects Models.
NUREG/CR-4217: A STATISTICAL ANALYSIS OF NUCLEAR POWER PLANT VALVE FAILURE-RATE VARIABILITY--SOME PRELIMINARY RESULTS.
NUREG/CR-4225: SUMMARY OF EFFICIENCY TESTING OF STANDARD AND HIGH-CAPACITY HIGH-EFFICIENCY PARTICULATE AIR FILTERS SUBJECTED TO SIMULATED TORNADO DEPRESSURIZATION AND EXPLOSIVE SHOCK WAVES.
NUREG/CR-4227: HUMAN ENGINEERING GUIDELINES FOR THE EVALUATION AND ASSESSMENT OF VIDEO DISPLAY UNITS.
NUREG/CR-4232: THE RESPONSE OF VENTILATION DAMPERS TO LARGE AIRFLOW PULSES.
NUREG/CR-4260: TORAC USER'S MANUAL.A Computer Code For Analyzing Tornado-Induced Flow And Material Transport In Nuclear Facilities.
NUREG/CR-4264: INVESTIGATION ON HIGH-EFFICIENCY PARTICULATE AIR FILTER PLUGGING BY COMBUSTION AEROSOLS.
NUREG/CR-4272: RESPONSE TREE EVALUATION:EXPERIMENTAL ASSESSMENT OF AN EXPERT SYSTEM FOR NUCLEAR REACTOR OPERATORS.
NUREG/CR-4280: THE EFFECTS OF SUPERVISOR EXPERIENCE AND ASSISTANCE OF A SHIFT TECHNICAL ADVISOR (STA) ON CREW PERFORMANCE IN CONTROL ROOM SIMULATORS.
NUREG/CR-4298: DESIGN AND INSTALLATION OF COMPUTER SYSTEMS TO MEET THE REQUIREMENTS OF 10 CFR 73.55.
NUREG/CR-4314: BRIEF SURVEY AND COMPARISON OF COMMON CAUSE FAILURE ANALYSIS.
NUREG/CR-4321: FULL SCALE MEASUREMENTS OF SMOKE TRANSPORT AND DEPOSITION IN VENTILATION SYSTEM DUCTWORK.
NUREG/CR-4342: UNCERTAINTY AND SENSITIVITY ANALYSIS OF A MODEL FOR MULTICOMPONENT AEROSOL DYNAMICS.
NUREG/CR-4350 V01: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 1 - PRA Fundamentals.
NUREG/CR-4350 V02: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 2-Probability And Statistics For PRA Applications.
NUREG/CR-4350 V03: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 3 - Systems Reliability And Analysis Techniques,Session A -Reliability.
NUREG/CR-4350 V04: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 4 - System Reliability And Analysis Techniques,Sessions B/C - Event Trees/Fault Trees.

NUREG/CR-4350 V05: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 5 - Systems Reliability And Analysis Techniques,Session D - Quantification.
NUREG/CR-4350 V06: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 6 - Data Development.
NUREG/CR-4350 V07: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 7 - Environmental Transport And Consequence Analysis.
NUREG/CR-4360 V01: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.
NUREG/CR-4360 V02: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.
NUREG/CR-4377: EVALUATIONS AND UTILIZATIONS OF RISK IMPORTANCES.
NUREG/CR-4385: EFFECTS OF CONTROL SYSTEM FAILURES IN TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A WESTINGHOUSE PRESSURIZED WATER REACTOR.
NUREG/CR-4386: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A BABCOCK AND WILCOX PRESSURIZED WATER REACTOR.
NUREG/CR-4387: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS AND CORE-MELT FREQUENCIES AT A GENERAL ELECTRIC PRESSURIZED WATER REACTOR.

DIVISION OF RADIATION PROGRAMS & EARTH SCIENCES (POST 840429)

NUREG/CR-2663 V01: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Main Report.
NUREG/CR-2663 V02: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Appendices.
NUREG/CR-3145 V03: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION - ANNUAL REPORT.(October 1982 - September 1983, Volume 3).
NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983.
NUREG/CR-3178: STRUCTURAL AND TECTONIC STUDIES IN NEW YORK STATE.Final Report,July 1981 - June 1982.
NUREG/CR-3413: OFF-SITE CONSEQUENCES OF RADIOLOGICAL ACCIDENTS:METHODS, COSTS AND SCHEDULES FOR DECONTAMINATION.
NUREG/CR-3442: RADTWO:A COMPUTER CODE FOR SIMULATING FAST-TRANSIENT, TWO-DIMENSIONAL,TWO-LAYER RADIONUCLIDE CONCENTRATION CONDITIONS IN LAKES,RESERVOIRS,RIVERS,ESTUARIES,AND COASTAL REGIONS.
NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.
NUREG/CR-3609: EVALUATION OF NEUTRON DOSIMETRY TECHNIQUES FOR WELL-LOGGING OPERATIONS.
NUREG/CR-3657: PRELIMINARY SCREENING OF FUEL CYCLE AND BY-PRODUCT MATERIAL LICENSES FOR EMERGENCY PLANNING.
NUREG/CR-3709: METHODS OF MINIMIZING GROUND-WATER CONTAMINATION FROM IN SITU LEACH URANIUM MINING.Final Report.
NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT.
NUREG/CR-3736: FIELD AND THEORETICAL INVESTIGATIONS OF FRACTURED CRYSTALLINE ROCK NEAR ORACLE,ARIZONA.
NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS.
NUREG/CR-3752: EFFECTS OF HYDROLOGIC VARIABLES ON ROCK RIPRAP DESIGN FOR URANIUM TAILINGS IMPOUNDMENTS.
NUREG/CR-3900 V02: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,July-September 1984.
NUREG/CR-3900 V03: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,October-December 1984.
NUREG/CR-3900 V04: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Annual Report,April 1984 - April 1985.
NUREG/CR-3906: URANIUM MILL TAILINGS NEUTRALIZATION:CONTAMINANT COMPLEXATION AND TAILINGS LEACHING STUDY.
NUREG/CR-3953: THE USE OF MAG-1 SPECTACLES WITH POSITIVE- AND NEGATIVE-PRESSURE RESPIRATORS.
NUREG/CR-3972: SETTLEMENT OF URANIUM MILL TAILINGS PILES: A COMPARISON OF ANALYSIS TECHNIQUES.
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.
NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984.
NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER.(Final Report)
NUREG/CR-4033: THE ROLE OF PERSONAL AIR SAMPLING IN RADIATION SAFETY PROGRAMS AND RESULTS OF A LABORATORY EVALUATION OF PERSONAL AIR-SAMPLING EQUIPMENT.
NUREG/CR-4042: A 3-DIMENSIONAL COMPUTER MODEL TO SIMULATE FLUID FLOW AND CONTAINMENT TRANSPORT THROUGH A ROCK FRACTURE SYSTEM.
NUREG/CR-4045: LITERATURE REVIEW ON AEROSOL-SAMPLING DEVICES FOR RESPIRATORY FIELD STUDIES.
NUREG/CR-4061: LEACHATE PLUME MIGRATION DOWNGRADIENT FROM URANIUM TAILINGS DISPOSAL IN MINE STOPES.
NUREG/CR-4070 V02: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Volume 2:Current Status Of Biofouling Surveillance And Control Techniques.
NUREG/CR-4070 V03: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Factors That May Intensify The Safety Consequences Of Biofouling.
NUREG/CR-4072: THE ESTIMATION OF ATMOSPHERIC DISPERSION AT NUCLEAR POWER PLANTS UTILIZING REAL TIME ANEMOMETER STATISTICS.
NUREG/CR-4075: DESIGNING PROTECTIVE COVERS FOR URANIUM MILL TAILINGS PILES: A Review.
NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION.
NUREG/CR-4087: MEASUREMENTS OF URANIUM MILL TAILINGS CONSOLIDATION CHARACTERISTICS.
NUREG/CR-4088: METHODS FOR ESTIMATING RADIOACTIVE AND TOXIC AIRBORNE SOURCE TERMS FOR URANIUM MILLING OPERATIONS.
NUREG/CR-4089: EVALUATION OF FIELD-TESTED FUGITIVE DUST CONTROL TECHNIQUES FOR URANIUM MILL TAILINGS PILES.
NUREG/CR-4094: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN SULFATE LAKE ENVIRONMENTS.
NUREG/CR-4100: EVALUATION OF INSTRUMENTAL METHODS FOR THE MEASUREMENT OF YELLOWCAKE EMISSIONS.
NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET-GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE,AND SODIUM CHLORIDE.
NUREG/CR-4114: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS.II.
NUREG/CR-4117: FAULTING AND JOINTING IN AND NEAR SURFACE MINES OF SOUTHWESTERN INDIANA.
NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.
NUREG/CR-4120: MATHEMATICAL MODELING OF ULTIMATE HEAT SINK COOLING PONDS.
NUREG/CR-4131: INVESTIGATION OF ALTERNATIVE MEANS TO ACCOMPLISH THE GOALS OF BIENNIAL ION CHAMBER CALIBRATION.
NUREG/CR-4145: EARTHQUAKE RECURRENCE INTERVALS AT NUCLEAR POWER PLANTS: ANALYSIS AND RANKING.
NUREG/CR-4157: A SCIENTIFIC CRITIQUE OF AVAILABLE MODELS FOR REAL-TIME SIMULATIONS OF DISPERSION.
NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING.
NUREG/CR-4159: COMPARISON OF THE 1981 INEL DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS.
NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt.
NUREG/CR-4166: ANALYSIS OF FLECHT-SEASET 163-ROD BLOCKED BUNDLE DATA USING COBRA-TF.
NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.
NUREG/CR-4176: EMISSION CONTROL TECHNOLOGY AND QUALITY ASSURANCE NEEDS AT URANIUM MILLING FACILITIES.Includes Supporting Methods For Testing,Operating,And Maintaining Air Pollution Control Devices.
NUREG/CR-4181: LEACHABILITY OF RADIONUCLIDES FROM CEMENT SOLIDIFIED WASTE FORMS PRODUCED AT OPERATING NUCLEAR POWER REACTORS.
NUREG/CR-4192: THE ANALYSIS OF DRAINAGE AND CONSOLIDATION AT TYPICAL URANIUM MILL TAILINGS SITES.

NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984.
NUREG/CR-4198: FRACTURE IN GLASS/HIGH LEVEL WASTE CANISTERS.
NUREG/CR-4203: A CALCULATIONAL METHOD FOR DETERMINING BIOLOGICAL DOSE RATES FROM IRRADIATED RESEARCH REACTOR FUEL.
NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS.Application To Establishing Values Of f1 For Soluble Plutonium.
NUREG/CR-4226: NEW MADRID SEISMOTECTONIC STUDY.Activities During Fiscal Year 1983.
NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.
NUREG/CR-4239: ANALYSIS OF THE ABILITY OF CURRENT HEALTH PHYSICS INSTRUMENTS TO PREDICT DOSE IN EXPOSED INDIVIDUALS.
NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUNDWATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions.
NUREG/CR-4251 V02: MITIGATIVE TECHNIQUES FOR GROUNDWATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 2:Case Study Analysis Of Hydrologic Characterization And Mitigative Schemes.
NUREG/CR-4254: OCCUPATIONAL DOSE REDUCTION AND ALARA AT NUCLEAR POWER PLANTS.Study On High-Dose Jobs,Radwaste Handling,And ALARA Incentives.
NUREG/CR-4259: TAILINGS NEUTRALIZATION AND OTHER ALTERNATIVES FOR IMMOBILIZING TOXIC MATERIALS IN TAILINGS.Final Report.
NUREG/CR-4266: STANDARD BETA-PARTICLE AND MONOENERGETIC ELECTRON SOURCES FOR THE CALIBRATION OF BETA-RADIATION PROTECTION INSTRUMENTATION.
NUREG/CR-4268: RATIO METHODS FOR COST-EFFECTIVE FIELD SAMPLING OF COMMERCIAL RADIOACTIVE LOW-LEVEL WASTES.
NUREG/CR-4288: FOCAL MECHANISM ANALYSES FOR VIRGINIA AND EASTERN TENNESSEE EARTHQUAKES (1978-1984).
NUREG/CR-4297: EXTREMITY MONITORING:Considerations For Use,Dosimeter Placement,And Evaluation.
NUREG/CR-4303: HIGH-LEVEL WASTE PRECLOSURE SYSTEMS SAFETY ANALYSIS.Phase 1. Final Report.
NUREG/CR-4317 V01: CANADIAN SEISMIC AGREEMENT.Technical Report Covering 1979-1985.
NUREG/CR-4333: STE. GENEVIEVE FAULT ZONE,MISSOURI AND ILLINOIS.
NUREG/CR-4352: SUGGESTED STATE REQUIREMENTS AND CRITERIA FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE REGULATORY PROGRAM.
NUREG/CR-4354: A STUDY OF SEISMICITY AND TECTONICS IN NEW ENGLAND.Final Report.
NUREG/CR-4355 V01: 238 PU(IV) IN MONKEYS.Overview Of Metabolism.
NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.
NUREG/CR-4379 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report,Year Four April-June 1985.
NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.
NUREG/CR-4430: CURRENT METHODOLOGIES FOR ASSESSING THE POTENTIAL FOR EARTHQUARKE-INDUCED LIQUEFACTION IN SOILS.
NUREG/CR-4435: ORGANIC COMPLEXANT-ENHANCED MOBILITY OF TOXIC ELEMENTS IN LOW-LEVEL WASTES.Annual Report,July 1984 - June 1985.

EARTH SCIENCES BRANCH (POST 840429)
NUREG/CR-4339: A REVIEW OF RECENT RESEARCH ON THE SEISMOTECTONICS OF THE SOUTHEASTERN SEABOARD AND AN EVALUATION OF HYPOTHESES ON THE SOURCE OF THE 1886 CHARLESTON,SOUTH CAROLINA EARTHQUAKE.

DIVISION OF ENGINEERING TECHNOLOGY
NUREG/CR-1677 V02: PIPING BENCHMARK PROBLEMS,VOLUME II DYNAMIC ANALYSIS INDEPENDENT SUPPORT MOTION RESPONSE SPECTRUM METHOD.
NUREG/CR-1755 ADD01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING NUCLEAR REACTORS AT MULTIPLE-REACTOR STATIONS.Effects On Decommissioning Of Interim Inability To Dispose Of Wastes Offsite.
NUREG/CR-3228 V03: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1984.
NUREG/CR-3293 V01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Main Report.
NUREG/CR-3293 V02: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Appendices.
NUREG/CR-3317: TECHNICAL BASES AND USER'S MANUAL FOR THE PROTOTYPE OF SPARC - A SUPPRESSION POOL AEROSOL REMOVAL CODE.
NUREG/CR-3319: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium.
NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.
NUREG/CR-3558: HANDBOOK OF NUCLEAR POWER PLANT SEISMIC FRAGILITIES. Seismic Safety Margins Research Program.
NUREG/CR-3613 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Annual Report For 1984.
NUREG/CR-3613 V03 N1: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Semiannual Report For October 1984 Through March 1985.
NUREG/CR-3638: HYDROGEN-STEAM JET-FLAME FACILITY AND EXPERIMENTS.
NUREG/CR-3647: DESIGN AND FABRICATION OF A 1/8-SCALE STEEL CONTAINMENT MODEL.
NUREG/CR-3660 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 1:Summary Report.
NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3660 V04: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 4:Pipe Failure Induced By Crack Growth In West Coast Plants.
NUREG/CR-3663 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 1:Summary Report.
NUREG/CR-3663 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 3:Double Ended Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3723: STRESS-INTENSITY-FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS.
NUREG/CR-3744 V02: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR APRIL-SEPTEMBER 1984.
NUREG/CR-3746 V02: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.Semiannual Progress Report,April 1984 - September 1984.
NUREG/CR-3746 V03: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.1984 Annual Report,October 1,1983 - September 30, 1984.
NUREG/CR-3805 V02: ENGINEERING CHARACTERIZATION OF GROUND MOTION.Task II: Effects Of Ground Motion Characteristics On Structural Response Considering Localized Structural Nonlinearities And Soil-Structure Interaction Effects.
NUREG/CR-3810 V03: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,July-September 1984.
NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,October-December 1984.
NUREG/CR-3819: SURVEY OF AGED POWER PLANT FACILITIES.
NUREG/CR-3825 V03-4: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Quarterly Report, April 1984 - September 1984.Volumes 3 and 4.
NUREG/CR-3854: FABRICATION CRITERIA FOR SHIPPING CONTAINERS.
NUREG/CR-3855: CHARACTERIZATION OF NUCLEAR REACTOR CONTAINMENT PENETRATION - FINAL REPORT.
NUREG/CR-3863: ASSESSMENT OF CLASS 1E PRESSURE TRANSMITTER RESPONSE WHEN SUBJECTED TO HARSH ENVIRONMENT SCREENING TESTS.
NUREG/CR-3872: DATA ACQUISITION AND CONTROL OF THE HSST SERIES V IRRADIATION EXPERIMENT AT THE ORR.
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.

NUREG/CR-3911 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Quarterly Report,April-June 1984.

NUREG/CR-3915: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL.

NUREG/CR-3945: FATIGUE CRACK GROWTH RATES OF LOW-CARBON AND STAINLESS PIPING STEELS IN PWR ENVIRONMENT.

NUREG/CR-3949 V01: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Semiannual Progress Report For Period Ending June 30,1984.

NUREG/CR-3949 V02: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM. Annual Progress Report For Period Ending December 31,1984.

NUREG/CR-3978: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL PLATE AND WELDS FOR THE FOURTH HSST IRRADIATION SERIES.

NUREG/CR-3991: FAILURE MODES AND EFFECTS ANALYSIS (FMEA) OF THE ICS/NNI ELECTRIC POWER DISTRIBUTION CIRCUITRY AT THE OCONEE-1 NUCLEAR PLANT.

NUREG/CR-3998 V02: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,April-June 1984.

NUREG/CR-3998 V03: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,October-December 1984.

NUREG/CR-4008: GENERAL EXTRAPOLATION MODEL FOR AN IMPORTANT CHEMICAL DOSE-RATE EFFECT.

NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.

NUREG/CR-4031 V02: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutronics Calculations."

NUREG/CR-4031 V03: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutron Exposure Parameters."

NUREG/CR-4039: GAMMA-RAY CHARACTERIZATION OF THE TWO-YEAR IRRADIATION EXPERIMENT PERFORMED AT THE POOLSIDE FACILITY.

NUREG/CR-4056: PARTICULATE AND GAS RELEASE FROM LIGHT-WATER-REACTOR (LWR) FUEL RODS STORED IN INERT AND DRY AIR ATMOSPHERES.

NUREG/CR-4064: STRUCTURAL RESPONSE OF LARGE PENETRATIONS AND CLOSURES FOR CONTAINMENT VESSELS SUBJECTED TO LOADINGS BEYOND DESIGN BASIS.

NUREG/CR-4074: THE PERFORMANCE OF DEFECTED SPENT LWR FUEL RODS IN INERT GAS AND DRY AIR STORAGE ATMOSPHERES.

NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT.

NUREG/CR-4080: DETERMINATION OF THE AVAILABILITY OF CORE EXIT THERMOCOUPLES DURING SEVERE ACCIDENT SITUATIONS.

NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.

NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.

NUREG/CR-4084: DRY SPENT FUEL STORAGE TEST PLAN FOR DESTRUCTIVE FUEL ROD EXAMINATIONS.

NUREG/CR-4086: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL WELDS FOR THE THIRD HSST IRRADIATION SERIES.

NUREG/CR-4090: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Annual Summary Report - Fiscal Year 1984.

NUREG/CR-4091: THE EFFECT OF ALTERNATIVE AGING AND ACCIDENT SIMULATIONS ON POLYMER PROPERTIES.

NUREG/CR-4092: ORNL CHARACTERIZATION OF HEAVY-SECTION STEEL TECHNOLOGY PROGRAM PLATES 01,02,AND 03.

NUREG/CR-4095: TEST SERIES 2:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E EXIDE FHC-19 BATTERY CELLS.

NUREG/CR-4096: TEST SERIES 3:SEISMIC FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E C&D LCU-13 BATTERY CELLS.

NUREG/CR-4097: TEST SERIES 4:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED EXIDE EMP-13 BATTERY CELLS.

NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.

NUREG/CR-4119: INTEGRITY OF CONTAINMENT PENETRATIONS UNDER SEVERE ACCIDENT CONDITIONS FY84 ANNUAL REPORT.

NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS.

NUREG/CR-4123: SEISMIC FRAGILITY OF REINFORCED CONCRETE STRUCTURES AND COMPONENTS FOR APPLICATION TO NUCLEAR FACILITIES.

NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984.

NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.

NUREG/CR-4137: PRETEST PREDICTIONS FOR THE RESPONSE OF A 1:8-SCALE STEEL LWR CONTAINMENT BUILDING MODEL TO STATIC OVERPRESSURIZATION.

NUREG/CR-4138: DATA ANALYSES FOR NEVADA TEST SITE (NTS) PREMIXED COMBUSTION TESTS.

NUREG/CR-4141: CONTAINMENT PURGE AND VENT VALVE TEST PROGRAM FINAL REPORT.

NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.

NUREG/CR-4147: THE EFFECT OF ENVIRONMENTAL STRESS ON SYLGARD 70 SILICONE ELASTOMER.

NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.

NUREG/CR-4170: AN ULTRA-HIGH SPEED RESIDUE PROCESSOR FOR SAFT INSPECTION SYSTEM IMAGE ENHANCEMENT.

NUREG/CR-4180: STATE-OF-THE-ART OF SOLID-STATE MOTOR CONTROLLERS.

NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS.

NUREG/CR-4204: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Annual Report,October 1983 - September 1984.

NUREG/CR-4209: COMPARISON OF ANALYTICAL PREDICTIONS AND EXPERIMENTAL RESULTS FOR A 1:8-SCALE STEEL CONTAINMENT MODEL PRESSURIZED TO FAILURE.

NUREG/CR-4212: IN-PLACE THERMAL ANNEALING OF NUCLEAR REACTOR PRESSURE VESSELS.

NUREG/CR-4219 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.

NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.

NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION.

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NUREG/CR-4290 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF BABCOCK AND WILCOX PWR PLANTS.Volume 2:Guillotine Break Indirectly Induced By Earthquakes.

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NUREG/CR-4300 V01: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Progress Report, October-March 1985.
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NUREG/CR-4318 V02: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,April-June 1985.
NUREG/CR-4325: A PARAMETRIC STUDY OF PWR PRESSURE VESSEL INTEGRITY DURING OVERCOOLING ACCIDENTS,CONSIDERING BOTH 2-D AND 3-D FLAWS.
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NUREG/CR-4362: STEAM GENERATOR GROUP PROJECT.Annual Report - 1984.
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NUREG/CR-4357: THE FEASIBILITY OF DETECTING THE IMPORT OF UNAUTHORIZED RADIOACTIVE MATERIALS INTO THE UNITED STATES.

AMES LABORATORY, ENERGY & MINERAL RESOURCES RESEARCH INSTITUTE

NUREG/CR-3952: SEQUOYAH EQUIPMENT HATCH SEAL LEAKAGE.

ANALYSIS & TECHNOLOGY, INC.

NUREG/CR-4344: INSTRUCTIONAL SKILLS EVALUATION IN NUCLEAR INDUSTRY TRAINING.

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NUREG/CR-3710: LABORATORY STUDIES OF A BREACHED NUCLEAR WASTE REPOSITORY IN BASALT.

NUREG/CR-3804 V03: PHYSICS OF REACTOR SAFETY.Quarterly Report,July-September 1984.

NUREG/CR-3804 V04: PHYSICS OF REACTOR SAFETY.Quarterly Report,October-December 1984.

NUREG/CR-3855: CHARACTERIZATION OF NUCLEAR REACTOR CONTAINMENT PENETRATION - FINAL REPORT.

NUREG/CR-3980 V02: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,April-June 1984.

NUREG/CR-3980 V03: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,July-September 1984.

NUREG/CR-3980 V04: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,October-December 1984.

NUREG/CR-3989: TIME- AND VOLUME-AVERAGED CONSERVATION EQUATIONS FOR MULTIPHASE FLOW.Part One:System Without Internal Solid Structures.

NUREG/CR-3998 V02: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,April-June 1984.

NUREG/CR-3998 V03: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,October-December 1984.

NUREG/CR-4064: STRUCTURAL RESPONSE OF LARGE PENETRATIONS AND CLOSURES FOR CONTAINMENT VESSELS SUBJECTED TO LOADINGS BEYOND DESIGN BASIS.

NUREG/CR-4120: MATHEMATICAL MODELING OF ULTIMATE HEAT SINK COOLING PONDS.

NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984.

NUREG/CR-4180: STATE-OF-THE-ART OF SOLID-STATE MOTOR CONTROLLERS.

NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.

NUREG/CR-4204: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Annual Report,October 1983 - September 1984.

NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS.Application To Establishing Values Of f1 For Soluble Plutonium.

NUREG/CR-4240 V01: PHYSICS OF REACTOR SAFETY.Quarterly Report,January-March 1985.

NUREG/CR-4277: INVERTED ANNUAL FLOW EXPERIMENTAL STUDY.

NUREG/CR-4287: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Annual Report,October 1983 - September 1984.

NUREG/CR-4368: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Semiannual Report,October 1984 - March 1985.

NUREG/CR-4432: COMPARISON OF DYNAMIC CHARACTERISTICS OF FUKUSHIMA NUCLEAR POWER PLANT CONTAINMENT BUILDING DETERMINED FROM TESTS AND EARTHQUAKES.

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NUREG/CR-4174: ROCK MASS SEALING - EXPERIMENTAL ASSESSMENT OF BOREHOLE PLUG PERFORMANCE.Annual Report,June 1983 - May 1984.

NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984.

ARMY, DEPT. OF, ARMY ENGINEER WATERWAYS EXPERIMENT STATION

NUREG/CR-3774 V02: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2A:Technical Requirements For Belowground Vault Disposal Of Low Level Radioactive Waste.

NUREG/CR-3774 V03: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTES.Task 2B:Technical Requirements For Aboveground Vault Disposal Of Low Level Radioactive Waste.

NUREG/CR-3774 V04: ALTERNATIVE METHOD FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2C:Technical Requirements For Earth Mounded Concrete Bunker Disposal Of Low Level Radioactive Waste.

NUREG/CR-3774 V05: ALTERNATIVE METHODS FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE.Task 2E:Technical Requirements For Shaft Disposal Of Low Level Radioactive Waste.

NUREG/CR-4430: CURRENT METHODOLOGIES FOR ASSESSING THE POTENTIAL FOR EARTHQUARKE-INDUCED LIQUEFACTION IN SOILS.

ARVIN/CALSPAN ADVANCED TECHNOLOGY CENTER

NUREG/CR-4257: INSPECTION,SURVEILLANCE,AND MONITORING OF ELECTRICAL EQUIPMENT INSIDE CONTAINMENT OF NUCLEAR POWER PLANTS--WITH APPLICATIONS TO ELECTRICAL CABLES.

ASPEN SYSTEMS, INC.

NUREG-0386 D03: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.JULY 1972 - SEPTEMBER 1983.

ATOMIC ENERGY OF CANADA, LTD.

NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT.

BATTELLE HUMAN AFFAIRS RESEARCH CENTERS

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NUREG/CR-4125 V01: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER

PLANT.Volume 1:Guidelines For Utility Organization And Administration Plan.
NUREG/CR-4125 V02: GUIDELINES AND WORKBOOK FOR ASSESSMENT OF ORGANIZATION AND ADMINISTRATION OF UTILITIES SEEKING OPERATING LICENSE FOR A NUCLEAR POWER PLANT.Volume 2:Workbook For Assessment Of Organization And Management.
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BATTELLE MEMORIAL INSTITUTE, COLUMBUS LABORATORIES
NUREG/CR-3900 V02: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,July-September 1984.
NUREG/CR-3900 V03: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,October-December 1984.
NUREG/CR-3900 V04: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Annual Report,April 1984 - April 1985.
NUREG/CR-3937: STEAM GENERATOR TUBE RUPTURE IODINE TRANSPORT MECHANISMS. Task 1 Experimental Studies.
NUREG/CR-4082 V01: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report,March 1984 - September 1984.
NUREG/CR-4082 V02: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, October 1984 - March 1985.
NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.
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NUREG/CR-4379 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report,Year Four April-June 1985.
NUREG/CR-4388: AEROSOL BEHAVIOR MODELING (TASK 3) - SUPPORT SERVICES FOR RESEARCH AND EVALUATION OF SEVERE ACCIDENT PHENOMENA AND MITIGATION.

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NUREG/CR-1755 ADD01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING NUCLEAR REACTORS AT MULTIPLE-REACTOR STATIONS.Effects On Decommissioning Of Interim Inability To Dispose Of Wastes Offsite.
NUREG/CR-2800 S03: GUIDELINES FOR NUCLEAR POWER PLANT SAFETY ISSUE PRIORITIZATION INFORMATION DEVELOPMENT.
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NUREG/CR-3293 V01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Main Report.
NUREG/CR-3293 V02: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Appendices.
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NUREG/CR-3609: EVALUATION OF NEUTRON DOSIMETRY TECHNIQUES FOR WELL-LOGGING OPERATIONS.
NUREG/CR-3613 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Annual Report For 1984.
NUREG/CR-3613 V03 N1: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Semiannual Report For October 1984 Through March 1985.
NUREG/CR-3659: A MATHEMATICAL MODEL FOR ASSESSING THE UNCERTAINTIES OF INSTRUMENTATION MEASUREMENTS FOR POWER AND FLOW OF PWR REACTORS.
NUREG/CR-3709: METHODS OF MINIMIZING GROUND-WATER CONTAMINATION FROM IN SITU LEACH URANIUM MINING.Final Report.
NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS.
NUREG/CR-3752: EFFECTS OF HYDROLOGIC VARIABLES ON ROCK RIPRAP DESIGN FOR URANIUM TAILINGS IMPOUNDMENTS.
NUREG/CR-3810 V03: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,July-September 1984.
NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,October-December 1984.
NUREG/CR-3825 V03-4: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Quarterly Report, April 1984 - September 1984.Volumes 3 and 4.
NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE.
NUREG/CR-3906: URANIUM MILL TAILINGS NEUTRALIZATION:CONTAMINANT COMPLEXATION AND TAILINGS LEACHING STUDY.
NUREG/CR-3911 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Quarterly Report,April-June 1984.
NUREG/CR-3915: ACOUSTIC EMISSION RESULTS OBTAINED FROM TESTING THE ZB-1 INTERMEDIATE SCALE PRESSURE VESSEL.
NUREG/CR-3950 V01: FUEL PERFORMANCE ANNUAL REPORT FOR 1983.
NUREG/CR-3972: SETTLEMENT OF URANIUM MILL TAILINGS PILES: A COMPARISON OF ANALYSIS TECHNIQUES.
NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS.
NUREG/CR-3999: ELECTRICALLY HEATED EX-REACTOR PELLET-CLADDING INTERACTION (PCI) SIMULATIONS UTILIZING IRRADIATED ZIRCALOY CLADDING.
NUREG/CR-4023: FIELD PERFORMANCE ASSESSMENT OF SYNTHETIC LINERS FOR URANIUM TAILINGS POND.A Status Report.
NUREG/CR-4030: RADIONUCLIDE MIGRATION IN GROUND WATER.(Final Report)
NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.
NUREG/CR-4057: RADIOLOGICAL ASSESSMENT OF THE TOWN OF EDGEMONT.
NUREG/CR-4061: LEACHATE PLUME MIGRATION DOWNGRADIENT FROM URANIUM TAILINGS DISPOSAL IN MINE STOPES.
NUREG/CR-4070 V02: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Volume 2:Current Status Of Biofouling Surveillance And Control Techniques.
NUREG/CR-4070 V03: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Factors That May Intensify The Safety Consequences Of Biofouling.
NUREG/CR-4075: DESIGNING PROTECTIVE COVERS FOR URANIUM MILL TAILINGS PILES: A Review.
NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION.
NUREG/CR-4087: MEASUREMENTS OF URANIUM MILL TAILINGS CONSOLIDATION CHARACTERISTICS.
NUREG/CR-4088: METHODS FOR ESTIMATING RADIOACTIVE AND TOXIC AIRBORNE SOURCE TERMS FOR URANIUM MILLING OPERATIONS.
NUREG/CR-4089: EVALUATION OF FIELD-TESTED FUGITIVE DUST CONTROL TECHNIQUES FOR URANIUM MILL TAILINGS PILES.
NUREG/CR-4100: EVALUATION OF INSTRUMENTAL METHODS FOR THE MEASUREMENT OF YELLOWCAKE EMISSIONS.
NUREG/CR-4108: DEVELOPMENT OF MC&A ALARM RESOLUTION PROCEDURES.
NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.
NUREG/CR-4130: ICEDF:A CODE FOR AEROSOL PARTICLE CAPTURE IN ICE COMPARTMENTS.
NUREG/CR-4139: THE MAILED SURVEY:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL.
NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.
NUREG/CR-4151: INTEGRATION OF EMERGENCY ACTION LEVELS WITH COMBUSTION ENGINEERING EMERGENCY OPERATING

PROCEDURES.By Use Of Combustion Engineering Owners Group Emergency Operating Procedure Technical Guidelines.
NUREG/CR-4153: APPLICATIONS OF FOREIGN PROBABILISTIC SAFETY ASSESSMENT EXPERIENCE TO THE U.S. NUCLEAR REGULATORY PROCESS.
NUREG/CR-4160: HISTORICAL SUMMARY OF OCCUPATIONAL RADIATION EXPOSURE EXPERIENCE IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.
NUREG/CR-4166: ANALYSIS OF FLECHT-SEASET 163-ROD BLOCKED BUNDLE DATA USING COBRA-TF.
NUREG/CR-4168: GT2F:A COMPUTER CODE FOR ESTIMATING LIGHT WATER REACTOR FUEL ROD FAILURES.
NUREG/CR-4176: EMISSION CONTROL TECHNOLOGY AND QUALITY ASSURANCE NEEDS AT URANIUM MILLING FACILITIES.Includes Supporting Methods For Testing,Operating,And Maintaining Air Pollution Control Devices.
NUREG/CR-4192: THE ANALYSIS OF DRAINAGE AND CONSOLIDATION AT TYPICAL URANIUM MILL TAILINGS SITES.
NUREG/CR-4218: LOCA SIMULATION IN THE NATIONAL RESEARCH UNIVERSAL REACTOR PROGRAM.Postirradiation Examination Results For The Third Materials Test (MT-3) - Second Campaign.
NUREG/CR-4220: RELIABILITY ANALYSIS OF CONTAINMENT ISOLATION SYSTEMS.
NUREG/CR-4248: RECOMMENDATIONS FOR NRC POLICY ON SHIFT SCHEDULING AND OVERTIME AT NUCLEAR POWER PLANTS.
NUREG/CR-4251 V01: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 1:Analysis Of Generic Site Conditions.
NUREG/CR-4251 V02: MITIGATIVE TECHNIQUES FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE NUCLEAR ACCIDENTS.Volume 2:Case Study Analysis Of Hydrologic Characterization And Mitigative Schemes.
NUREG/CR-4259: TAILINGS NEUTRALIZATION AND OTHER ALTERNATIVES FOR IMMOBILIZING TOXIC MATERIALS IN TAILINGS.Final Report.
NUREG/CR-4267: VESSEL INTEGRITY SIMULATION (VISA) CODE SENSITIVITY STUDY.
NUREG/CR-4268: RATIO METHODS FOR COST-EFFECTIVE FIELD SAMPLING OF COMMERCIAL RADIOACTIVE LOW-LEVEL WASTES.
NUREG/CR-4276: VIBRATION AND WEAR IN STEAM GENERATOR TUBES FOLLOWING CHEMICAL CLEANING - SEMIANNUAL REPORT.
NUREG/CR-4281: AN EMPIRICAL ANALYSIS OF SELECTED NUCLEAR POWER PLANT MAINTENANCE FACTORS AND PLANT SAFETY.
NUREG/CR-4297: EXTREMITY MONITORING:Considerations For Use,Dosimeter Placement,And Evaluation.
NUREG/CR-4298: DESIGN AND INSTALLATION OF COMPUTER SYSTEMS TO MEET THE REQUIREMENTS OF 10 CFR 73.55.
NUREG/CR-4300 V01: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Progress Report, October-March 1985.
NUREG/CR-4318 V01: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,January-March 1985.
NUREG/CR-4318 V02: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,April-June 1985.
NUREG/CR-4361: STEAM GENERATOR GROUP PROJECT.Annual Report - 1983.
NUREG/CR-4362: STEAM GENERATOR GROUP PROJECT.Annual Report - 1984.
NUREG/CR-4385: EFFECTS OF CONTROL SYSTEM FAILURES IN TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A WESTINGHOUSE PRESSURIZED WATER REACTOR.
NUREG/CR-4386: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS, AND CORE-MELT FREQUENCIES AT A BABCOCK AND WILCOX PRESSURIZED WATER REACTOR.
NUREG/CR-4387: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS,ACCIDENTS AND CORE-MELT FREQUENCIES AT A GENERAL ELECTRIC PRESSURIZED WATER REACTOR.
NUREG/CR-4399: POSSIBLE OPTIONS FOR REDUCING OCCUPATIONAL DOSE FROM THE TMI-2 BASEMENT.
NUREG/CR-4435: ORGANIC COMPLEXANT-ENHANCED MOBILITY OF TOXIC ELEMENTS IN LOW-LEVEL WASTES.Annual Report,July 1984 - June 1985.

BOSTON COLLEGE, CHESTNUT HILL, MA

NUREG/CR-3178: STRUCTURAL AND TECTONIC STUDIES IN NEW YORK STATE.Final Report,July 1981 - June 1982.
NUREG/CR-4354: A STUDY OF SEISMICITY AND TECTONICS IN NEW ENGLAND.Final Report.

BROOKHAVEN NATIONAL LABORATORY

NUREG/CP-0059 V01: PROCEEDINGS OF THE MITI-NRC SEISMIC INFORMATION EXCHANGE MEETING.VOLUME I.
NUREG/CP-0066: PROCEEDINGS OF AN INTERNATIONAL WORKSHOP ON HISTORIC DOSE EXPERIENCE AND DOSE REDUCTION (ALARA) AT NUCLEAR POWER PLANTS.MAY 29-JUNE 1,1984.
NUREG/CP-0070: PROCEEDINGS OF THE WORKSHOP ON SEISMIC AND DYNAMIC FRAGILITY OF NUCLEAR POWER PLANT COMPONENTS.
NUREG/CR-1677 V02: PIPING BENCHMARK PROBLEMS.VOLUME II DYNAMIC ANALYSIS INDEPENDENT SUPPORT MOTION RESPONSE SPECTRUM METHOD.
NUREG/CR-2331 V04 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April 1-June 30,1984.
NUREG/CR-2331 V04 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July 1 -September 30,1984.
NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.
NUREG/CR-2331 V05 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,January 1-March 31,1985.
NUREG/CR-2482 V06: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program,October 1983 - March 1984.
NUREG/CR-2482 V07: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program April 1984-September 1984.
NUREG/CR-2482 V08: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period October 1984 - March 1985.
NUREG/CR-2482 V09: REVIEW OF DOE WASTE PACKAGE PROGRAM.Semiannual Report Covering The Period April 1985-September 1985.
NUREG/CR-2815 V01 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 1-7 And Appendices.
NUREG/CR-2815 V02 R1: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE.Sections 8-12.
NUREG/CR-3026: FEASIBILITY STUDY ON THE ACQUISTION OF LICENSEE EVENT DATA.
NUREG/CR-3091 V04: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1983 - March 1984.
NUREG/CR-3091 V05: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period April 1984 - September 1984.
NUREG/CR-3091 V06: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1984 - March 1985.
NUREG/CR-3444 V02: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.
NUREG/CR-3469 V02: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: Annotated Bibliography Of Selected Readings In Radiation Protection And ALARA.
NUREG/CR-3485: PRA REVIEW MANUAL.
NUREG/CR-3498: TWO-DIMENSIONAL MODELING OF INTRA-SUBASSEMBLY HEAT TRANSFER AND BUOYANCY-INDUCED FLOW REDISTRIBUTION IN LMFBRS.
NUREG/CR-3519: HUMAN ERROR PROBABILITY ESTIMATION USING LICENSEE EVENT REPORTS.
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-3829: AN EVALUATION OF THE STABILITY TESTS RECOMMENDED IN THE BRANCH TECHNICAL POSITION ON WASTE FORMS AND CONTAINER MATERIALS.
NUREG/CR-3837: MULTIPLE-SEQUENTIAL FAILURE MODEL.Evaluation Of And Procedures For Human Error Dependency.
NUREG/CR-3865: EVALUATION OF THE RADIOACTIVE INVENTORY IN,AND ESTIMATION OF ISOTOPIC RELEASE FROM,THE WASTE IN EIGHT TRENCHES AT THE SHEFFIELD LOW-LEVEL WASTE BURIAL SITE.
NUREG/CR-3876: PROBABILITY BASED LOAD COMBINATION CRITERIA FOR DESIGN OF CONCRETE CONTAINMENT STRUCTURES.
NUREG/CR-3914: PUMP AND VALVE QUALIFICATION REVIEW GUIDE.
NUREG/CR-3943: THE BWR PLAN ANALYZER.
NUREG/CR-4016 V01: APPLICATION OF SLIM-MAUD:A TEST OF AN INTERACTIVE COMPUTER-BASED METHOD FOR ORGANIZING EXPERT ASSESSMENT OF HUMAN PERFORMANCE AND RELIABILITY.Volume I:Main Report.
NUREG/CR-4050: A REVIEW OF THE SHOREHAM NUCLEAR POWER STATION PROBABILISTIC RISK ASSESSMENT.Internal Events And Core Damage Frequency.
NUREG/CR-4062: EXTENDED STORAGE OF LOW-LEVEL RADIOACTIVE WASTES.Potential Problem Areas.

NUREG/CR-4067: SUMMARY OF BARRIER DEGRADATION EVENTS AND SMALL ACCIDENTS IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.
NUREG/CR-4068: SUMMARY OF HISTORICAL EXPERIENCE WITH RELEASES OF RADIOACTIVE MATERIALS FROM COMMERCIAL NUCLEAR POWER PLANTS IN THE UNITED STATES.
NUREG/CR-4069: ANALYSES OF SOILS FROM AN AREA ADJACENT TO THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE AT SHEFFIELD,ILLINOIS.
NUREG/CR-4083: ANALYSES OF SOILS FROM THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITES AT BARNWELL,SC AND RICHLAND,WA.
NUREG/CR-4093: SAFETY/SAFEGUARDS INTERACTIONS DURING SAFETY-RELATED EMERGENCIES AT NUCLEAR POWER REACTOR FACILITIES.
NUREG/CR-4103: USES OF HUMAN RELIABILITY ANALYSIS PROBABILISTIC RISK ASSESSMENT RESULTS TO RESOLVE PERSONNEL PERFORMANCE ISSUES THAT COULD AFFECT SAFETY.
NUREG/CR-4143: REVIEW AND EVALUATION OF THE MILLSTONE UNIT 3 PROBABILISTIC SAFETY STUDY.Containment Failure Modes,Radiological Source-Terms And Offsite Consequences.
NUREG/CR-4149: ULTIMATE PRESSURE CAPACITY OF REINFORCED AND PRESTRESSED CONCRETE CONTAINMENT.
NUREG/CR-4152: AN INDEPENDENT SAFETY ORGANIZATION.
NUREG/CR-4156: OPERATING EXPERIENCE AND AGING-SEISMIC ASSESSMENT OF ELECTRIC MOTORS.
NUREG/CR-4182: VERIFICATION OF SOIL STRUCTURE INTERACTION METHODS.
NUREG/CR-4200: BIODEGRADATION TESTING OF SOLIDIFIED LOW-LEVEL WASTE STREAMS.
NUREG/CR-4201: THERMAL STABILITY TESTING OF LOW-LEVEL WASTE FORMS.
NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA.
NUREG/CR-4221: AN EVALUATION OF STRESS CORROSION CRACK GROWTH IN BWR PIPING SYSTEMS.
NUREG/CR-4228: REVIEW OF THE VOGTLE UNITS 1 AND 2 AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS.
NUREG/CR-4229: EVALUATION OF CURRENT METHODOLOGY EMPLOYED IN PROBABILISTIC RISK ASSESSMENT (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.
NUREG/CR-4230: PROBABILITY-BASED EVALUATION OF SELECTED FIRE PROTECTION FEATURES IN NUCLEAR POWER PLANTS.
NUREG/CR-4231: EVALUATION OF AVAILABLE DATA FOR PROBABILISTIC RISK ASSESSMENTS (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.
NUREG/CR-4252: INDEPENDENT ASSESSMENT OF TRAC-PD2/MOD1 CODE WITH BCL ECC BYPASS TESTS.
NUREG/CR-4253: REVIEW OF TRAC CALCULATIONS FOR CALVERT CLIFFS PTS STUDY.
NUREG/CR-4254: OCCUPATIONAL DOSE REDUCTION AND ALARA AT NUCLEAR POWER PLANTS.Study On High-Dose Jobs,Radwaste Handling,And ALARA Incentives.
NUREG/CR-4291: CONCLUSION AND SUMMARY REPORT ON PHYSICAL BENCHMARKING OF PIPING SYSTEMS.
NUREG/CR-4292: A COMPARATIVE ANALYSIS OF CONSTITUTIVE RELATIONS IN TRAC-PFL AND RELAP5/MOD1.
NUREG/CR-4329: RELIABILITY EVALUATION OF CONTAINMENTS INCLUDING SOIL-STRUCTURE INTERACTION.
NUREG/CR-4373: COMPENDIUM OF COST-EFFECTIVENESS EVALUATIONS OF MODIFICATIONS FOR DOSE REDUCTION AT NUCLEAR POWER PLANTS.
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP).A Methodology For Estimating Risk Impacts Of Safeguards Measures.
NUREG/CR-4400: THE IMPACT OF MECHANICAL-AND MAINTENANCE-INDUCED FAILURES OF MAIN REACTOR COOLANT PUMP SEALS ON PLANT SAFETY.
NUREG/CR-4406: AN ANALYSIS OF LOW-LEVEL WASTES.Review of Hazardous Waste Regulations And Identification of Radioactive Mixed Wastes. Final Report.
NUREG/CR-4440: A REVIEW OF EMERGENCY DIESEL GENERATOR PERFORMANCE AT NUCLEAR POWER PLANTS.

CALIFORNIA, STATE OF
NUREG/CR-4190: CALIFORNIA OFFSHORE SURVEY OF LICENSEES USING RADIOACTIVE MATERIAL.

CANADA, GOVT. OF
NUREG/CR-4317 V01: CANADIAN SEISMIC AGREEMENT.Technical Report Covering 1979-1985.

COLORADO STATE UNIV., FT. COLLINS, CO
NUREG/CR-4072: THE ESTIMATION OF ATMOSPHERIC DISPERSION AT NUCLEAR POWER PLANTS UTILIZING REAL TIME ANEMOMETER STATISTICS.

COLUMBIA UNIV., NEW YORK, NY
NUREG/CR-4094: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN SULFATE LAKE ENVIRONMENTS.
NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS.

COMMERCE, DEPT. OF, NATIONAL BUREAU OF STANDARDS
NUREG/CR-4266: STANDARD BETA-PARTICLE AND MONOENERGETIC ELECTRON SOURCES FOR THE CALIBRATION OF BETA-RADIATION PROTECTION INSTRUMENTATION.

COMMERCE, DEPT. OF, NATL. OCEANOGRAPHIC & ATMOSPHERIC ADMINISTRATION
NUREG/CR-3488 V03: IDAHO FIELD EXPERIMENT 1981.Volume 3:Comparison Of Trajectories,Concentration Patterns And MESODIF Model Calculations.
NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE.

CONFERENCE OF RADIATION CONTROL PROGRAM DIRECTORS, INC.
NUREG/CR-4352: SUGGESTED STATE REQUIREMENTS AND CRITERIA FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE REGULATORY PROGRAM.

CONTROL DATA CORP.
NUREG/CR-3741 V02: EVALUATION OF POWER REACTOR FUEL ROD ANALYSIS CAPABILITIES. Phase 2 Topical Report.Volume 2:Code Evaluation.

CREARE, INC.
NUREG/CR-3426 V01: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY. Facility And Test Design Report.
NUREG/CR-3426 V02: THERMAL AND FLUID MIXING IN 1/2-SCALE TEST FACILITY.Data Report.

DAVID W. TAYLOR NAVAL RESEARCH & DEVELOPMENT CENTER
NUREG/CR-4283: STUDY OF THE EFFECTS OF ELASTIC UNLOADINGS ON THE JI-R CURVES FROM COMPACT SPECIMENS.

DELAWARE, UNIV. OF, LEWES, DE
NUREG/CR-4233: DISTRIBUTION OF CORBICULA FLUMINEA AT NUCLEAR FACILITIES.

E.C. RODABAUGH ASSOCIATES, INC.
NUREG/CR-4305: COMMENTS ON THE LEAK-BEFORE-BREAK CONCEPT FOR NUCLEAR POWER PLANT PIPING SYSTEMS.

EARTH TECHNOLOGY CO.
NUREG/CR-4145: EARTHQUAKE RECURRENCE INTERVALS AT NUCLEAR POWER PLANTS: ANALYSIS AND RANKING.

EG&G IDAHO, INC. (SUBS. OF EG&G, INC.)
NUREG/CR-3301: CATALOG OF PRA DOMINANT ACCIDENT SEQUENCE INFORMATION.
NUREG/CR-3633 V01 S1: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.
NUREG/CR-3633 V04: TRAC-BD1/MOD1:AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS.Volume 4: Developmental Assessment.
NUREG/CR-3819: SURVEY OF AGED POWER PLANT FACILITIES.
NUREG/CR-3862: DEVELOPMENT OF TRANSIENT INITIATING EVENT FREQUENCIES FOR USE IN PROBABILISTIC RISK ASSESSMENTS.
NUREG/CR-3935: THERMAL-HYDRAULIC ANALYSES OF OVERCOOLING SEQUENCES FOR THE H.B. ROBINSON UNIT 2 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-3948: EXPERIMENTAL RESULTS OF THE OPERATIONAL TRANSIENT (OPTRAN) TESTS 1-1 AND 1-2 IN THE POWER BURST FACILITY.
NUREG/CR-4040: OPERATIONAL DECISIONMAKING AND ACTION SELECTION UNDER PSYCHOLOGICAL STRESS IN NUCLEAR POWER PLANTS.
NUREG/CR-4080: DETERMINATION OF THE AVAILABILITY OF CORE EXIT THERMOCOUPLES DURING SEVERE ACCIDENT SITUATIONS.
NUREG/CR-4141: CONTAINMENT PURGE AND VENT VALVE TEST PROGRAM FINAL REPORT.
NUREG/CR-4150: EPICOR-II RESIN DEGRADATION RESULTS FROM FIRST RESIN SAMPLES OF PF-8 AND PF-20.
NUREG/CR-4164: DATA REPORT FOR THE TPFL TEE/CRITICAL FLOW EXPERIMENTS.
NUREG/CR-4178 DRFT: AN EVALUATION OF SELECTED LICENSEE EVENT REPORTS PREPARED PURSUANT TO 10 CFR 50.73.Draft Report.

NUREG/CR-4181: LEACHABILITY OF RADIONUCLIDES FROM CEMENT SOLIDIFIED WASTE FORMS PRODUCED AT OPERATING NUCLEAR POWER REACTORS.
NUREG/CR-4195: OVERVIEW OF TRAC-PD2 ASSESSMENT CALCULATIONS.
NUREG/CR-4196: OVERVIEW OF TRAC-BD1 (VERSION 12) ASSESSMENT STUDIES.
NUREG/CR-4203: A CALCULATIONAL METHOD FOR DETERMINING BIOLOGICAL DOSE RATES FROM IRRADIATED RESEARCH REACTOR FUEL.
NUREG/CR-4212: IN-PLACE THERMAL ANNEALING OF NUCLEAR REACTOR PRESSURE VESSELS.
NUREG/CR-4227: HUMAN ENGINEERING GUIDELINES FOR THE EVALUATION AND ASSESSMENT OF VIDEO DISPLAY UNITS.
NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION.
NUREG/CR-4262 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR.Main Report.
NUREG/CR-4262 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR. Appendices.
NUREG/CR-4272: RESPONSE TREE EVALUATION:EXPERIMENTAL ASSESSMENT OF AN EXPERT SYSTEM FOR NUCLEAR REACTOR OPERATORS.
NUREG/CR-4326 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Main Report.
NUREG/CR-4326 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A 3-LOOP WESTINGHOUSE PRESSURIZED WATER REACTOR.Appendices.
NUREG/CR-4345: INVESTIGATION OF THE STABILITY OF LWR SPENT FUEL RODS BELOW 250 C.
NUREG/CR-4351: SUMMARY REPORT FOR LOFT ANTICIPATED TRANSIENT EXPERIMENT SERIES L6-8.
NUREG/CR-4375: THEORY,DESIGN,AND OPERATION OF LIQUID METAL FAST BREEDER REACTORS,INCLUDING OPERATIONAL HEALTH PHYSICS.
NUREG/CR-4397: IN-PLANT SOURCE TERM MEASUREMENTS AT PRAIRIE ISLAND NUCLEAR GENERATING STATION.
NUREG/CR-4428: OVERVIEW OF TRAC-BD1/MOD1 ASSESSMENT STUDIES.

EG&G, INC.
NUREG/CR-2531 R03: INTRODUCTORY USER'S MANUAL FOR THE U.S. NUCLEAR REGULATORY COMMISSION REACTOR SAFETY RESEARCH DATA BANK.
NUREG/CR-3005: SUMMARY OF THE NUCLEAR REGULATORY COMMISSION'S LOFT PROGRAM RESEARCH FINDINGS.
NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT.
NUREG/CR-3237: CONTROL OF EXPLOSIVE MIXTURES IN PWR WASTE GAS SYSTEMS.
NUREG/CR-3767: INTERACTIVE SIMULATOR EVALUATION FOR CRT-GENERATED DISPLAYS.
NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR.
NUREG/CR-4033: THE ROLE OF PERSONAL AIR SAMPLING IN RADIATION SAFETY PROGRAMS AND RESULTS OF A LABORATORY EVALUATION OF PERSONAL AIR-SAMPLING EQUIPMENT.
NUREG/CR-4041: SYSTEM ANALYSIS HANDBOOK.
NUREG/CR-4041 R01: SYSTEM ANALYSIS HANDBOOK.
NUREG/CR-4046: DETERMINING CRITICAL FLOW VALVE CHARACTERISTICS USING EXTRAPOLATION TECHNIQUES.
NUREG/CR-4056: PARTICULATE AND GAS RELEASE FROM LIGHT-WATER-REACTOR (LWR) FUEL RODS STORED IN INERT AND DRY AIR ATMOSPHERES.
NUREG/CR-4071: EXPLORATORY TREND AND PATTERN ANALYSIS FOR 1981 LICENSEE EVENT REPORT DATA.
NUREG/CR-4073: RESULTS OF THE SEMISCALE MOD-2B STEAM GENERATOR TUBE RUPTURE TEST SERIES.
NUREG/CR-4074: THE PERFORMANCE OF DEFECTED SPENT LWR FUEL RODS IN INERT GAS AND DRY AIR STORAGE ATMOSPHERES.
NUREG/CR-4084: DRY SPENT FUEL STORAGE TEST PLAN FOR DESTRUCTIVE FUEL ROD EXAMINATIONS.
NUREG/CR-4115: INTERNATIONAL STANDARD PROBLEM 13 (LOFT EXPERIMENT L2-5). Final Comparison Report.

ENERGY, INC.
NUREG/CR-4350 V01: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 1 - PRA Fundamentals.
NUREG/CR-4350 V04: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 4 - System Reliability And Analysis Techniques,Sessions B/C - Event Trees/Fault Trees.

ERTEC WESTERN, INC.
NUREG/CR-2663 V01: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Main Report.
NUREG/CR-2663 V02: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Appendices.

FACTORY MUTUAL RESEARCH CORP.
NUREG/CR-4231: EVALUATION OF AVAILABLE DATA FOR PROBABILISTIC RISK ASSESSMENTS (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS.

FEDERAL EMERGENCY MANAGEMENT AGENCY
NUREG-0981 R01: NRC/FEMA OPERATIONAL RESPONSE PROCEDURES FOR RESPONSE TO A COMMERCIAL NUCLEAR REACTOR ACCIDENT.

FINLAND, GOVT. OF
NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS.

GA TECHNOLOGIES, INC./GENERAL ATOMIC CO.
NUREG/CR-4303: HIGH-LEVEL WASTE PRECLOSURE SYSTEMS SAFETY ANALYSIS.Phase 1. Final Report.

GEEB'S, INC.
NUREG/CR-4271: RECOMMENDED SAFETY,RELIABILITY,QUALITY ASSURANCE AND MANAGEMENT AEROSPACE TECHNIQUES WITH POSSIBLE APPLICATION BY THE DOE TO THE HIGH-LEVEL RADIOACTIVE WASTE REPOSITORY PROGRAM.

GENERAL ELECTRIC CO.
NUREG/CR-4127 V01: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 1:Numerical Methods.
NUREG/CR-4127 V02: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 2:Models.
NUREG/CR-4127 V03: BWR FULL INTEGRAL SIMULATION TEST (FIST) PROGRAM TRAC-BWR MODEL DEVELOPMENT.Volume 3:Development Assessment For Plant Application.
NUREG/CR-4128: BWR FULL INTEGRAL SIMULATION TEST (FIST) PHASE II TEST RESULTS AND TRAC-BWR MODEL QUALIFICATION.

GENERAL PHYSICS CORP.
NUREG/CR-3688 V01: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 1:Main Report.
NUREG/CR-3688 V02: GENERATING HUMAN RELIABILITY ESTIMATES USING EXPERT JUDGMENT.Volume 2:Appendices.
NUREG/CR-3887: HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS.
NUREG/CR-4009: HUMAN RELIABILITY DATA BANK:Evaluation Results.
NUREG/CR-4010: SPECIFICATION OF A HUMAN RELIABILITY DATA BANK FOR CONDUCTING HRA SEGMENTS OF PRAS FOR NUCLEAR POWER PLANTS.
NUREG/CR-4258: AN APPROACH TO TEAM SKILLS TRAINING OF NUCLEAR POWER PLANT CONTROL ROOM CREWS.
NUREG/CR-4280: THE EFFECTS OF SUPERVISOR EXPERIENCE AND ASSISTANCE OF A SHIFT TECHNICAL ADVISOR (STA) ON CREW PERFORMANCE IN CONTROL ROOM SIMULATORS.

GEORGIA INSTITUTE OF TECHNOLOGY, ATLANTA, GA
NUREG/CR-3981: BIOACCUMULATION OF P-32 IN BLUEGILL AND CATFISH.

HANFORD ENGINEERING DEVELOPMENT LABORATORY
NUREG/CR-3319: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM:LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium.
NUREG/CR-3746 V02: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.Semiannual Progress Report,April 1984 - September 1984.
NUREG/CR-3746 V03: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.1984 Annual Report,October 1,1983 - September 30, 1984.

HARVARD UNIV., CAMBRIDGE, MA
NUREG/CR-4214: HEALTH EFFECTS MODEL FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Part I:Introduction,Integration & Summary.Part II:Scientific Basis For Health Effects Models.

HOUSTON, UNIV. OF, HOUSTON, TX

NUREG/CR-4424: DROPLET SIZES,DYNAMICS AND DEPOSITION IN VERTICAL ANNULAR FLOW.

IDAHO NATIONAL ENGINEERING LABORATORY

NUREG/CR-4429: TRAC-BD1/MOD1 USERS GUIDELINE.

ILLINOIS, STATE OF

NUREG/CR-4333: STE. GENEVIEVE FAULT ZONE,MISSOURI AND ILLINOIS.

IMPELL CORP.

NUREG/CR-4400: THE IMPACT OF MECHANICAL-AND MAINTENANCE-INDUCED FAILURES OF MAIN REACTOR COOLANT PUMP SEALS ON PLANT SAFETY.

INDIANA GEOLOGICAL SURVEY

NUREG/CR-4117: FAULTING AND JOINTING IN AND NEAR SURFACE MINES OF SOUTHWESTERN INDIANA.

INHALATION TOXICOLOGY RESEARCH INSTITUTE

NUREG/CR-3984: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS.Annual Progress Report,April 1983 - March 1984.

INSTITUTE OF SYSTEMS SCIENCES

NUREG/CR-4350 V04: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 4 - System Reliability And Analysis Techniques,Session B/C - Event Trees/Fault Trees.

INTERIOR, DEPT. OF, GEOLOGICAL SURVEY

NUREG/CR-4339: A REVIEW OF RECENT RESEARCH ON THE SEISMOTECTONICS OF THE SOUTHEASTERN SEABOARD AND AN EVALUATION OF HYPOTHESES ON THE SOURCE OF THE 1886 CHARLESTON,SOUTH CAROLINA EARTHQUAKE.

IOWA STATE UNIV., AMES, IA

NUREG/CR-4198: FRACTURE IN GLASS/HIGH LEVEL WASTE CANISTERS.

JBF ASSOCIATES

NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3922 V01: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Main Report And Appendices A And B.
NUREG/CR-3922 V02: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Appendices C And D.

LAWRENCE BERKELEY LABORATORY

NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt.
NUREG/CR-4355 V01: 238 PU(IV) IN MONKEYS.Overview Of Metabolism.

LAWRENCE LIVERMORE NATIONAL LABORATORY

NUREG/CR-3019: RECOMMENDED WELDED CRITERIA FOR USE IN THE FABRICATION OF SHIPPING CONTAINERS FOR RADIOACTIVE MATERIALS.
NUREG/CR-3558: HANDBOOK OF NUCLEAR POWER PLANT SEISMIC FRAGILITIES. Seismic Safety Margins Research Program.
NUREG/CR-3660 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 1:Summary Report.
NUREG/CR-3660 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANTS.Volume 3:Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3660 V04: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 4:Pipe Failure Induced By Crack Growth In West Coast Plants.
NUREG/CR-3663 V01: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 1:Summary Report.
NUREG/CR-3663 V03: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Volume 3:Double Ended Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-3854: FABRICATION CRITERIA FOR SHIPPING CONTAINERS.
NUREG/CR-4035: A HIGHWAY ACCIDENT INVOLVING RADIOPHARMACEUTICALS NEAR BROOKHAVEN,MISSISSIPPI ON DECEMBER 3,1983.
NUREG/CR-4123: SEISMIC FRAGILITY OF REINFORCED CONCRETE STRUCTURES AND COMPONENTS FOR APPLICATION TO NUCLEAR FACILITIES.
NUREG/CR-4239: ANALYSIS OF THE ABILITY OF CURRENT HEALTH PHYSICS INSTRUMENTS TO PREDICT DOSE IN EXPOSED INDIVIDUALS.
NUREG/CR-4263: RELIABILITY ANALYSIS OF STIFF VERSUS FLEXIBLE PIPING FINAL PROJECT REPORT.
NUREG/CR-4290 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF BABCOCK AND WILCOX PWR PLANTS.Volume 2:Guillotine Break Indirectly Induced By Earthquakes.
NUREG/CR-4331: SIMPLIFIED SEISMIC PROBABILISTIC RISK ASSESSMENT.Procedures And Limitations.
NUREG/CR-4334: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS.

LOS ALAMOS SCIENTIFIC LABORATORY

NUREG/CR-3208: TRAC-PD2 DEVELOPMENTAL ASSESSMENT.
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.
NUREG/CR-3706: TRAC ANALYSES OF SEVERE OVERCOOLING TRANSIENTS FOR THE OCONEE-1 PWR.
NUREG/CR-3866: TRAC-PD2 INDEPENDENT ASSESSMENT.
NUREG/CR-3953: THE USE OF MAG-1 SPECTACLES WITH POSITIVE- AND NEGATIVE-PRESSURE RESPIRATORS.
NUREG/CR-4020: HMS:A COMPUTER PROGRAM FOR TRANSIENT,THREE-DIMENSIONAL MIXING GASES.
NUREG/CR-4045: LITERATURE REVIEW ON AEROSOL-SAMPLING DEVICES FOR RESPIRATORY FIELD STUDIES.
NUREG/CR-4079: ANALYTIC STUDIES PERTAINING TO STEAM GENERATOR TUBE RUPTURE ACCIDENTS.
NUREG/CR-4107: SEQUENTIAL TEST PROCEDURES FOR DETECTING PROTRACTED MATERIALS LOSSES.
NUREG/CR-4109: TRAC-PF1 ANALYSES OF POTENTIAL PRESSURIZED-THERMAL SHOCK TRANSIENTS AT CALVERT CLIFFS/UNIT 1,A Combustion Engineering PWR.
NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET-GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE,AND SODIUM CHLORIDE.
NUREG/CR-4140: DOMINANT ACCIDENT SEQUENCES IN OCONEE-1 PRESSURIZED WATER REACTOR.
NUREG/CR-4217: A STATISTICAL ANALYSIS OF NUCLEAR POWER PLANT VALVE FAILURE-RATE VARIABILITY--SOME PRELIMINARY RESULTS.
NUREG/CR-4225: SUMMARY OF EFFICIENCY TESTING OF STANDARD AND HIGH-CAPACITY HIGH-EFFICIENCY PARTICULATE AIR FILTERS SUBJECTED TO SIMULATED TORNADO DEPRESSURIZATION AND EXPLOSIVE SHOCK WAVES.
NUREG/CR-4232: THE RESPONSE OF VENTILATION DAMPERS TO LARGE AIRFLOW PULSES.
NUREG/CR-4260: TORAC USER'S MANUAL.A Computer Code For Analyzing Tornado-Induced Flow And Material Transport In Nuclear Facilities.
NUREG/CR-4264: INVESTIGATION ON HIGH-EFFICIENCY PARTICULATE AIR FILTER PLUGGING BY COMBUSTION AEROSOLS.
NUREG/CR-4274: ANALYSIS AND TESTS ON SMALL-SCALE SHEAR WALLS FY-82 FINAL REPORT.
NUREG/CR-4278: TRAC-PF1/MOD1 DEVELOPMENT ASSESSMENT.
NUREG/CR-4314: BRIEF SURVEY AND COMPARISON OF COMMON CAUSE FAILURE ANALYSIS.
NUREG/CR-4321: FULL SCALE MEASUREMENTS OF SMOKE TRANSPORT AND DEPOSITION IN VENTILATION SYSTEM DUCTWORK.
NUREG/CR-4396: SIMMER POSTPROCESSOR MANUAL.

LOUISIANA, STATE OF

NUREG/CR-3516: A SURVEY OF THE USES OF RADIOACTIVE MATERIALS IN LOUISIANA'S OFFSHORE WATERS.

MASSACHUSETTS INSTITUTE OF TECHNOLOGY, CAMBRIDGE, MA

NUREG/CR-4376: HEAT TRANSFER,CARRYOVER AND FALL BACK IN PWR STEAM GENERATORS DURING TRANSIENTS.

MATERIALS ENGINEERING ASSOCIATES, INC.

NUREG/CR-3228 V03: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1984.
NUREG/CR-3945: FATIGUE CRACK GROWTH RATES OF LOW-CARBON AND STAINLESS PIPING STEELS IN PWR ENVIRONMENT.
NUREG/CR-4121: EFFECTS OF SULFUR CHEMISTRY AND FLOW RATE ON FATIGUE CRACK GROWTH RATES IN LWR ENVIRONMENTS.
NUREG/CR-4395: CORRELATION OF CV AND KIC/KJC TRANSITION TEMPERATURE INCREASES DUE TO IRRADIATION.

NUREG/CR-4422: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN PWR ENVIRONMENTS.
NUREG/CR-4437: EXPLORATORY STUDIES OF ELEMENT INTERACTIONS AND COMPOSITION DEPENDENCIES IN RADIATION SENSITIVITY DEVELOPMENT.

MATHTECH, INC.
NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY CONTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

MEMPHIS STATE UNIV., MEMPHIS, TN
NUREG/CR-4288: FOCAL MECHANISM ANALYSES FOR VIRGINIA AND EASTERN TENNESSEE EARTHQUAKES (1978-1984).

MICHIGAN, UNIV. OF, ANN ARBOR, MI
NUREG/CR-3145 V03: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION - ANNUAL REPORT.(October 1982 - September 1983, Volume 3).
NUREG/CR-4365: DESIGN AND DEVELOPMENT OF A SPECIAL PURPOSE SAFT SYSTEM FOR NONDESTRUCTIVE EVALUATION OF NUCLEAR REACTOR VESSELS AND PIPING COMPONENTS.

NAVY, DEPT. OF, NAVAL RESEARCH LAB.
NUREG/CR-3990: CHARCOAL PERFORMANCE UNDER ACCIDENT CONDITIONS IN LIGHT-WATER REACTORS.

NEW MEXICO STATE UNIV., LAS CRUCES, NM
NUREG/CR-4264: INVESTIGATION ON HIGH-EFFICIENCY PARTICULATE AIR FILTER PLUGGING BY COMBUSTION AEROSOLS.

NEW YORK, STATE UNIV. OF, ALBANY, NY
NUREG/CR-3178: STRUCTURAL AND TECTONIC STUDIES IN NEW YORK STATE.Final Report,July 1981 - June 1982.

NORTHWESTERN UNIV., EVANSTON, IL
NUREG/CR-4414: DIRECT-CONTACT CONDENSATION OF STEAM ON COLD WATER IN STRATIFIED COUNTERCURRENT FLOW.
NUREG/CR-4415: COUNTER-CURRENT STEAM/WATER FLOW ABOVE A PERFORATED PLATE-VERTICAL INJECTION OF WATER.
NUREG/CR-4416: STABILITY OF STEAM-WATER COUNTERCURRENT STRATIFIED FLOW.
NUREG/CR-4417: LOCAL PROPERTIES OF COUNTERCURRENT STRATIFIED STEAM-WATER FLOW.

OAK RIDGE NATIONAL LABORATORY
NUREG/CP-0062: PROCEEDINGS OF THE CONFERENCE ON THE APPLICATION OF GEOCHEMICAL MODELS TO HIGH-LEVEL NUCLEAR WASTE REPOSITORY ASSESSMENT.
NUREG/CR-2000 V03N12: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of December 1984.
NUREG/CR-2000 V04 N1: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of January 1985.
NUREG/CR-2000 V04 N2: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of February 1985.
NUREG/CR-2000 V04 N3: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of March 1985.
NUREG/CR-2000 V04 N4: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of April 1985.
NUREG/CR-2000 V04 N5: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of May 1985.
NUREG/CR-2000 V04 N6: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of June 1985.
NUREG/CR-2000 V04 N7: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of July 1985.
NUREG/CR-2000 V04 N8: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of August 1985.
NUREG/CR-2000 V04 N9: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of September 1985.
NUREG/CR-2000 V04N10: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of October 1985.
NUREG/CR-2000 V04N11: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of November 1985.
NUREG/CR-3430 V02: NUCLEAR POWER PLANT OPERATING EXPERIENCE - 1982 Annual Report.
NUREG/CR-3442: RADTWO:A COMPUTER CODE FOR SIMULATING FAST-TRANSIENT, TWO-DIMENSIONAL,TWO-LAYER RADIONUCLIDE CONCENTRATION CONDITIONS IN LAKES,RESERVOIRS,RIVERS,ESTUARIES,AND COASTAL REGIONS.
NUREG/CR-3481 V02: NUCLEAR POWER PLANT PERSONNEL QUALIFICATIONS AND TRAINING: TAPS -- The Task Analysis Profiling System.
NUREG/CR-3514 V02: THE CHEMICAL BEHAVIOR OF IODINE IN AQUEOUS SOLUTIONS UP TO 150 C.II.Radiation-Redox Conditions.
NUREG/CR-3551: SAFETY IMPLICATIONS ASSOCIATED WITH IN-PLANT PRESSURIZED GAS STORAGE AND DISTRIBUTION SYSTEMS IN NUCLEAR POWER PLANTS.
NUREG/CR-3626 V02: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: DESCRIPTION OF MODEL CONTENT,STRUCTURE,AND SENSITIVITY TESTING.
NUREG/CR-3634: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: USER'S MANUAL.
NUREG/CR-3651: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES.
NUREG/CR-3723: STRESS-INTENSITY-FACTOR INFLUENCE COEFFICIENTS FOR SURFACE FLAWS IN PRESSURE VESSELS.
NUREG/CR-3738: ENVIRONMENTAL EFFECTS OF THE URANIUM FUEL CYCLE.A Review Of Data For Technetium.
NUREG/CR-3744 V02: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR APRIL-SEPTEMBER 1984.
NUREG/CR-3764: BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE.
NUREG/CR-3830 V02: AEROSOL RELEASE AND TRANSPORT PROGRAM.Semiannual Progress Report For April 1984-September 1984.
NUREG/CR-3831: THE IN-PLANT RELIABILITY DATA BASE FOR NUCLEAR PLANT COMPONENTS.Interim Report - Diesel Generators,Batteries, Chargers And Inverters.
NUREG/CR-3851 V03: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE RESPOSITORY SITE PROJECTS.Report For April-June 1984.
NUREG/CR-3851 V04: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Annual Progress Report For October 1983-September 1984.
NUREG/CR-3872: DATA ACCUISITION AND CONTROL OF THE HSST SERIES V IRRADIATION EXPERIMENT AT THE ORR.
NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1-September 30,1984.
NUREG/CR-3885 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, October 1-December 31,1984.
NUREG/CR-3887: HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS.
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.
NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3922 V01: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Main Report And Appendices A And B.
NUREG/CR-3922 V02: SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES.Appendices C And D.
NUREG/CR-3930: OBSERVED BEHAVIOR OF CESIUM,IODINE,AND TELLURIUM IN THE ORNL FISSION PRODUCT RELEASE PROGRAM.
NUREG/CR-3949 V01: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM: Semiannual Progress Report For Period Ending June 30,1984.
NUREG/CR-3949 V02: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM: Annual Progress Report For Period Ending December 31,1984.
NUREG/CR-3978: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL PLATE AND WELDS FOR THE FOURTH HSST IRRADIATION SERIES.
NUREG/CR-3991: FAILURE MODES AND EFFECTS ANALYSIS (FMEA) OF THE ICS/NNI ELECTRIC POWER DISTRIBUTION CIRCUITRY AT THE OCONEE-1 NUCLEAR PLANT.
NUREG/CR-3992: COLLECTION AND EVALUATION OF COMPLETE AND PARTIAL LOSSES OF OFF-SITE POWER AT NUCLEAR POWER PLANTS.
NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1.
NUREG/CR-4022: PRESSURIZED THERMAL SHOCK EVALUATION OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT.

NUREG/CR-4031 V02: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutronics Calculations."
NUREG/CR-4031 V03: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutron Exposure Parameters."
NUREG/CR-4037: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-5.
NUREG/CR-4038: SENSITIVITY AND UNCERTAINTY STUDIES OF THE CRAC2 COMPUTER CODE.
NUREG/CR-4039: GAMMA-RAY CHARACTERIZATION OF THE TWO-YEAR IRRADIATION EXPERIMENT PERFORMED AT THE POOLSIDE FACILITY.
NUREG/CR-4043: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-6.
NUREG/CR-4081: ABSORPTION OF GASEOUS IODINE BY WATER DROPLETS.
NUREG/CR-4086: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL WELDS FOR THE THIRD HSST IRRADIATION SERIES.
NUREG/CR-4092: ORNL CHARACTERIZATION OF HEAVY-SECTION STEEL TECHNOLOGY PROGRAM PLATES 01,02 AND 03.
NUREG/CR-4104: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL. Field Evaluation/Validation.
NUREG/CR-4105: AN ASSESSMENT OF THERMAL GRADIENT TUBE RESULTS FROM THE HI SERIES OF FISSION PRODUCT RELEASE TESTS.
NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest.
NUREG/CR-4114: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS.II.
NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.
NUREG/CR-4206: A SELECT REVIEW OF THE RECENT (1979-1983) BEHAVIORAL RESEARCH LITERATURE ON TRAINING SIMULATORS.
NUREG/CR-4219 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.
NUREG/CR-4234 V01: AGING AND SERVICE WEAR OF ELECTRIC MOTOR-OPERATED VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.
NUREG/CR-4236 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS:REPORT FOR OCTOBER-DECEMBER 1984.
NUREG/CR-4236 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for January-March 1985.
NUREG/CR-4249: PRESSURE VESSEL FRACTURE STUDIES PENETRATING TO THE PWR THERMAL-SHOCK ISSUE:EXPERIMENTS TSE-5,TSE-5A AND TSE-6.
NUREG/CR-4255 V01: AEROSAL RELEASE AND TRANSPORT PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1984 - MARCH 1985.
NUREG/CR-4256: MEASUREMENT OF RESPONSE TIME AND DETECTION OF DEGRADATION IN PRESSURE SENSOR/SENSING LINE SYSTEMS.
NUREG/CR-4257: INSPECTION,SURVEILLANCE,AND MONITORING OF ELECTRICAL EQUIPMENT INSIDE CONTAINMENT OF NUCLEAR POWER PLANTS--WITH APPLICATIONS TO ELECTRICAL CABLES.
NUREG/CR-4275: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM - FIVE-YEAR PLAN FY 1984-1988.
NUREG/CR-4280: THE EFFECTS OF SUPERVISOR EXPERIENCE AND ASSISTANCE OF A SHIFT TECHNICAL ADVISOR (STA) ON CREW PERFORMANCE IN CONTROL ROOM SIMULATORS.
NUREG/CR-4284: NEUTRON EXPOSURE PARAMETERS FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY IRRADIATION SERIES.
NUREG/CR-4304: PRESSURE VESSEL FRACTURE STUDIES PERTAINING TO THE PWR THERMAL-SHOCK ISSUE.Experiment TSE-7.
NUREG/CR-4305: COMMENTS ON THE LEAK-BEFORE-BREAK CONCEPT FOR NUCLEAR POWER PLANT PIPING SYSTEMS.
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NUREG/CR-4360 V02: CALCULATIONAL METHODS FOR ANALYSIS OF POSTULATED UF6 RELEASES.
NUREG/CR-4367: ORVIRT.PC:A 2-D FINITE-ELEMENT FRACTURE ANALYSIS PROGRAM FOR A MICROCOMPUTER.
NUREG/CR-4402 V01: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31,1985.
NUREG/CR-4403: SUMMARY OF THE WASTE MANAGEMENT PROGRAMS AT URANIUM RECOVERY FACILITIES AS THEY RELATE TO THE 40 CFR PART 192 STANDARDS.

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NUREG/CR-3816 V02: REACTOR SAFETY RESEARCH.Quarterly Report,April-June 1984.
NUREG/CR-3816 V03: REACTOR SAFETY RESEARCH.Quarterly Report,July-September 1984.
NUREG/CR-3816 V04: REACTOR SAFETY RESEARCH.Quarterly Report,October-December 1984.
NUREG/CR-3820 V03: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM.Quarterly Report,July-September 1984.
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NUREG/CR-4335: POTENTIAL BENEFITS OBTAINED BY REQUIRING SAFTEY-GRADE COLD SHUTDOWN SYSTEMS.
NUREG/CR-4340 V01: REACTOR SAFETY RESEARCH SEMIANNUAL REPORT.January - June 1985.
NUREG/CR-4342: UNCERTAINTY AND SENSITIVITY ANALYSIS OF A MODEL FOR MULTICOMPONENT AEROSOL DYNAMICS.
NUREG/CR-4350 V01: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 1 - PRA Fundamentals.
NUREG/CR-4350 V02: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 2:Probability And Statistics For PRA Applications.
NUREG/CR-4350 V03: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 3 - Systems Reliability And Analysis Techniques,Session A - Reliability.
NUREG/CR-4350 V04: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 4 - System Reliability And Analysis Techniques.Sessions B/C - Event Trees/Fault Trees.
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NUREG/CR-4350 V06: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 6 - Data Development.
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NUREG/CR-4358: APPLICATIONS OF DENSITY PROFILING TO EQUIPMENT QUALIFICATION ISSUES.
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NUREG/CR-4398: COST ANALYSIS OF REVISIONS TO 10 CFR PART 50,APPENDIX J, LEAK TESTS FOR PRIMARY AND SECONDARY C'NTAINMENTS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

SCIENCE APPLICATIONS INTERNATIONAL CORP. (FORMERLY SCIENCE APPLICATIONS,

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NUREG/CR-4110: REPOSITORY SITE DATA REPORT FOR UNSATURATED TUFF,YUCCA MOUNTAIN,NEVADA.
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NUREG/CR-4350 V05: PROBABILISTIC RISK ASSESSMENT COURSE DOCUMENTATION.Volume 5 - Systems Reliability And Analysis Techniques,Session D - Quantification.
NUREG/CR-4392: MEASURES OF SAFEGUARDS RISK EMPLOYING PRA (MOSREP):A Methodology For Estimating Risk Impacts Of Safeguards Measures.

SIGMA RESEARCH, INC.
NUREG/CR-4170: AN ULTRA-HIGH SPEED RESIDUE PROCESSOR FOR SAFT INSPECTION SYSTEM IMAGE ENHANCEMENT.

ST. LOUIS UNIV., ST. LOUIS, MO
NUREG/CR-4226: NEW MADRID SEISMOTECTONIC STUDY.Activities During Fiscal Year 1983.

STEVENSON & ASSOCIATES
NUREG-1061 V02 ADD: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2 Addendum:Summary And Evaluation Of Historical Strong-Motion Earthquake Seismic Response And Damage To Aboveground Industrial Piping.

STRUCTURAL MECHANICS ASSOCIATES
NUREG/CR-3805 V02: ENGINEERING CHARACTERIZATION OF GROUND MOTION.Task II: Effects Of Ground Motion Characteristics On Structural Response Considering Localized Structural Nonlinearities And Soil-Structure Interaction Effects.

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TOUCHE ROSS & CO.
NUREG/CR-4322 V01: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 1: Enterprise Model.
NUREG/CR-4322 V02: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 2: Data Entity Dictionary.
NUREG/CR-4322 V03: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 3: Data Model.
NUREG/CR-4322 V04: CORPORATE DATA NETWORK (CDN) DATA REQUIREMENTS TASK.Vol 4: Preliminary Strategic Data Plan.

UNDERWRITERS LABORATORY, INC.
NUREG/CR-4112 V01: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task A:IEEE Flame Test.
NUREG/CR-4112 V02: INVESTIGATION OF CABLE AND CABLE SYSTEM FIRE TEST PARAMETERS.Task B:Firestop Test Method.

UNITED KINGDOM ATOMIC ENERGY AUTHORITY
NUREG/CR-3646: TRAC-PF1 INDEPENDENT ASSESSMENT.

UNITED NUCLEAR CORP. (SUBS. OF UNC RESOURCES, INC.)
NUREG/CR-4090: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Annual Summary Report - Fiscal Year 1984.

UTAH, UNIV. OF, SALT LAKE CITY, UT
NUREG/CR-4382: CONCENTRATIONS OF URANIUM AND THORIUM ISOTOPES IN URANIUM MILLERS' AND MINERS' TISSUES.

VIRGINIA POLYTECHNIC INSTITUTE & STATE UNIV., BLACKSBURG, VA
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WESTINGHOUSE ELECTRIC CORP.
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NUREG/CR-4166: ANALYSIS OF FLECHT-SEASET 163-ROD BLOCKED BUNDLE DATA USING COBRA-TF.
NUREG/CR-4167: FLECHT SEASET PROGRAM.Final Report.NRC/EPRI Westinghouse Report Number 16.

WISCONSIN, UNIV. OF, MADISON, WI
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