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Regulatory and Technical Reports (Abstract Index Journal)

Annual Compilation for 1993

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International Agreement Report

NUREG/IA-0001: ASSESSMENT OF TRAC-PD2 USING SUPER CANNON AND HDR EXPERIMENTAL DATA. NEUMANN, U. Kraftwerk Union. August 1986. 223 pp. 8608270424. 37659:138.

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

All these report codes are controlled and assigned by the staff of the Publishing and Translations Section of the NRC Division of Publications Services.

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:

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

Secondary Report Number Index
Personal Author Index
Subject Index
NRC Originating Organization Index (Staff Reports)
NRC Originating Organization Index (International Agreements)
NRC Contract Sponsor Index (Contractor Reports)
Contractor Index
International Organization 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-0808: 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).

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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, NUREG/CR-XXXX is an NRC contractor-prepared report, and NUREG/IA-XXXX is an international agreement report. The bibliographic information (see Preface for details) is followed by a brief abstract of this report.

NUREG-0020 V17: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of December 31, 1992.(Gray Book I) HARTFIELD,R.A. Division of Computer & Telecommunications Services (Post 890205). March 1993. 335pp. 9304080036. 74493:001.

The Nuclear Regulatory Commission's annual summary of licensed nuclear power reactor data is based primarily on the report of operating data submitted by licensees for each unit for the month of December because that report contains data for the month of December, the year to date (in this case calendar year 1992) and cumulative data, usually from the date of commercial operation. The data is not independently verified, but various computer checks are made. The report is divided into two sections. The first contains summary highlights and the second contains data on each individual unit in commercial operation. Section 1 capacity and availability factors are simple arithmetic averages. Section 2 items in the cumulative column are generally as reported by the licensee and notes as to the use of weighted averages and starting dates other than commercial operation are provided.

NUREG-0040 V16 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1992.(White Book) * Division of Reactor Inspection & Licensee Performance (Post 921004). January 1993. 198pp. 9302230396. 64961:001.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from October through December 1992.

NUREG-0040 V17 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1993.(White Book) * Division of Reactor Inspection & Licensee Performance (Post 921004). May 1993. 226pp. 9306180278. 75388:204.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from January through March 1993.

NUREG-0040 V17 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,April-June 1993.(White Book) * Division of Reactor Inspection & Licensee Performance (Post 921004). August 1993. 109pp. 9309210034. 76483:049.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from April through June 1993.

NUREG-0040 V17 N03: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,July-September 1993.(White Book) * Division of Reactor Inspection & Licensee Performance (Post 921004). November 1993. 57pp. 9312220138. 77543:150.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from July through September 1993.

NUREG-0090 V15 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1992. * Office for Analysis & Evaluation of Operational Data, Director. March 1993. 32pp. 9305250129. 75004:301.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event that 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 through December 1992. There were two abnormal occurrences at nuclear power plants. Six abnormal occurrences involving medical midadministrations (all therapeutic) at NRC-licensed facilities are discussed in this report. No abnormal occurrences were reported by NRC's Agreement States. The report also contains information updating previously reported abnormal occurrences.

NUREG-0090 V16 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.January-March 1993. * Office for Analysis & Evaluation of Operational Data, Director. June 1993. 30pp. 9307220177. 75743:332.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event that 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 through March 1993. There is one abnormal occurrence at a nuclear power plant discussed in this report that involved a steam generator tube rupture at Palo Verde Unit 2, and none for fuel cycle facilities. Three abnormal occurrences involving medical misadministrations (two therapeutic and one diagnostic) at NRC-licensed facilities are also discussed in this report. No abnormal occurrences were reported by NRC's Agreement States. The report also contains information updating previously reported abnormal occurrences.

NUREG-0090 V16 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1993. * Office for Analysis & Evaluation of Operational Data, Director. September 1993. 31pp. 9311010015. 76982:001.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event that 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 through June 1993, and discusses four abnormal occurrences at NRC-licensed facilities, three involving medical brachytherapy misadministrations and one involving a research reactor that operated without a safety system. One pool irradiation facility contamination event, two medical misadministrations (one "sodium iodide" and one brachytherapy), and one industrial radiographer overexposure event that were reported by NRC Agreement States are also discussed. The report also contains information updating one previously reported abnormal occurrence and information on three other events of interest.

NUREG-0304 V17 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1992. * Division of Freedom of Information & Publications Services (Post 890205). February 1993. 138pp. 9303240099. 74344:150.

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

NUREG-0304 V18 N01: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1993,January-March. * Division of Freedom of Information & Publications Services (Post 890205). May 1993. 48pp. 9306110059. 75338:252.

See NUREG-0304,V17,N04 abstract.

NUREG-0304 V18 N02: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Second Quarter 1993,April-June. * Division of Freedom of Information & Publications Services (Post 890205). August 1993. 48pp. 9309210040. 76483:158.

See NUREG-0304,V17,N04 abstract.

NUREG-0304 V18 N03: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Third Quarter 1993,July-September. * Division of Freedom of Information & Publications Services (Post 890205). November 1993. 49pp. 9312160335. 77511:249.

See NUREG-0304,V17,N04 abstract.

NUREG-0325 R16: U.S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS.March 15, 1993. * Ofc of Personnel (Post 870413). March 1993. 66pp. 9304060302. 74467:001.

Functional organization charts for the U.S. Nuclear Regulatory Commission offices, divisions, and branches are presented.

NUREG-0383 V01 R16: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Report Of NRC Approved Packages. * Division of Industrial & Medical Nuclear Safety (Post 870729). October 1993. 491pp. 9311190081. 77265:114.

This directory contains a Report of NRC Approved Packages (Volume 1), Certificates of Compliance (Volume 2), and a Report of NRC Approved Quality Assurance Programs for Radioactive Materials Packages (Volume 3). The purpose of this directory is to make available a convenient source of information on Quality Assurance Programs and Packagings which have been approved by the U.S. Nuclear Regulatory Commission. Shipments of radioactive material utilizing these packagings must be in accordance with the provisions of 49 CFR Part 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 themselves that they have a copy of the current approval and conduct their transportation activities in accordance with an NRC approved quality assurance program.

NUREG-0383 V02 R16: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Certificates Of Compliance. * Division of Industrial & Medical Nuclear Safety (Post 870729). October 1993. 590pp. 9311190087. 77267:032.

See NUREG-0383,V01,R16 abstract.

NUREG-0383 V03 R13: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Report Of NRC Approved Quality Assurance Programs For Radioactive Materials Packages. * Division of Industrial & Medical Nuclear Safety (Post 870729). October 1993. 146pp. 9311190096. 77266:246.

See NUREG-0383,V01,R16 abstract.

NUREG-0386 D06 R05: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission,Appeal Board And Licensing Board Decisions.July 1972 - March 1992. * Office of the General Counsel (Post 860701). February 1993. 602pp. 9303110316. 74215:001.

This 5th revision of the sixth edition of the NRC 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 of July 1, 1972 to March 31, 1992, interpreting the NRC's Rules of Practice in 10 CFR Part 2.

NUREG-0386 D06 R06: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission,Appeal Board And Licensing Board Decisions.July 1972 - June 1992. * Office of the General Counsel (Post 860701). May 1993. 600pp. 9306010258. 75055:001.

This 6th revision of the sixth edition of the NRC 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 of July 1, 1972 to June 30, 1992, interpreting the NRC's Rules of Practice in 10 CFR Part 2.

NUREG-0386 D06 R07: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission, Appeal Board And Licensing Board Decisions.July 1972 - September 1992. * Office of the General Counsel (Post 860701). August 1993. 600pp. 9309090041. 76379:138.

This 7th revision of the sixth edition of the NRC 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 of July 1, 1972 to September 30, 1992, interpreting the NRC's Rules of Practice in 10 CFR Part 2.

NUREG-0430 V12: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.July 1, 1991 - June 30, 1992.(Gray Book II) JOY,D.; BROWN,C. Office of Nuclear Material Safety & Safeguards. April 1993. 20pp. 9305100071. 74857:295.

NRC is committed to the periodic publication of licensed fuel facilities inventory difference data, following agency review of the information and completion of any related 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-0525 V02 R01: SAFEGUARDS SUMMARY EVENT LIST (SSEL).January 1, 1990 Through December 31, 1992. FADDEN,M.; YARDUMIAN,J. Operations Branch. July 1993. 250pp. 9308160105. 76142:001.

The Safeguards Summary Event List provides brief summaries of hundreds of safeguards-related events involving nuclear material or facilities regulated by the U.S. Nuclear Regulatory Commission. Events are described under the categories: Bomb-related, Intrusion, Missing/Allegedly Stolen, Transportation-related, Tampering/Vandalism, Arson, Firearms-related, Radiological Sabotage, Non-radiological Sabotage, and Miscellaneous. Because of the public interest, the Miscellaneous category also includes events reported involving source material, byproduct material, and natural uranium, which are exempt from safeguards requirements. Information in the event descriptions was obtained from official NRC sources.

NUREG-0540 V14 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.November 1-30, 1992. * Division of Freedom of Information & Publications Services (Post 890205). January 1993. 333pp. 9302230138. 64980:001.

This document is a monthly publication containing descriptions of information received and generated by the U.S. Nuclear Regulatory Commission (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, Corporate Source, Report Number, and Cross Reference of Enclosures to Principal Documents.

NUREG-0540 V14 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.December 1-31, 1992. * Division of Freedom of Information & Publications Services (Post 890205). February 1993. 343pp. 9303110330. 74213:085.

See NUREG-0540,V14,N11 abstract.

NUREG-0540 V15 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.January 1-31,1993. * Division of Freedom of Information & Publications Services (Post 890205). March 1993. 320pp. 9304020324. 74448:001.

See NUREG-0540,V14,N11 abstract.

NUREG-0540 V15 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.February 1-28, 1993. * Division of Freedom of Information & Publications Services (Post 890205). April 1993. 340pp. 9304300306. 74773:005.

See NUREG-0540,V14,N11 abstract.

NUREG-0540 V15 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.March 1-31, 1993. * Division of Freedom of Information & Publications Services (Post 890205). May 1993. 400pp. 9306010264. 75057:001.

See NUREG-0540,V14,N11 abstract.

NUREG-0540 V15 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.April 1-30, 1993. * Division of Freedom of Information & Publications Services (Post 890205). June 1993. 350pp. 9306290175. 75499:053.

See NUREG-0540,V14,N11 abstract.

NUREG-0540 V15 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.May 1-31, 1993. * Division of Freedom of Information & Publications Services (Post 890205). July 1993. 350pp. 9308160099. 76115:010.

See NUREG-0540,V14,N11 abstract.

NUREG-0540 V15 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.June 1-30, 1993. * Division of Freedom of Information & Publications Services (Post 890205). August 1993. 421pp. 9309030160. 76325:158.

See NUREG-0540,V14,N11 abstract.

NUREG-0540 V15 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.July 1-31, 1993. * Division of Freedom of Information & Publications Services (Post 890205). September 1993. 350pp. 9309210229. 76485:077.

See NUREG-0540,V14,N11 abstract.

NUREG-0540 V15 N08: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.August 1-31, 1993. * Division of Freedom of Information & Publications Services (Post 890205). October 1993. 400pp. 9311080070. 77068:001.

See NUREG-0540,V14,N11 abstract.

NUREG-0540 V15 N09: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.September 1-30, 1993. * Division of Freedom of Information & Publications Services (Post 890205). November 1993. 365pp. 9312160329. 77510:001.

See NUREG-0540,V14,N11 abstract.

NUREG-0540 V15 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.October 1-31, 1993. * Division of Freedom of Information & Publications Services (Post 890205). December 1993. 274pp. 9401060244. 77687:001.

See NUREG-0540,V14,N11 abstract.

NUREG-0713 V12: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES,1990.Twenty-Third Annual Report. RADDATZ,C.T. Division of Regulatory Applications (Post 870413). HAGEMEYER,D. Science Applications International Corp. (formerly Science Applications, Inc.). January 1993. 294pp. 9302020464. 64729:203.

This report summarizes the occupational radiation exposure information that has been reported to the NRC's Radiation Exposure Information Reporting System (REIRS) by nuclear power facilities and certain other categories of NRC licensees during the years 1969 through 1990. 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 the technical specifications of nuclear power plants. Data on workers terminating their employment at certain NRC licensed facilities were obtained from reports submitted pursuant to 10 CFR 20.408. The 1990 annual reports submitted by about 443 licensees indicated that approximately 214,568 individuals were monitored, 110,204 of whom were monitored by nuclear power facilities. They incurred an average individual dose of 0.19 rem (cSv) and an average measurable dose of about 0.36 (cSv). Termination radiation exposure reports were analyzed to reveal that about 113,361 individuals completed their employment with one or more of the 443 covered licensees during 1990. Some 77,633 of these individuals terminated from power reactor facilities, and about 11,083 of them were considered to be transient workers who received an average dose of 0.67 rem (cSv).

NUREG-0713 V13: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES,1991.Twenty-Fourth Annual Report. RADDATZ,C.T. Division of Regulatory Applications (Post 870413). HAGEMEYER,D. Science Applications International Corp. (formerly Science Applications, Inc.). July 1993. 300pp. 9308160144. 76114:077.

This report summarizes the occupational radiation exposure information that has been reported to the NRC's Radiation Exposure Information Reporting System (REIRS) by nuclear power facilities and certain other categories of NRC licensees during the years 1969 through 1991. The bulk of the data presented in the report was obtained from annual radiation exposure reports submitted in accordance with the requirements of 10CFR20.407 and the technical specifications of nuclear power plants. Data on workers terminating their employment at certain NRC licensed facilities were obtained from reports submitted pursuant to 10CFR20.408. The 1991 annual reports submitted by about 436 licensees indicated that approximately 206,732 individuals were monitored, 182,334 of whom were monitored by nuclear power facilities. They incurred an average individual dose of 0.15 rem (cSv) and an average measurable dose of about 0.31 (cSv). Termination radiation exposure reports were analyzed to reveal that about 96,231 individuals completed their employment with one or more of the 436 covered licensees during 1991. Some 68,115 of these individuals terminated from power reactor facilities, and about 7,763 of them were considered to be transient workers who received an average dose of 0.52 rem (cSv).

NUREG-0713 V14: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER

FACILITIES 1992.Twenty-Fifth Annual Report. RADDATZ,C.T. Division of Regulatory Applications (Post 870413). HAGEMEYER,D. Science Applications International Corp. (formerly Science Applications, Inc.). December 1993. 300pp. 9401120298. 77769:001.

This report summarizes the occupational radiation exposure information that has been reported to the NRC's Radiation Exposure Information Reporting System (REIRS) by nuclear power facilities and certain other categories of NRC licensees during the years 1969 through 1992. 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 the technical specifications of nuclear power plants. Data on workers terminating their employment at certain NRC licensed facilities were obtained from reports submitted pursuant to 10 CFR 20.408. The 1992 annual reports submitted by about 364 licensees indicated that approximately 204,365 individuals were monitored, 183,927 of whom were monitored by nuclear power facilities. They incurred an average individual dose of 0.16 rem (cSv) and an average measurable dose of about 0.30 (cSv). Termination radiation exposure reports were analyzed to reveal that about 74,566 individuals completed their employment with one or more of the 364 covered licensees during 1992. Some 71,846 of these individuals terminated from power reactor facilities, and about 9,724 of them were considered to be transient workers who received an average dose of 0.50 rem (cSv).

NUREG-0725 R09: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL. * Division of Safeguards & Transportation (870413-930206). March 1993. 37pp. 9304190142. 74624:305.

This circular has been prepared to provide information on the shipment of irradiated reactor fuel (spent fuel) subject to regulation by the Nuclear Regulatory Commission (NRC), and to meet the requirements of Public Law 96-295. The report provides a brief description of NRC authority for certain aspects of transporting spent fuel. It provides descriptive statistics on spent fuel shipments regulated by the NRC from 1979 to 1992. It also lists detailed highway and railway segments used within each state from October 1, 1987 through December 31, 1992.

NUREG-0750 V36 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-September 1992. * Division of Freedom of Information & Publications Services (Post 890205). April 1993. 52pp. 9305250147. 75005:052.

Digests and indexes for issuances of the Commission, the Atomic Safety and Licensing Board Panel, the Administrative Law Judges, the Directors' Decisions, and the Denials of Petitions for Rulemaking are presented.

NUREG-0750 V36 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-December 1992. * Division of Freedom of Information & Publications Services (Post 890205). June 1993. 73pp. 9307060044. 75563:001.

See NUREG-0750,V36,I01 abstract.

NUREG-0750 V36 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1992.Pages 1-45. * Division of Freedom of Information & Publications Services (Post 890205). January 1993. 45pp. 9302230178. 64977:308.

Legal issuances of the Commission, the Atomic Safety and Licensing Board Panel, the Administrative Law Judges, and NRC Program Offices are presented.

NUREG-0750 V36 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1992. Pages 47-148. * Division of Freedom of Information & Publications Services (Post 890205). February 1993. 110pp. 9302230128. 64978:220.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V36 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1992. Pages 149-220. * Division of Freedom of Information & Publications Services (Post 890205). February 1993. 77pp. 9303090074. 74138:142.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V36 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1992. Pages 221-249. * Division of Freedom of Information & Publications Services (Post 890205). February 1993. 34pp. 9303120069. 74214:298.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V36 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1992. Pages 251-350. * Division of Freedom of Information & Publications Services (Post 890205). March 1993. 109pp. 9303300160. 74408:001.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V36 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1992. Pages 351-396. * Division of Freedom of Information & Publications Services (Post 890205). March 1993. 53pp. 9304060309. 74467:069.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V37 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-March 1993. * Division of Freedom of Information & Publications Services (Post 890205). July 1993. 55pp. 9308160096. 76122:200.

See NUREG-0750,V36,I01 abstract.

NUREG-0750 V37 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January-June 1993. * Division of Freedom of Information & Publications Services (Post 890205). November 1993. 77pp. 9312160325. 77505:001.

See NUREG-0750,V36,I01 abstract.

NUREG-0750 V37 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1993. Pages 1-54. * Division of Freedom of Information & Publications Services (Post 890205). March 1993. 62pp. 9304080072. 74499:001.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V37 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1993. Pages 55-134. * Division of Freedom of Information & Publications Services (Post 890205). April 1993. 85pp. 9305250140. 75004:213.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V37 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1993. Pages 135-249. * Division of Freedom of Information & Publications Services (Post 890205). May 1993. 150pp. 9306210229. 75403:016.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V37 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1993.Pages 251-354. * Division of Freedom of Information & Publications Services (Post 890205). July 1993. 110pp. 9308160091. 76119:007.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V37 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1993.Pages 355-418. * Division of Freedom of Information & Publications Services (Post 890205). August 1993. 69pp. 9308190001. 76152:038.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V37 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1993.Pages 419-515. * Division of Freedom of Information & Publications Services (Post 890205). August 1993. 103pp. 9309210012. 76481:034.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V38 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1993.Pages 1-24. * Division of Freedom of Information & Publications Services (Post 890205). September 1993. 32pp. 9310120315. 76741:114.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V38 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1993. Pages 25-79. * Division of Freedom of Information & Publications Services (Post 890205). November 1993. 62pp. 9311180066. 77231:199.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V38 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1993. Pages 81-168. * Division of Freedom of Information & Publications Services (Post 890205). November 1993. 98pp. 9312160311. 77505:100.

See NUREG-0750,V36,N01 abstract.

NUREG-0750 V38 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1993. Pages 169-186. * Division of Freedom of Information & Publications Services (Post 890205). December 1993. 24pp. 9401030180. 77642:309.

See NUREG-0750,V36,N01 abtract.

NUREG-0797 S26: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF COMANCHE PEAK STEAM ELECTRIC STATION,UNIT 2.Docket No. 50-446.(Texas Utilities Electric Company,et al.) * Division of Reactor Projects - III,IV,V (Post 901216). February 1993. 230pp. 9303110338. 74214:068.

Supplement 26 to the Safety Evaluation Report related to the operation of the Comanche Peak Steam Electric Station (CPSES), Unit 2, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission (NRC). 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 when the Safety Evaluation Report and Supplements 1, 2, 3, 4, 6, 12, 21, 22, 23, 24, and 25 to that report were published. This supplement deals primarily with Unit 2 issues; however, it also references evaluations for several licensing issues that relate to Unit 1, which have been resolved since Supplement 25 was issued.

NUREG-0797 S27: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 2.Docket No. 50-446.(Texas Utilities Electric Company,et al.) * Division of Reactor Projects - III,IV,V (Post 901216). April 1993. 49pp. 9305100006. 74859:074.

Supplement No. 27 to the Safety Evaluation Report related to the operation of the Comanche Peak Steam Electric Station, Unit 2, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Somervell County, Texas, approximately 40 miles southwest of Fort Worth, Texas. This supplement reports the status of certain issues that had not been resolved when the Safety Evaluation Report and Supplements 1, 2, 3, 4, 6, 12, 21, 22, 23, 24, 25, and 26 to that report were published. This supplement deals primarily with Unit 2 issues.

NUREG-0837 V12 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. October-December 1992. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). March 1993. 326pp. 9304060314. 74466:001.

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 facilities throughout the country for the fourth quarter 1992.

NUREG-0837 V13 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. January-March 1993. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). May 1993. 231pp. 9306180289. 75389:073.

This report provides the status and results of the NRC Thermoluminiscent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facilities throughout the country for the first quarter of 1993.

NUREG-0837 V13 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. April-June 1993. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). August 1993. 250pp. 9309210028. 76482:103.

This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facilities throughout the country for the second quarter of 1993.

NUREG-0837 V13 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. July-September 1993. STRUCKMEYER,R. Region 1 (Post 820201). November 1993. 234pp. 9312160321. 77511:009.

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 facilities throughout the country for the third quarter of 1993.

NUREG-0847 S11: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT,UNITS 1 AND 2.Docket Nos. 50-390 And 50-391.(Tennesee Valley Authority) * Division of Reactor Projects - I/II (Post 870411). April 1993. 50pp. 9305250123. 75005:001.

Supplement No. 11 to the Safety Evaluation Report for the application filed by the Tennessee Valley Authority for license to operate Watts Bar Nuclear Plant, Units 1 and 2, Docket Nos. 50-390 and 50-391, located in Rhea County, Tennessee, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation of: (1) additional information submitted by the applicant since Supplement No. 10 was issued; and (2) matters that the staff had under review when Supplement No. 10 was issued.

NUREG-0847 S12: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT,UNITS 1 AND 2.Docket Nos. 50-390 And 50-391.(Tennesee Valley Authority) TAM,P.S. Division of Reactor Projects - I/II (Post 870411). October 1993. 60pp. 9311150028. 77221:001.

Supplement No. 12 to the Safety Evaluation Report for the application filed by the Tennessee Valley Authority for license to operate Watts Bar Nuclear Plant, Units 1 and 2, Docket Nos. 50-390 and 50-391, located in Rhea County, Tennessee, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation of (1) additional information submitted by the applicant since Supplement No. 11 was issued, and (2) matters that the staff had under review when Supplement No. 11 was issued.

NUREG-0910 R02 S01: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE. * Division of Information Support Services (Post 890205). September 1993. 89pp. 9310120062. 76738:121.

The approved records disposition schedules specify the appropriate duration of retention and the final disposition for records created or maintained by the NRC. NUREG-0910, Revision 2, Supplement 1 makes editorial and administrative changes to the NRC Schedule and forwards 3 sets of changes to the National Archives and Records Administrations's General Record Schedule.

NUREG-0933 S15: A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R. Division of Safety Issue Resolution (Post 880717). May 1993. 186pp. 9306110046. 75338:007.

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 S16: A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R. Division of Safety Issue Resolution (Post 880717). November 1993. 250pp. 9401030183. 77642:001.

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-0936 V11 N04: NRC REGULATORY AGENDA.Quarterly Report,October-December 1992. * Division of Freedom of Information & Publications Services (Post 890205). February 1993. 147pp. 9303110325. 74212:297.

The NRC Regulatory Agenda is a compilation of all rules on which the NRC has recently completed action, or has proposed action, 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.

NUREG-0936 V12 N01: NRC REGULATORY AGENDA.Quarterly Report,January-March 1993. * Division of Freedom of Information & Publications Services (Post 890205). April 1993. 135pp. 9305250050. 75004:072.

See NUREG-0936,V11,N04 abstract.

NUREG-0936 V12 N02: NRC REGULATORY AGENDA.Quarterly Report,April-June 1993. * Division of Freedom of Information & Publications Services (Post 890205). July 1993. 138pp. 9308160147. 76121:001.

See NUREG-0936,V11,N04 abstract.

NUREG-0936 V12 N03: NRC REGULATORY AGENDA.Quarterly Report,July-September 1993. * Division of Freedom of Information & Publications Services (Post 890205). October 1993. 139pp. 9312070041. 77353:279.

See NUREG-0936,V11,N04 abstract.

NUREG-0940 V11 N04: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,October-December 1992. * Ofc of Enforcement (Post 870413). March 1993. 321pp. 9303300180. 74407:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (October - December 1992) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

NUREG-0940 V12 N01: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,January-March 1993. * Ofc of Enforcement (Post 870413). June 1993. 250pp. 9306210211. 75406:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January - March 1993) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

NUREG-0940 V12 N02: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June 1993. * Ofc of Enforcement (Post 870413). September 1993. 395pp. 9310120039. 76737:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1993) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

NUREG-0980 V01 N02: NUCLEAR REGULATORY LEGISLATION.102d Congress. * Office of the General Counsel (Post 860701). October 1993. 522pp. 9401100130. 77736:001.

This document is a compilation of nuclear regulatory legislation and other relevant material through the 102d Congress, 2d Session. This compilation has been prepared for use as a resource document, which the NRC intends to update at the end of every Congress. The contents of NUREG-0980 include The Atomic Energy Act of 1954, as amended; Energy Reorganization Act of 1974, as amended, Uranium Mill Tailings Radiation Control Act of 1978; Low-Level Radioactive Waste Policy Act; Nuclear Waste Policy Act of 1982; and NRC Authorization and Appropriations Acts. Other materials included are statutes and treaties on export licensing, nuclear non-proliferation, and environmental protection.

NUREG-0980 V02 N02: NUCLEAR REGULATORY LEGISLATION.102d Congress. * Office of the General Counsel (Post 860701). October 1993. 463pp. 9401100134. 77734:001.

See NUREG-0980,V01,N02 abstract.

NUREG-1021 R07: OPERATOR LICENSING EXAMINER STANDARDS. * Division of Reactor Controls & Human Factors (Post 921004). January 1993. 352pp. 9302230189. 64962:001.

The Operator Licensing Examiner Standards provide policy and guidance to NRC examiners and establish the procedures and practices for examining licensees and applicants for reactor operator and senior reactor operator licenses at power reactor facilities pursuant to Part 55 of Title 10 of the Code of Federal Regulations (10 CFR 55). The Examiner Standards are intended to assist NRC examiners and facility licensees to better understand the initial and requalification examination processes and to ensure the equitable and consistent administration of examinations to all applicants. These standards are not a substitute for the operator licensing regulations and are subject to revision or other internal operator licensing policy changes. This revision will officially become effective 90 days after its publication is noticed in the Federal Register. The revised dynamic simulator requalification examination procedure (ES-604) may be used immediately, if requested by the facility licensee. The corporate notification letters issued after the effective date will provide facility licensees with at least 90 days notice that the examinations will be administered in accordance with the revised procedures.

NUREG-1100 V09: BUDGET ESTIMATES.Fiscal Years 1994-1995. * Division of Budget & Analysis (Post 890205). April 1993. 213pp. 9304160046. 74643:208.

This report contains the fiscal year budget justification to Congress. The budget provides estimates for salaries and expenses and for the Office of the Inspector General for fiscal years 1994 and 1995.

NUREG-1125 V14: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS.1992 Annual. * ACRS - Advisory Committee on Reactor Safeguards. April 1993. 212pp. 9304160042. 74612:001.

This compilation contains 50 ACRS reports submitted to the Commission, Executive Director for Operations, or to the Office of Nuclear Regulatory Research, during calendar year 1992. It also includes a report to the Congress on the NRC Safety Research Program. All reports have been made available to the public through the NRC Public Document Room and the U.S. Library of Congress. 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 by chronological order within project name. Part 2 categorizes the reports by the most appropriate generic subject area and by chronological order within subject area.

NUREG-1145 V09: U.S. NUCLEAR REGULATORY COMMISSION 1992 ANNUAL REPORT. * Office of Administration (Post 890205). July 1993. 287pp. 9309090029. 76391:001.

This report covers the major activities, events, decisions and planning that took place during fiscal year 1992 within the U.S. Nuclear Regulatory Commission (NRC) or involving the NRC.

NUREG-1214 R11: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. ALLENSPACH,F. Division of Reactor Inspection & Licensee Performance (Post 921004). February 1993. 129pp. 9303250054. 74363:155.

The Historical Data Summary of the Systematic Assessment of Licensee Performance (SALP) is produced periodically by the U.S. Nuclear Regulatory Commission. This summary provides the results of the assessment for each facility by NRC region and is further divided into the following sections: Section 1 presents the most recent SALP report ratings for facilities in operation and under construction; Section 2 presents a chronological listing of all SALP report ratings for each operating facility; Section 3 presents a chronological listing of all SALP report ratings for each facility under construction. For historical purposes, past construction ratings for facilities that recently have been licensed also are listed in Section 3.

NUREG-1214 R12: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. ALLENSPACH,F. Division of Reactor Inspection & Licensee Performance (Post 921004). August 1993. 134pp. 9309210018. 76500:101.

See NUREG-1214,R11 abstract.

NUREG-1220 R01: TRAINING REVIEW CRITERIA AND PROCEDURES. * Division of Licensee Performance & Quality Evaluation (870411-921003). January 1993. 104pp. 9303150141. 74242:188.

This document provides direction to NRC personnel for reviewing training programs at nuclear power plants to verify compliance with the requirements of 10 CFR 50.120 and 10 CFR 55 as applicable. It describes the process for evaluating the effectiveness of training programs, provides aids for collection of information during interviews and observations, and provides criteria for evaluating the implementation of a systems approach to training. This document is not intended to have the effect of a regulation, it establishes no binding requirements or interpretations of NRC regulations. It is intended as guidance only.

NUREG-1266 V07: NRC SAFETY RESEARCH IN SUPPORT OF REGULATION - FY 1992. * Office of Nuclear Regulatory Research (Post 860720). May 1993. 76pp. 9306210373. 75401:319.

This report, the eighth in a series of annual reports, was prepared in response to congressional inquiries concerning how nuclear regulatory research is used. It summarizes the accomplishments of the Office of Nuclear Regulatory Research during FY 1992. A special emphasis on accomplishments in nuclear power plant aging research reflects recognition that a number of plants are entering the final portion of their original 40-year operating licenses and that, in addition to current aging effects a focus on safety considerations for license renewal becomes timely. The primary purpose of performing regulatory research is to develop and provide the Commission and its staff with the technical bases for regulatory decisions on the safe operation of licensed nuclear reactors and facilities, to find unknown or unexpected safety problems, and to develop data and related information for the purpose of revising the Commission's rules, regulatory guides, or other guidance.

NUREG-1272 V07 N01: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA.1992 Annual Report - Power Reactors. * Office for Analysis & Evaluation of Operational Data, Director. July 1993. 300pp. 9309210044. 76483:206.

The annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) is devoted to the activities performed during 1992. The report is published in two separate parts. NUREG-1272, Vol. 7, No. 1, covers power reactors and presents an overview of the operating experience of the nuclear power industry from the NRC perspective, including comments about the trends of some key performance measures. The report also includes the principal findings and issues identified in AEOD studies over the past year and summarizes information from such sources as licensee event reports, diagnostic evaluations, and reports to the NRC's Operations Center. The reports contain a discussion of the Incident Investigation Team program and summarize the Incident Investigation Team and Augmented Inspection Team reports for that group of licensees. NUREG-1272, Vol. 7, No. 2, covers nonreactors and presents a review of the events and concerns during 1992 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations. Each volume contains a list of the AEOD reports issued for 1984-1992.

NUREG-1272 V07 N02: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA. 1992 Annual Report - Nonreactors. * Office for Analysis & Evaluation of Operational Data, Director. October 1993. 164pp. 9312070235. 77350:001.

The annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) is devoted to the activities performed during 1992. The report is published in two separate parts. NUREG-1272, Vol. 7, No. 1, covers power reactors and presents an overview of the operating experience of the nuclear power industry from the NRC perspective, including comments about the trends of some key performance measures. The report also includes the principal findings and issues identified in AEOD studies over the past year and summarizes information from such sources as licensee event reports, diagnostic evaluations, and reports to the NRC's Operations Center. NUREG-1272, Vol. 7, No. 2, covers nonreactors and presents a review of the events and concerns during 1992 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations. Both reports also contain a discussion of the Incident Investigation Team program and summarize both the Incident Investigation Team and Augmented Inspection Team reports. Each volume contains a list of the AEOD reports issued for 1981-1992.

NUREG-1275 V09: OPERATING EXPERIENCE FEEDBACK REPORT - PRESSURE LOCKING AND THERMAL BINDING OF GATE VALVES.Commercial Power Reactors. HSU,C. Division of Safety Programs (Post 870413). March 1993. 30pp. 9304020327. AEOD/S92-07. 74435:275.

The potential for valve inoperability caused by pressure locking and thermal binding has been known for many years in the nuclear industry. In spite of numerous generic communications issued in the past by the Nuclear Regulatory Commission (NRC) and industry, pressure locking and thermal binding continues to occur to gate valves installed in safety-related systems of both boiling water reactors (BWRs) and pressurized water reactors (PWRs). The generic communications to date have not led to effective industry action to fully identify, evaluate, and correct the problem. This report identifies: (1) conditions when the failure mechanisms have occurred; (2) the spectrum of safety systems that have been subjected to the failure mechanisms; and (3) conditions that may introduce the failure mechanisms under both normal and accident conditions. On the basis of the evaluation of the operating events, the Office for Analysis and Evaluation of Operational Data (AEOD) of the NRC concludes that the binding problems with gate valves are an important safety issue that needs priority NRC and industry attention. This report also provides AEOD's recommendation for actions to effectively prevent the occurrence of valve binding failures.

NUREG-1307 R03: REPORT ON WASTE BURIAL CHARGES.Escalation Of Decommissioning Waste Disposal Costs At Low-Level Waste Burial Facilities. * Division of Regulatory Applications (Post 870413). May 1993. 59pp. 9305110042. 75338:203.

One of the requirements placed upon nuclear power reactor licensees by the U.S. Nuclear Regulatory Commission (NRC) is for the licensees to periodically adjust the estimate of the cost of decommissioning their plants, in dollars of the current year, as part of the process to provide reasonable assurance that adequate funds for decommissioning will be available when needed. This report, which is scheduled to be revised annually, contains the development of a formula for escalating decommissioning cost estimates that is acceptable to the NRC, and contains values for the escalation of radioactive waste burial costs, by site and by year. The licensees may use the formula, the coefficients, and the burial escalation from this report in their escalation analyses, or they may use an escalation rate at least equal to the escalation approach presented herein. Revision 3 of this report corrects several errors in the calculations and disposal costs for the reference PWR and the reference BWR.

NUREG-1350 V05: NUCLEAR REGULATORY COMMISSION INFORMATION DIGEST.1993 Edition. OLIVE,K.L. Division of Budget & Analysis (Post 890205). March 1993. 127pp. 9305250029. 75005:098.

The Nuclear Regulatory Commission Information Digest (digest) provides a summary of information about the U.S. Nuclear Regulatory Commission (NRC), NRC's regulatory responsibilities, the activities NRC licenses, and general information on domestic and worldwide nuclear energy. The digest, published annually, is a compilation of nuclear and NRC-related data and is designed to provide a quick reference to major facts about the agency and the industry it regulates. In general, the data cover 1975 through 1992, with exceptions noted. Information on generating capacity and average capacity factor for operating U.S. commercial nuclear power reactors is obtained from monthly operating reports that are submitted directly to the NRC by the licensee. This information is reviewed by the NRC for consistency only and no independent validation and/or verification is performed.

NUREG-1363 V05: ATOMIC SAFETY AND LICENSING BOARD PANEL ANNUAL REPORT.Fiscal Year 1992. COTTER,B.P. Atomic Safety & Licensing Board Panel. September 1993. 43pp. 9310120269. 76761:001.

In Fiscal Year 1992, the Atomic Safety and Licensing Board Panel ("the Panel") handled 38 proceedings. The cases addressed issues in the construction, operation, and maintenance of commercial nuclear power reactors and other activities requiring a license from the Nuclear Regulatory Commission. This report sets out the Panel's caseload during the year and summarizes, highlights, and analyzes how the wide-ranging issues raised in those proceedings were addressed by the Panel's judges and licensing boards.

NUREG-1364: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC SAFETY ISSUE 106: PIPING AND THE USE OF HIGHLY COMBUSTIBLE GASES IN VITAL AREAS. GRAVES,C.C. Division of Safety Issue Resolution (Post 880717). June 1993. 48pp. 9307130111. 75653:153.

Highly combustible gases such as hydrogen, propane, and acetylene are used at all nuclear power plants. Hydrogen is of particular importance because it is stored in large quantities and is distributed and used continuously in buildings containing safety-related equipment. Large hydrogen releases at the hydrogen storage facilities or in these buildings could lead to fires or explosions that might result in loss of safety-related equipment. This report gives the regulatory analysis for the resolution of Generic Safety Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas." Scoping analyses showed that the risk associated with the storage and distribution of hydrogen for cooling electric generators at boiling water reactors (BWRs), the off-gas system at BWRs, the waste gas system at pressurized-water reactors (PWRs), and station battery rooms and portable bottles of combustible gas used for maintenance at PWRs and BWRs is small. On the basis of generic evaluations, the NRC staff has concluded that several possible methods to reduce risk could provide cost-effective safety benefits at some plants. However, in view of the observed large differences in plant-specific characteristics affecting the risk associated with the use of hydrogen, and the marginal generic safety benefits that can be achieved in a cost-effective manner, it is recommended that this generic issue be resolved simply by making these results available in a generic letter. This information may help licensees in their plant evaluations recommended by Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events for Severe Accident Vulnerabilities," June 28, 1991.

NUREG-1366: IMPROVEMENTS TO TECHNICAL SPECIFICATIONS SURVEILLANCE REQUIREMENTS. LOBEL,R.; TJADER,T.R. Division of Operational Events Assessment (870411-921003). December 1992. 93pp. 9301220193. 64652:313.

In August 1983 an NRC task group was formed to investigate problems with surveillance testing required by Technical Specifications, and to recommend approaches to effect improvements. NUREG-1024 ("Technical Specifications-Enhancing Safety Impact") resulted, and it contained recommendations to review the basis for test frequencies; to ensure that the tests promote safety and do not degrade equipment; and to review surveillance tests so that they do not unnecessarily burden personnel. The Technical Specifications Improvement Program (TSIP) was established in December 1984 to provide the framework for rewriting and improving the Technical Specifications. As an element of the TSIP, all Technical Specifications surveillance requirements were comprehensively examined as recommended in NUREG-1024. The results of that effort are presented in this report. The study found that while some testing at power is essential, safety can be improved, equipment degradation decreased, and unnecessary personnel burden relaxed by reducing the amount of testing at power.

NUREG-1377 R04: NRC RESEARCH PROG[illegible] PLANT AGING. LISTING AND SUMMARIES OF R[illegible] ISSUED THROUGH SEPTEMBER 1993. VORA,J.P. Divis[illegible] of Engineering (Post 870413). December 1993. 118pp. 9401120296. 77770:172.

The U.S. Nuclear Regulatory Commission is conducting the Nuclear Plant Aging Research (NPAR) Program. This is a comprehensive hardware-oriented engineering research program focused on understanding the aging mechanisms of components and systems in nuclear power plants. The NPAR program also focuses on methods for simulating and monitoring the aging-related degradation of these components and systems. In addition, it provides recommendations for effective maintenance to manage aging and for the implementation of the research results in the regulatory process. This document contains a listing and index of reports generated in the NPAR program that were issued through September 1933 and summaries of those reports. Each summary describes the elements of the research covered in the report and outlines the significant results. For the convenience of the user, the reports are indexed by personal author, corporate author, and subject.

NUREG-1400: AIR SAMPLING IN THE WORKPLACE.Final Report. HICKEY,E.E.; STOETZEL,G.A.; STROM,D.J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1993. 104pp. 9310120325. 76742:181.

This report provides technical information on air sampling that will be useful for facilities following the recommendations in the NRC's Regulatory Guide 8.25, Revision 1, "Air Sampling in the Workplace." That guide addresses air sampling to meet the requirements in NRC's regulations on radiation protection, 10CFR20. This report describes how to determine the need for air sampling based on the amount of material in process modified by the type of material, release potential, and confinement of the material. The purposes of air sampling and how the purposes affect the types of air sampling provided are discussed. The report discusses how to locate air samplers to accurately determine the concentrations of iarborne radioactive materials that workers will be exposed to. The need for and the methods of performing airflow pattern studies to improve the accuracy of air sampling results are included. The report presents and give examples of several techniques that can be used to evaluate whether the airborne concentrations of material are representative of the air inhaled by workers. Methods to adjust derived air concentrations for particle size are described. Methods to calibrate for volume of air sampled and estimate the uncertainty in the volume of air sampled are described. Statistical tests for determining minimum detectable concentrations are presented. How to perform an annual evaluation of the adequacy of the air sampling is also discussed.

NUREG-1415 V05 N02: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report,October 1, 1992 - March 31, 1993. * Office of the Inspector General (Post 890417). April 1993. 38pp. 9306210379. 75401:278.

The Inspector General is required by statute to prepare a semiannual report to Congress which summarizes the significant investigative and audit activities of the office. The 6-month reporting period ends March 31 and September 30. The report is submitted to the Chairman not later than April 30 and October 31, respectively, of each year. The Chairman prepares comments and his own report and submits both reports to Congress.

NUREG-1415 V06 N01: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report,April 1, 1993 - September 30, 1993. NORTON,L.J.; BARCHI,T.J.; FREDERICK,L.; et al. Office of the Inspector General (Post 890417). October 1993. 46pp. 9312160316. 77509:285.

See NUREG-1415,V05,N02 abstract.

NUREG-1423 V04: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON NUCLEAR WASTE.July 1992 - June 1993. * Advisory Committee on Nuclear Waste. August 1993. 81pp. 9309210031. 76482:337.

This compilation contains 17 reports issued by the Advisory Committee on Nuclear Waste (ACNW) during the fifth year of its operation. The reports were submitted to the Chairman and Commissioners of the U.S. Nuclear Regulatory Commission, the Executive Director for Operations, the Director, Office of Nuclear Material Safety and Safeguards, or to the Director, Division of High Level Waste Management, Office of Nuclear Material Safety and Safeguards. All reports prepared by the Committee have been made available to the public through the NRC Public Document Room and the U.S. Library of Congress.

NUREG-1427: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC ISSUE 143: AVAILABILITY OF CHILLED WATER SYSTEM AND ROOM COOLING. LEUNG,V.T. Division of Safety Issue Resolution (Post 880717). December 1993. 79pp. 9401140029. 77797:001.

This report presents the regulatory analysis for Generic Issue (GI-143), "Availability of Chilled Water System and Room Cooling." The heating, ventilating, and a[illegible]ditioning (HVAC) systems and related auxiliaries are required to provide control in environmental conditions in areas in light water reactor (LWR) plants that contain safety-related equipment. In some plants, the HVAC and chilled water systems serve to maintain a suitable environment for both safety and non-safety-related areas. Although some plants have an independent chilled water system for the safety-related areas, the heat removal capability often depends on the operability of other supporting systems such as the service water system or the component cooling water system. The operability of safety-related components depends upon operation of the HVAC and chilled water systems to remove heat from areas containing the equipment. If cooling to dissipate the heat generated is unavailable, the ability of the safety-related equipment to operate as intended cannot be assured. Typical components or areas in the nuclear power plant that could be affected by the failure of cooling from HVAC or chilled water systems include the (1) emergency switchgear and battery rooms, (2) emergency diesel generator room, (3) pump rooms for residual heat removal, reactor core isolation cooling, high-pressure core spray, and low-pressure core spray, and (4) control room. The unavailability of such safety-related equipment or areas could cause the core damage frequency (CDF) to increase significantly.

NUREG-1444: SITE DECOMMISSIONING MANAGEMENT PLAN. FAUVER,D.N.; AUSTIN,J.H.; JOHNSON,T.C.; et al. Division of Low-Level Waste Management & Decommissioning (Post 870413). October 1993. 200pp. 9311080087. 77069:099.

The Nuclear Regulatory Commission (NRC) staff has identified 48 sites contaminated with radioactive material that require special attention to ensure timely decommissioning. While none of these sites represent an immediate threat to public health and safety, they have contamination that exceeds existing NRC criteria for unrestricted use. All of these sites require some degree of remediation, and several involve regulatory issues that must be addressed by the Commission before they can be released for unrestricted use and the applicable licenses terminated. This report contains the NRC staff's strategy for addressing the technical, legal, and policy issues affecting the timely decommissioning of the 48 sites and describes the status of decommissioning activities at the sites.

NUREG-1449: SHUTDOWN AND LOW-POWER OPERATION AT NUCLEAR POWER PLANTS IN THE UNITED STATES.Final Report. * Division of Systems Safety & Analysis (Post 921004). Septemb[illegible]. 200pp. 9310130052. 76743:001.

The r[illegible]ontains the results of the NRC staff's evaluation of sh[illegible] and low-power operations at U.S. commercial nucle[illegible] plants. The report describes studies conducted by the sta[illegible]n the following areas: operating experience related to shutdown and low-power operations, probabilistic risk assessment of shutdown and low-power conditions and utility programs for planning and conducting activities during periods the plant is shut down. The report also documents evaluations of a number of technical issues regarding shutdown and low-power operations performed by the staff, including the principal findings and conclusions. Potential new regulatory requirements are discussed, as well as potential changes in NRC programs. A draft report was issued for comment in February 1992. This report is the final version and includes the responses to the comments along with the staff regulatory analysis of potential new requirements.

NUREG-1453: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC ISSUE 142: LEAKAGE THROUGH ELECTRICAL ISOLATORS IN INSTRUMENTATION CIRCUITS. ROURK,C.J. Division of Safety Issue Resolution (Post 880717). September 1993. 21pp. 9310130042. 76744:266.

Generic Issue (GI) 142 deals with staff concerns about the design of isolation devices used to ensure separation between Class 1E and non-Class 1E electrical control and instrumentation circuits. This issue was initiated in June 1987. Staff reviews of the implementation of the Safety Parameter Display System (SPDS) requirement indicated that some isolation devices used to provide an interface between the non-Class 1E SPDS and the Class 1E safety systems would allow signal leakage if electrically challenged. It was unknown if the amount of leakage posed a hazard to safe operation of the Class 1E system. A review of failure records does not reveal any incidents of system damage caused by isolation device challenge. Furthermore, a review of existing PRA data indicates that the safety significance of ID challenge is low, at the expected challenge event frequency. However, based upon the potential design variations in future control systems resulting from application of computer technology, additional design and qualification test requirements for future plants are recommended.

NUREG-1461: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC ISSUE 153: LOSS OF ESSENTIAL SERVICE WATER IN LWRS. SU,T.-M. Division of Safety Issue Resolution (Post 880717). August 1993. 32pp. 9309030241. 76328:181.

In this report, the staff of the U.S. Nuclear Regulatory Commission provides a regulatory analysis for the proposed resolution of Generic Issue 153 (GI-153), "Loss of Essential Service Water in LWRs." GI-153 deals with the concerns pertaining to the reliability of essential service water (ESW) system and related problems for all light water reactors except the seven multi-unit sites addressed by GI-130, "Essential Service Water Pump failures at Multi-Unit Sites." On the basis of the technical findings of a scoping study for GI-153, the staff recommends that the insights gained from the study serve as a complement to the on-going ESW performance inspection program. The staff also concludes that ESW system reliability is being addressed by various on-going regulatory programs. Therefore, the staff recommends that GI-153 should be considered "RESOLVED." The need for future action(s) on ESW reliability is expected to be determined from these on-going programs.

NUREG-1463: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC SAFETY ISSUE 105: INTERFACING SYSTEM LOSS-OF-COOLANT ACCIDENT IN LIGHT-WATER REACTORS. * Division of Safety Issue Resolution (Post 880717). July 1993. 77pp. 9308160153. 76114:001.

An interfacing systems loss of coolant accident (ISLOCA) involves failure or improper operation of pressure isolation valves (PIVs) that compose the boundary between the reactor coolant system and low-pressure rated systems. Some ISLOCAs can bypass containment and result in direct release of fission products to the environment. A cost/benefit evaluation, using three PWR analyses, calculated the benefit of two potential modifications to the plants. Alternative I is improved plant operations to optimize the operator's performance and reduce human error probabilities. Alternative II adds pressure sensing devices, cabling, and instrumentation between two PIVs to provide operators with continuous monitoring of the first PIV. These two alternatives were evaluated for the base case plants (Case 1) and for each plant, assuming the plants had a particular auxiliary building design in which severe flooding would be a problem if an ISLOCA occurred. The auxiliary building design (Case 2) was selected from a survey that revealed a number of designs with features that provided less than optimal resistance to ECCS equipment loss caused by a ISLOCA-induced environment. The results were judged not to provide sufficient basis for generic requirements. It was concluded that the most viable course of action to resolve Generic Issue 105 is licensee participation in individual plant examinations (IPEs).

NUREG-1467: FEDERAL GUIDE FOR A RADIOLOGICAL RESPONSE.Supporting The Nuclear Regulatory Commission During The Initial Hours Of A Serious Accident. HOGAN,R.T. Division of Operational Assessment (Post 870413). November 1993. 20pp. 9401030173. 77642:290.

This document is a planning guide for those Federal agencies that work with the Nuclear Regulatory Commission (NRC) during the initial hours of response to a serious radiological emergency in which the NRC is the Lead Federal Agency (LFA). These Federal agencies are: DOE, EPA, USDA, HHS, NOAA, and FEMA. This guide is intended to help these agencies prepare for a prompt response. Instructions are provided on receiving the initial notification, the type of person to send to the scene, the facility at which people are needed, how to get them to that facility, and what they should do when they arrive. Federal agencies not specifically mentioned in this guide may also be asked to support the NRC.

NUREG-1470 V02: CHIEF FINANCIAL OFFICER'S ANNUAL REPORT -- 1993. * Office of the Controller (Post 890205). September 1993. 116pp. 9311030220. 77029:003.

The Chief Financial Officers Act of 1990 requires the NRC Chief Financial Officer to prepare and submit an annual report to the agency head and the Director of the Office of Management and Budget. This 1993 report is the second annual report for the NRC and includes a description and analysis of the status of financial management for Fiscal Year 1993, an audited financial statement and audit reports for Fiscal Year 1992, and a summary of the reports on internal accounting and administrative control systems for 1992.

NUREG-1472: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC ISSUE 57. Effects Of Fire Protection System Actuation On Safety-Related Equipment. WOODS,H.W. Division of Safety Issue Resolution (Post 880717). October 1993. 40pp. 9311080077. 77068:321.

Actuation of Fire Protection Systems (FPS) in Nuclear Power Plants have resulted in adverse interactions with equipment important to safety. Precursor operational experience has shown that 37% of all FPS actuations damaged some equipment, and 20% of all FPS actuations have resulted in a plant transient and reactor trip. On an average, 0.17 FPS actuations per reactor year have been experienced in nuclear power plants in this country. This report presents the regulatory analysis for GI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment". The risk reduction estimates, cost/benefit analyses, and other insights gained during this effort have shown that implementation of the recommendations contained in this report can significantly reduce risk, and that these improvements can be warranted in accordance with the backfit rule, 10 CFR 50.109(a)(3). However, plant specific analyses are required in order to identify such improvements. Generic analyses can not serve to identify improvements that could be warranted for individual, specific plants. Plant specific analyses of the type needed for this purpose are underway as part of the Individual Plant Examination of External Events (IPEEE) program.

NUREG-1473: ELECTRICAL DISTRIBUTION SYSTEM FUNCTIONAL INSPECTION (EDSFI) DATA BASE PROGRAM. GAUTAM,A.S. Division of Reactor Inspection & Licensee Performance (Post 921004). January 1993. 45pp. 9303120055. 74237:224.

This document describes the organization, installation procedures, and operating instructions for the database computer program containing inspection findings from the U.S. Nuclear Regulatory Commission's (NRC's) Electrical Distribution System Functional Inspections (EDSFIs). The program enables the user to search and sort findings, ascertain trends, and obtain printed reports of the findings. The findings include observations, unresolved issues, or possible deficiencies in the design and implementation of electrical distribution systems in nuclear plants. This database will assist those preparing for electrical inspections, searching for deficiencies in a plant, and determining the corrective actions previously taken for similar deficiencies. This database will be updated as new EDSFIs are completed.

NUREG-1474: EFFECT OF HURRICANE ANDREW ON THE TURKEY POINT NUCLEAR GENERATING STATION FROM AUGUST 20-30, 1992. HEBDON,F.J. Office for Analysis & Evaluation of Operational Data, Director. * Institute of Nuclear Power Operations. March 1993. 80pp. 9307060041. 75585:001.

On August 24, 1992, Hurricane Andrew, a Category 4 hurricane, struck the Turkey Point Electrical Generating Station with sustained winds of 145 mph (233 km/h). This is the report of the team that the U.S. Nuclear Regulatory Commission and the Institute of Nuclear Power Operations jointly sponsored: (1) to review the damage that the hurricane caused the nuclear units and the utility's actions to prepare for the storm and recover from it; and (2) to compile lessons that might benefit other nuclear reactor facilities.

NUREG-1476: FINAL ENVIRONMENTAL IMPACT STATEMENT TO CONSTRUCT AND OPERATE A FACILITY TO RECEIVE,STORE, AND DISPOSE OF 11E.(2) BYPRODUCT MATERIAL NEAR CLIVE,UTAH.Docket No. 40-8989, Envirocare Of Utah,Inc. BRUMMETT,E.; ABU-EID,R.; MULLINS,A.; et al. Division of Low-Level Waste Management & Decommissioning (Post 870413). August 1993. 206pp. 9309210022. 76481:269.

A Final Environmental Impact Statement (FEIS) related to the licensing of Envirocare of Utah, Inc.'s proposed disposal facility in Tooele County, Utah (Docket No. 40-8989) for byproduct material as defined in Section 11e.(2) of the Atomic Energy Act, as amended, has been prepared by the Office of Nuclear Material Safety and Safeguards. This statement describes and evaluates: (1) the purpose of and need for the proposed action; (2) the alternatives considered; and (3) the environmental consequences of the proposed action. The NRC has concluded that the proposed action evaluated under the National Environmental Policy Act of 1969 (NEPA) and 10 CFR Part 51, is to permit the applicant to proceed with the project as described in this Statement.

NUREG-1476 DRFT: DRAFT ENVIRONMENTAL IMPACT STATEMENT TO CONSTRUCT AND OPERATE A FACILITY TO RECEIVE, STORE, AND DISPOSE OF 11E.(2) BYPRODUCT MATERIAL NEAR CLIVE, UTAH.Docket No. 40-8989, Envirocare Of Utah, Inc. BRUMMETT,E.; ABU-EID,R.; MULLINS,A.; et al. Division of Low-Level Waste Management & Decommissioning (Post 870413). February 1993. 224pp. 9303120020. 74238:001.

A Draft Environmental Statement (DEIS) related to the licensing of Envirocare of Utah, Inc.'s proposed disposal facility in Tooele County, Utah, (Docket No. 40-8989) for byproduct material as defined in Section 11e.(2) of the Atomic Energy Act, as amended, has been prepared by the Office of Nuclear Material Safety and Safeguards. This statement describes and evaluates (1) the purpose of and need for the proposed action, (2) the alternatives considered, and (3) the environmental consequences of the proposed action. The NRC has concluded that the proposed action evaluated under the National Environmental Policy Act of 1969 (NEPA) and 10 CFR Part 51, is to permit the applicant to proceed with the project as described in this Statement.

NUREG-1477 DRFT FC: VOLTAGE-BASED INTERIM PLUGGING CRITERIA FOR STEAM GENERATOR TUBES.Draft Report For Comment. * IPC Task Group. June 1993. 120pp. 9307060061. 75563:068.

This report presents the preliminary results of a special U.S. Nuclear Regulatory Commission (NRC) task group established: (1) to review the technical bases for and outstanding issues related to interim approval of voltage-based interim plugging criteria for outside-diameter stress corrosion cracking (ODSCC) of steam generator tubes; and (2) to prepare conclusions and recommendations concerning implementation of these criteria. The task group activities included identification and assessment of outstanding technical issues and concerns that had previously been raised regarding voltage-based plugging criteria for ODSCC. Most of these issues are relevant to the long-term approval of voltage-based plugging criteria. This report describes the results of the task group's review and evaluation of: (1) the issues related to tube integrity, including the potential for tube rupture or leakage under postulated-accident conditions and the safety implications of these issues; (2) the radiological doses and the potential for core damage associated with a range of assumed primary-to-secondary leak rates; and (3) the safety significance of ODSCC of steam generator tubes.

NUREG-1479: RESULTS FROM TWO WORKSHOPS: STATE RADIATION CONTROL PROGRAMS DEVELOPING AND AMENDING REGULATIONS AND FUNDING. PARKER,G. Office of State Programs (Post 911117). September 1993. 34pp. 9310130049. 76743:190.

The first section of this document presents the results of a technical workshop on the process of regulations development and amendment sponsored by the Nuclear Regulatory Commission (NRC). This workshop focused on methods for reducing the time it takes to promulgate regulations to help those States that are having difficulty meeting the three-year deadline for adopting new NRC regulations. Workshop participants responded to six questions, reviewed the procedures used by the various States for revising and adopting changes to their regulations, and reviewed the time-flow charts used by various States. This workshop was designed to provide guidance to States that are promulgating and revising regulations. The second section of this document summarizes the proceedings of a technical workshop, also sponsored by the NRC, on funding radiation control programs that emphasized fee schedules and effective strategies for the 1990s. This workshop focused on determining the true costs of running a program, on setting realistic fees for the various categories of licenses, and on the most efficient methods for sending invoices, recording receipts, depositing money received, and issuing licenses. Workshop participants responded to seven questions; reviewed the methods various States use to determine true costs; reviewed the procedure that the various States use to produce invoices and licenses; reviewed the procedures that the States are required to abide by when they receive money; and reviewed the method used by the NRC to determine the cost of its various programs.

NUREG-1480: LOSS OF AN IRIDIUM-192 SOURCE AND THERAPY MISADMINISTRATION AT INDIANA REGIONAL CANCER CENTER,INDIANA,PENNSYLVANIA,ON NOVEMBER 16, 1992. * NRC - No Detailed Affiliation Given. February 1993. 223pp. 9303120040. 74263:047.

On December 1, 1992, the Indiana Regional Cancer Center reported to the U.S. Nuclear Regulatory Commission's (NRC) Region I that they believed a 1.37 E + 11 becquerel (3.7-curie) iridium-192 source from their Omnitron 2000 high dose rate remote brachytherapy afterloader had been found at a biohazard waste transfer station in Carnegie, Pennsylvania. After notifying the NRC, this cancer center, one of several operated by the licensee, Oncology Services Corporation, retrieved the source, and Region I dispatched an inspector and a supervisor to investigate the event. The source was first detected when it triggered radiation alarms at a waste incinerator facility in Warren, Ohio. The licensee informed the NRC that the source wire had apparently broken during treatment of a patient on November 16, 1992, leaving the source in the patient. On the basis of the seriousness of the incident, the NRC elevated its response to an Incident Investigation. The Incident Investigation Team initiated its investigation on December 3, 1992. The investigation team concluded that the patient received a serious misadministration and died on November 21, 1992, and that over 90 individuals were exposed to radiation from November 16 to December 1, 1992. In a press release dated January 26, 1993, the, Indiana County Coroner stated that the cause of death listed in the official autopsy report was 'Acute Radiational

Exposure and Consequences Thereof.' An almost identical source-wire failure occurred with an afterloader in Pittsburgh, Pennsylvania, on December 7, 1992, but with minimal radiological consequences. This incident was included in the investigation. This report discusses the Omnitron 2000 high dose rate afterloader source-wire failure, the reasons why the failure was not detected by Indiana Regional Cancer Center, the potential consequences to the patient, the estimated radiological doses to workers and the public, and regulatory aspects associated with this incident.

NUREG-1482 DRFT FC: GUIDELINES FOR INSERVICE TESTING AT NUCLEAR POWER PLANTS. Draft Report For Comment. CAMPBELL,P. Division of Engineering (Post 921004). November 1993. 200pp. 9312220172. 77544:159.

In this report, the staff gives guidelines for developing and implementing programs for the inservice testing of pumps and valves at commercial nuclear power plants. The report includes U.S. Nuclear Regulatory Commission (NRC) guidance and recommendations on inservice testing issues. The staff discusses the regulations, the components to be included in an inservice testing program, and the preparation and content of cold shutdown and refueling outage justifications and requests for relief from the American Society of Mechanical Engineers Code requirements. The staff also gives specific guidance on relief acceptable to the NRC and advises licensees in the use of this information for application at their facilities. The staff discusses the revised standard technical specifications for the inservice testing program requirements and gives guidance on the process a licensee may follow upon finding an instance of noncompliance with the Code.

NUREG-1484 DRFT: DRAFT ENVIRONMENTAL IMPACT STATEMENT FOR THE CONSTRUCTION AND OPERATION OF CLAIBORNE ENRICHMENT CENTER,HOMER, LOUISIANA.Docket No. 70-3070,Louisiana Energy Services,L.P. * Division of Fuel Cycle Safety & Safeguards (Post 930207). November 1993. 361pp. 9312070063. 77354:061.

This Draft Environmental Impact Statement (DEIS) was prepared by the Nuclear Regulatory Commission in accordance with NRC regulation 10 CFR Part 51, which implements the National Environmental Policy Act (NEPA), to assess the potential environmental impacts of the construction and operation of a proposed gaseous centrifuge enrichment facility to be built in Claiborne Parish, LA. The proposed facility will have a production capacity of about 866 tonnes annually of up to 5 percent enriched UF(6), using a proven centrifuge technology. Included in the assessment are construction, both normal operations and potential accidents (internal and external events), and the eventual decontamination and decommissioning of the site. In order to help assure that releases from the operation of the facility and potential impacts on the public are as low as reasonably achievable, an environmental monitoring program was developed to detect significant changes in the background levels of uranium around the site. Other issues addressed include the purpose and need for the facility, the alternatives to the proposed action, and the site selection process. The NRC concludes that the facility can be constructed and operated with small and acceptable impacts on the public and the environment, and proposes to issue a license to the applicant, Louisiana Energy Services, to authorize construction and operation of the proposed facility. This DEIS provides the public with the opportunity to comment on the proposed action and the treatment of potential environmental impacts.

NUREG-1485: UNAUTHORIZED FORCED ENTRY INTO THE PROTECTED AREA AT THREE MILE ISLAND UNIT 1 ON FEBRUARY 7, 1993. * Ofc of the Executive Director for Operations. April 1993. 142pp. 9304210263. 74677:061.

On February 7, 1993, at 6:53 a.m. Eastern Standard Time (EST) an intruder drove into the site owner-controlled area, through a gate into the protected area of Three Mile Island Nuclear Generating Station, Unit 1 (TMI-1) and crashed through a roll-up door on the Turbine Building. TMI Security reported this event to the U.S. Nuclear Regulatory Commission's (NRC's) Headquarters operations officer and declared a Security Emergency upon determining that the protected area of the plant had been compromised. At 7:23 a.m., the TMI-1 shift supervisor officially notified the NRC Headquarters operations officer that he had declared a Site Area Emergency effective at 7:05 a.m. Upon considering the possible significance to physical security and the regulatory questions that could result from the event, the NRC Executive Director for Operations established an incident investigation team to determine what happened and make appropriate findings and conclusions. In this report the team described the event and the response to the event, evaluated the regulatory requirements, and presented the team's findings and conclusions.

NUREG-1487 V01: FISCAL YEAR 1994-1998 INFORMATION TECHNOLOGY STRATEGIC PLAN. * Office of Information Resources Management (Post 890205). November 1993. 28pp. 9312160341. 77509:332.

A team of senior managers from across the U.S. Nuclear Regulatory Commission (NRC), working with the Office of Information Resources Management (IRM), has completed an NRC Strategic Information Technology (IT) Plan. The Plan addresses three major areas: (1) IT Program Management, (2) IT Infrastructure, and (3) Information and Applications Management. Key recommendations call for accelerating the replacement of Agency workstations, implementing a new document management system, applying business process reengineering to selected Agency work processes, and establishing an Information Technology Council to advise the Director of IRM.

NUREG-1488 DRFT FC: REVISED LIVERMORE SEISMIC HAZARD ESTIMATES FOR 69 NUCLEAR POWER PLANT SITES EAST OF THE ROCKY MOUNTAINS.Draft Report For Comment. SOBEL,P. Division of Engineering (Post 921004). October 1993. 98pp. 9311080086. 77069:001.

This report presents updated Lawrence Livermore National Laboratory (LLNL) probabilistic seismic hazard analysis estimates for 69 nuclear power plant sites in the region of the United States east of the Rocky Mountains. LLNL performed a re-elicitation of seismicity and ground motion experts to improve their estimates of uncertainty in seismicity parameters and ground motion models. Using these revised inputs, LLNL updated the seismic hazard estimates documented in NUREG/CR-5250 (1989). These updated hazard estimates will be used in future NRC actions. A detailed summary of the revised probabilistic hazard methodology, the seismicity and ground motion inputs, and sensitivity studies is presented in a report by J. Savy (1993, LLNL Report UCRL-ID-115111). This report summarizes the Savy (1993) report and documents the 1993 LLNL hazard results for the 69 sites. For the purpose of comparing these probabilistic seismic hazard estimates to the seismic design of the nuclear power plants, a table of safe-shutdown earthquake spectral values is included.

NUREG/CP-0040: PROCEEDINGS OF WORKSHOP V: FLOW AND TRANSPORT THROUGH UNSATURATED FRACTURED ROCK -- RELATED TO HIGH-LEVEL RADIOACTIVE WASTE DISPOSAL.Held At Radisson Suite Hotel, Tucson, Arizona,January 7-10,1991. EVANS,D.D.; NICHOLSON,T.J. Arizona, Univ. of, Tucson, AZ. June 1993. 250pp. 9307220313. 75744:119.

The "Workshop on Flow and Transport Through Unsaturated Fractured Rock Related to High-Level Radioactive Waste Disposal" was cosponsored by the NRC, the Center for Nuclear Waste Regulatory Analyses, and the University of Arizona (UAZ) and was held in Tucson, Arizona, on January 7-10, 1991. The focus of this workshop, similar to the earlier four (the first being in 1982), related to hydrogeologic technical issues associated with possible disposal of commercial high-level nuclear waste (HLW) in a geologic repository within an unsaturated fractured rock system which coincides with the UAZ field studies on HLW disposal. The presentations and discussions centered on flow and transport processes and conditions, relevant parameters, as well as state-of-the-art measurement techniques, and modeling capabilities. The workshop consisted of: four half-day technical meetings; a one-day field visit to the Apache Leap test site to review ongoing field studies that are examining site characterization techniques and developing data sets for model validation studies; and a final half-day session devoted to examining research needs related to modeling groundwater flow and radionuclide transport in unsaturated, fractured rock. These proceedings provide extended abstracts of the technical presentations and short summaries of the research group reports.

NUREG/CP-0126 V01: PROCEEDINGS OF THE TWENTIETH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. March 1993. 535pp. 9304260111. 74713:001.

This three-volume report contains 93 papers out of the 108 that were presented at the Twentieth Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, during the week of October 21-23, 1992. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included 10 different papers presented by researchers from CEC, China, Finland, France, Germany, Japan, Spain and Taiwan. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

NUREG/CP-0126 V02: PROCEEDINGS OF THE TWENTIETH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. March 1993. 555pp. 9304190226. 74621:001.

See NUREG/CP-0126,V01 abstract.

NUREG/CP-0126 V03: PROCEEDINGS OF THE TWENTIETH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. March 1993. 585pp. 9304190227. 74623:001.

See NUREG/CP-0126,V01 abstract.

NUREG/CP-0128: PROCEEDINGS OF THE INTERNATIONAL WORKSHOP ON THE CONDUCT OF INSPECTIONS AND INSPECTOR QUALIFICATION AND TRAINING. GRIMES,B.K. Office of Nuclear Reactor Regulation, Director (Post 870411). February 1993. 231pp. 9303160283. NEA/CNRA/R(92)3. 74254:069.

The results of an international workshop on nuclear reactor inspection are presented. Topics include types of inspection programs (resident, teams, centralized), methods of inspection, investigation of incidents/accidents, achieving correction of deficiencies found during inspections, training and qualifications of inspectors, and inspections of shutdown activities and low power operations. Represented at the conference were Belgium, Bulgaria, Canada, CFSR, France, Finland, Germany, Hungary, IAEA, The Netherlands, Norway, OECD, Russia, Spain, Sweden, Switzerland, U.K., U.S., and the Ukraine.

NUREG/CP-0129: PROCEEDINGS OF THE WORKSHOP ON PROGRAM FOR ELIMINATION OF REQUIREMENTS MARGINAL TO SAFETY. DEY,M. Advanced Reactors Branch (Post 910830). ARSENAULT,F.; PATTERSON,M.; et al. SCIENTECH, Inc. September 1993. 194pp. 9310120259. 76741:245.

These are the proceedings of the Public Workshop on the U.S. Nuclear Regulatory Commission's Program for Elimination of Requirements Marginal to Safety. The workshop was held at the Holiday Inn, Bethesda, on April 27 and 28, 1993. The purpose of the workshop was to provide an opportunity for public and industry input to the program. The workshop addressed the institutionalization of the program to review regulations with the purpose of eliminating those that are marginal. The objective is to avoid the dilution of safety efforts. One session was devoted to discussion of the framework for a performance-based regulatory approach. In addition, panelists and attendees discussed scope, schedules, and status of specific regulatory items: containment leakage testing requirements, fire protection requirements, requirements for environmental qualification of electrical equipment, requests for information under 10CFR50.54(f), requirements for combustible gas control systems, and quality assurance requirements.

NUREG/CP-0130 V01: PROCEEDINGS OF THE 22ND DOE/NRC NUCLEAR AIR CLEANING CONFERENCE.Sessions 1-8.Held In Denver,Colorado,August 24-27,1992. FIRST,M.W. Harvard School of Public Health, Boston, MA. July 1993. 500pp. 9307270008. CONF-9020823. 75801:001.

This document contains the papers and the associated discussions of the 22nd DOE/NRC Nuclear Air Cleaning Conference. Major topics are: (1) advanced reactors; (2) reprocessing; (3) filter testing; (4) waste management; (5) instruments and sampling; (6) reactor accidents; (7) filters and filter performance; (8) adsorber testing and performance; (9) carbon testing; and (10) ventilation systems.

NUREG/CP-0130 V02: PROCEEDINGS OF THE 22ND DOE/NRC NUCLEAR AIR CLEANING CONFERENCE.Sessions 9-16.Held In Denver,Colorado,August 24-27, 1992. FIRST,M.W. Harvard School of Public Health, Boston, MA. July 1993. 500pp. 9307270050. CONF-9020823. 75803:070.

See NUREG/CP-0130,V01 abstract.

NUREG/CP-0131: PROCEEDINGS OF THE JOINT IAEA/CSNI SPECIALISTS' MEETING ON FRACTURE MECHANICS VERIFICATION BY LARGE-SCALE TESTING. Held At Pollard Auditorium,Oak Ridge,Tennessee. PUGH,C.E.; BASS,B.R.; KEENEY,J.A. Oak Ridge National Laboratory. October 1993. 1,000pp. 9311080091. ORNL/TM-12413. 77070:001.

This report contains 40 papers that were presented at the Joint IAEA/CSNI Specialists' Meeting - Fracture Mechanics Verification by Large-Scale Testing held at the Pollard Auditorium, Oak Ridge, Tennessee, during the week of October 26-29, 1992. The papers are printed in the order of their presentation in each session and describe recent large-scale fracture (brittle and/or ductile) experiments, analyses of these experiments, and comparisons between predictions and experimental results. The goal of the meeting was to allow international experts to examine the fracture behavior of various materials and structures under conditions relevant to nuclear reactor components and operating environments. The emphasis was on the ability of various fracture models and analysis methods to predict the wide range of experimental data now available. The international nature of the meeting is illustrated by the fact that papers were presented by researchers from CSFR, Finland, France, Germany, Japan, Russia, U.S., and the U.K. There were experts present from several other countries who participated in discussing the results presented. The titles for some of the final papers and the names of the authors have been updated in this report and may differ slightly from those that appeared in the final program of the meeting.

NUREG/CP-0132: TRANSACTIONS OF THE TWENTY-FIRST WATER REACTOR SAFETY INFORMATION MEETING. MONTELEONE,S. Office of Nuclear Regulatory Research (Post 860720). October 1993. 200pp. 9311010020. 76982:042.

This report contains summaries of papers on reactor safety research to be presented at the 21st Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel, Bethesda, Maryland, October 25-27, 1993. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. Summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the Electric Power Research Institute (EPRI), the nuclear industry, and from foreign governments and industry are also included. 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/CP-0134: INTERNATIONAL ATOMIC ENERGY AGENCY SPECIALISTS MEETING ON EXPERIENCE IN AGING,MAINTENANCE, AND MODERNIZATION OF INSTRUMENTATION AND CONTROL SYSTEMS FOR IMPROVING NUCLEAR POWER PLANT AVAILABILITY.Held At Rockville,MD,May 5-7,1993. * International Atomic Energy Agency. * Oak Ridge National Laboratory. October 1993. 580pp. 9312070237. 77350:147.

This report presents the proceedings of the Specialist's Meeting on Experience in Aging, Maintenance and Modernization of Instrumentation and Control Systems for Improving Nuclear Power Plant Availability that was held at the Ramada Inn in Rockville, Maryland on May 5-7, 1993. The Meeting was presented in cooperation with the Electric Power Research Institute, Oak Ridge National Laboratory and the International Atomic Energy Agency. There were approximately 65 participants from 13 countries at the Meeting. The program chairman was Jerry L. Mauck of the U.S. Nuclear Regulatory Commission.

NUREG/CR-2850 V11: DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1989. BAKER,D.A. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1993. 192pp. 9303120086. PNL-4221. 74239:001.

Population and individual radiation dose commitments have been estimated from reported radionuclide releases from commercial power reactors operating during 1989. Fifty-year dose commitments for a one- year exposure from both liquid and atmospheric releases were calculated for four population groups (infant, child, teen-ager and adult) residing between 2 and 80 km from each of 72 reactor sites. This report tabulates the results of these calculations showing the dose commitments for both water and airborne pathways for each age group and organ. Also included for each of the sites is an estimate of individual doses which are compared with 10 CFR Part 50, Appendix I design objectives. The total collective dose commitments (from both liquid and airborne pathways) for each site ranged from a high of 14 person-rem to a low of 0.005 person-rem for the sites with plants in operation and producing power during the year. The arithmetic mean was 1.2 person- rem. The total population dose for all sites was estimated at 84 person-rem for the 140 million people considered at risk. The individual dose commitments estimated for all sites were below the Appendix I design objectives.

NUREG/CR-2907 V11: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1990. TICHLER,J.; DOTY,K.; CONGEMI,J. Brookhaven National Laboratory. October 1993. 360pp. 9311080278. BNL-NUREG-51581. 77076:113.

Releases of radioactive materials in airborne and liquid effluents from commercial light water reactors during 1990 have been compiled and reported. Data on solid waste shipments as well as selected operating information have been included. This report supplements earlier annual reports issued by the former Atomic Energy Commission and the Nuclear Regulatory Commission. The 1990 release data are summarized in tabular form. Data covering specific radionuclides are summarized.

NUREG/CR-3469 V07: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA. KAURIN,D.G.; KHAN,T.A.; SULLIVAN,S.G.; et al. Brookhaven National Laboratory. July 1993. 106pp. 9308160137. BNL-NUREG-51708. 75120:001.

The ALARA Center at Brookhaven National Laboratory publishes a series of bibliographies of selected readings in radiation protection and ALARA in the continuing effort to collect and disseminate information on radiation dose reduction at nuclear power plants. This is volume 7 of the series. The abstracts in this bibliography were selected from proceedings of technical meetings and conferences, journals, research reports, and searches of the Energy Science and Technology database of the U.S. Department of Energy. The subject material of these abstracts relates to radiation protection and dose reduction, and ranges from use of robotics to operational health physics, to water chemistry. Material on the design, planning, and management of nuclear power stations is included, as well as information on decommissioning and safe storage efforts. Volume 7 contains 293 abstracts, an author index, and a subject index. The author index is specific for this volume. The subject index is cumulative and lists all abstract numbers from volumes 1 to 7. The numbers in boldface indicate the abstracts in this volume; the numbers not in boldface represent abstracts in previous volumes.

NUREG/CR-3950 V08: FUEL PERFORMANCE ANNUAL REPORT FOR 1990. PREBLE,E.A.; PAINTER,C.L.; ALVIS,J.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1993. 138pp. 9312220161. PNL-5210. 77543:243.

This annual report, the thirteenth in a series, provides a brief description of fuel performance during 1990 in commercial nuclear power plants. Brief summaries of fuel design changes, fuel surveillance programs, fuel operating experience and trends, fuel problems high-burnup fuel experience, and items of general significance are provided. References to additional, more detailed information, and related NRC evaluations are included where appropriate.

NUREG/CR-4214 R1P2A2: HEALTH EFFECTS MODELS FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Modification Of Models Resulting From Addition Of Effects Of Exposure To Alpha-Emitting Radionuclides.Part II: Scientific Bases For Health.... ABRAHAMSON,S. Wisconsin, Univ. of, Madison, WI. BENDER,M.A. Brookhaven National Laboratory. BOECKER,B.B.; et al. Inhalation Toxicology Research Institute. May 1993. 87pp. 9306020013. LMF-136. 75082:159.

Several studies designed to identify and quantify the potential health effects of accidental releases of radionuclides from nuclear power plants have been sponsored by the Nuclear Regulatory Commission. Report NUREG/CR-4214, Rev. 1, Part II (NRC, 1989a) describes in detail the most recent health effects models that have evolved from these efforts. Since the Part II report was published in 1989, two addenda to that report have been prepared to 1) incorporate other scientific information related to low-LET health effects models and 2) extend the models to consider the possible health consequences of including alpha-emitting actinide radionuclides in the exposure source term. The first addendum was published as NUREG/CR-4214, Rev. 1, Part II, Addendum 1 (NRC, 1991). This report, the second addendum to the Part II report, extends the health effects models to consider chronic irradiation from alpha-emitting radionuclides as well as low-LET sources. Consistent with the organization of past reports, this report has three main sections that address early-occurring and continuing effects, late somatic effects, and genetic effects. These results should be used with the basic NUREG/CR-4214 report and Addendum 1 to obtain current views on potential health effects models for radionuclides released accidentally from nuclear power plants.

NUREG/CR-4214 R2 PT1: HEALTH EFFECTS MODEL FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Part I: Introduction,Integration,And Summary. SCOTT,B.R. Inhalation Toxicology Research Institute. EVANS,J.S. Harvard School of Public Health, Boston, MA. ABRAHAMSON,S.; et al. Wisconsin, Univ. of, Madison, WI. October 1993. 200pp. 9311080251. ITRI-141. 77078:027.

This report is a revision of NUREG/CR-4214, Rev. 1, Part I (1990), "Health Effects Models for Nuclear Power Plant Accident Consequence Analysis." This revision has been made to incorporate changes to the Health Effects Models recommended in two addenda to the NUREG/CR-4214, Rev. 1, Part II, 1989 report. The first of these addenda provided recommended changes to the health effects models for low-LET radiations based on recent reports from UNSCEAR, ICRP and NAS/NRC (BEIR V). The second addendum presented changes needed to incorporate alpha-emitting radionuclides into the accident exposure source term. As in the earlier version of this report, models are provided for early and continuing effects, cancers and thyroid nodules, and genetic effects. Weibull dose-response functions are recommended for evaluating the risks of early and continuing health effects. Three potentially lethal early effects--the hematopoietic, pulmonary, and gastrointestinal syndromes--are considered. Linear and linear-quadratic models are recommended for estimating the risks of seven types of cancer in adults--leukemia, bone, lung, breast, gastrointestinal, thyroid, and "other." For most cancers, both incidence and mortality are addressed. Five classes of genetic disease--dominant, x-linked, aneuploidy, unbalanced translocations, and multifactorial diseases--are also considered. Data are provided that should enable analysts to consider the timing and severity of each type of health risk.

NUREG/CR-4219 V09 N2: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For April-September 1992. PENNELL,W.E. Oak Ridge National Laboratory. November 1993. 132pp. 9312070165. ORNL/TM-9593. 77355:061.

The Heavy-Section Steel Technology (HSST) Program is conducted for the Nuclear Regulatory Commission (NRC) by Oak Ridge National Laboratory (ORNL). The program focus is on the development and validation of technology for the assessment of fracture-prevention margins in commercial nuclear reactor pressure vessels. The HSST Program is organized in 11 tasks: (1) program management, (2) fracture methodology and analysis, (3) material characterization and properties, (4) special technical assistance, (5) fracture analysis computer programs, (6) cleavage-crack initiation, (7) cladding evaluations, (8) pressurized-thermal-shock technology, (9) analysis methods validation, (10) fracture evaluation tests and (11) warm prestressing. The program tasks have been structured to place emphasis on the resolution fracture issues with near-term licensing significance. Resources to execute the research tasks are drawn from ORNL with subcontract support from universities and other research laboratories. Close contact is maintained with the sister Heavy-Section Steel Irradiation (HSSI) Program at ORNL and with related research programs both in the United States and abroad. This report provides an overview of principal developments in each of the eleven program tasks from April 1992, to September 1992.

NUREG/CR-4273: CRACK PROPAGATION IN HIGH STRAIN REGIONS OF SEQUOYAH CONTAINMENT. GREIMANN,L.; FANOUS,F.; BLUHM,D. Iowa State Univ., Ames, IA. March 1993. 48pp. 9304080044. IS-4878. 74494:243.

The rate of release of radioactive materials from a containment during a severe accident has a significant impact on the consequences of the accident. One hypothesis for a containment leakage model states that the containment will develop a controlled, relatively small leak before the pressure reaches the point where a general rupture of the shell occurs. Another states that overall failure will occur with total release of the vessel contents almost instantaneously. The Sequoyah ice condenser containment vessel has been studied for some time to predict the possible location and extent of leakage which could occur during a severe accident. In this work, three critical high strain locations were studied to predict crack propagation from an initially small defect. The 1/2-inch plate near the Sequoyah springline was selected for further study. A detailed finite element model of the region was prepared and a virtual crack extension method for calculating the J integral was developed for use with the general purpose finite element program. The pressure in the model was increased to 78 psi which produced a maximum membrane strain of 6.5 percent. At this point the surface crack was assumed to propagate through the plate and leakage began. Using the virtual crack extension method, two through cracks with different lengths were found to be unstable at this pressure which would allow almost instantaneous release of the vessel contents.

NUREG/CR-4469 V15: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report,October 1991 - March 1992. DOCTOR,S.R.; DIAZ,A.A.; FRILEY,J.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1993. 46pp. 9311010023. PNL-5711. 76982:227.

The Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors (NDE Reliability) Program at the Pacific Northwest Laboratory was established by the Nuclear Regulatory Commission to determine the reliability of current inservice inspection (ISI) techniques and to develop recommendations that will ensure a suitably high inspection reliability. The objectives of this program include determining the reliability of ISI performed on the primary systems of commercial light-water reactors (LWRs); using probabilistic fracture mechanics analysis to determine the impact of NDE unreliability on system safety; and evaluating reliability improvements that can be achieved with improved and advanced technology. A final objective is to formulate recommended revisions to ASME Code and Regulatory requirements, based on material properties, service conditions, and NDE uncertainties. The program scope is limited to ISI of the primary systems including the piping, vessel, and other components inspected in accordance with Section XI of the ASME Code. This is a progress report covering the programmatic work from October 1991 through March 1992.

NUREG/CR-4469 V16: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report,April 1992-September 1992. DOCTOR,S.R.; DIAZ,A.A.; FRILEY,J.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1993. 45pp. 9312170058. PNL-5711. 77516:029.

The Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors (NDE Reliability) Program at the Pacific Northwest Laboratory was established by the Nuclear Regulatory Commission to determine the reliability of current inservice inspection (ISI) techniques and to develop recommendations that will ensure a suitably high inspection reliability. The objectives of this program include determining the reliability of ISI performed on the primary systems of commercial light-water reactors (LWRs); using probabilistic fracture mechanics analysis to determine the impact of NDE unreliability on system safety; and evaluating reliability improvements that can be achieved with improved and advanced technology. A final objective is to formulate recommended revisions to ASME Code and Regulatory requirements, based on material proper- ties, service conditions, and NDE uncertainties. The program scope is limited to ISI of the primary systems including the piping, vessel, and other components inspected in accordance with Section XI of the ASME Code. This is a progress report covering the programmatic work from April 1992 through September 1992.

NUREG/CR-4551 V7R1P1: EVALUATION OF SEVERE ACCIDENT RISKS: ZION UNIT 1.Main Report. PARK,C.K.; CAZZOLI,E.; GRIMSHAW,C.; et al. Brookhaven National Laboratory. March 1993. 200pp. 9304080019. BNL/NUREG-52029. 74498:023.

In support of the Nuclear Regulatory Commission's (NRC's) assessment of the risk from severe accidents at commercial nuclear power plants in the U.S. reported in NUREG-1150, revised calculation of the risk to the general public from severe accidents at the Zion Power Station, Unit 1 has been completed. This power plant, located on the western shore of Lake Michigan on the outskirts of Zion, is operated by the Commonwealth Edison Company. The emphasis in this risk analysis was not on determining a "so-called" point estimate of risk. Rather, it was to determine the distribution of risk, and to discover the uncertainties that account for the breadth of this distribution. Off-site risk initiation by events, both internal to the power station and external to the power station, was assessed.

NUREG/CR-4551V7R1P2A: EVALUATION OF SEVERE ACCIDENT RISKS: ZION UNIT 1.Appendix A. PARK,C.K.; CAZZOLI,E.; GRIMSHAW,C.; et al. Brookhaven National Laboratory. March 1993. 600pp. 9304080029. BNL/NUREG-52029. 74495:001.

See NUREG/CR-4551,V07,R01,PT1 abstract.

NUREG/CR-4551V7R1P2B: EVALUATION OF SEVERE ACCIDENT RISKS: ZION UNIT 1.Appendices B, C, D, And E. PARK,C.K.; CAZZOLI,E.; GRIMSHAW,C.; et al. Brookhaven National Laboratory. March 1993. 300pp. 9304080012. BNL/NUREG-52029. 74497:001.

See NUREG/CR-4551,V07,R01,PT1 abstract.

NUREG/CR-4599 V02 N2: SHORT CRACKS IN PIPING AND PIPING WELDS.Semiannual Report, October 1991 - March 1992. WILKOWSKI,G.M.; BRUST,F.; FRANCINI,R.; et al. Battelle Memorial Institute, Columbus Laboratories. May 1992. 53pp. 9306180295. BMI-2173. 75389:304.

This is the fourth semiannual report of the U.S. Nuclear Regulatory Commission's Short Cracks in Piping and Piping Welds research program. This 4-year program began in March 1990. The overall objective of this program is to verify and improve fracture analyses for circumferentially cracked large-diameter nuclear piping with crack sizes typically used in leak-before-break analyses or in-service flaw evaluations. Progress during this reporting period involved: (1) completing two through-wall-cracked pipe experiments and supplementary material property data, (2) an internal circumferential surface-cracked pipe experiment was completed which showed that the R/t effects on the Net-Section-Collapse predicted loads for surface-cracked pipe to be independent of crack size, (3) the anisotropy investigation showed that pipe dimensions may be as important in determining the out-of-plane crack growth angle as the anisotropy of the toughness, (4) we initiated a probabilistic analysis of LBB to assess the potential changes in the leakage detection criteria in NRC Reg Guide 1.45, and (5) other efforts involved a sensitivity study on the effect of thermal aging of cast stainless steel on the moment-carrying capacity of the pipe as a function of time.

NUREG/CR-4599 V03 N1: SHORT CRACKS IN PIPING AND PIPING WELDS.Semiannual Report, April-September 1992. WILKOWSKI,G.M.; BRUST,F.; FRANCINI,R.; et al. Battelle Memorial Institute, Columbus Laboratories. October 1993. 200pp. 9311080268. BMI-2173. 77077:218.

This is the fifth semiannual report of the U.S. Nuclear Regulatory Commission's Short Cracks in Piping and Piping Welds research program. This 4-year program began in March 1990. The overall objective of this program is to verify and improve fracture analyses for circumferentially cracked large-diameter nuclear piping with crack sizes typically used in leak-before-break analyses or in-service flaw evaluations. During this reporting period, the overall program and results were critically reviewed and consequently several changes to the current program were made to meet the final program objectives. Progress during this reporting period involved: (1) for the surface-cracked pipe evaluations, the tensile and Charpy V-notch data for a carbon-manganese submerged arc weld metal were completed, and 3D finite-element (FE) analyses of uncracked stainless steel pipe experiments to resolve the discrepancies between experimental data and FE predictions were completed. (2) Significant leak-rate analyses for cracked pipe using advanced probabilistic analysis was conducted to provide a technical basis for changes to NRC Reg. Guide 1.45. (3) A new PC-based circumferential surface-cracked pipe code, NRCPIPES Version 1.0, was completed. (4) Subcontracted efforts include: numerical analysis of residual stresses on elastic-plastic fracture of cracks in welds (being conducted at the University of Michigan), and evaluation of "Validity Limits on J-R Curve Determination" (being conducted at Brown University). (5) Finally, technical efforts related to the ASME Section XI pipe flaw evaluation efforts are summarized.

NUREG/CR-4667 V15: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April-September 1992. RUTHER,W.E.; CHUNG,H.M.; CHOPRA,O.K.; et al. Argonne National Laboratory. June 1993. 76pp. 9306290083. ANL-93/2. 75496:107.

This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors (LWRs) during the six months from April 1992 to September 1992. Topics that have been investigated include: (1) fatigue and stress corrosion cracking (SCC) of low-alloy steel used in piping, steam generators, and reactor pressure vessels; (2) EAC of cast stainless steels (SSs); and (3) radiation-induced segregation and irradiation-assisted SCC of Type 304 SS after accumulation of relatively high fluence. Data on fatigue of low-alloy steel in LWR environments have been reviewed. Based on fracture-mechanics models and engineering judgement, interim fatigue design curves were developed that are consistent with available fatigue-life data. Crack growth data were obtained on fracture-mechanics specimens of A533-Gr B and A106-Gr B ferritic steels and on cast austenitic SSs in the as-received and thermally aged conditions in simulated BWR water at 289 degrees C. The data were compared with predictions based on crack growth correlations for ferritic steels in oxygenated water and correlations for wrought austenitic SS in oxygenated water developed at ANL and rates in air from Section XI of the ASME Code. Microchemical and microstructural changes in high-and commercial-purity Type 304 SS specimens from control-blade absorber tubes and a control-blade sheath from operating BWRs were studied by Auger electron spectroscopy and scanning electron microscopy. Slow-strain-rate-tensile tests were conducted on irradiated specimens in air and simulated BWR water.

NUREG/CR-4667 V16: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1992 - March 1993. CHUNG,H.M.; CHOPRA,O.K.; RUTHER,W.E.; et al. Argonne National Laboratory. September 1993. 67pp. 9310120274. ANL-93/27. 76742:069.

This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors (LWRs) during the six months from October 1992 to March 1993. Fatigue and EAC of piping, pressure vessels, and core components in LWRs are important concerns as extended reactor lifetimes are envisaged. Topics that have been investigated include (1) fatigue of low-alloy steel used in piping, steam generators, and reactor pressure vessels, (2) EAC of cast stainless steels (SSs), (3) radiation- induced segregation and irradiation-assisted stress corrosion cracking of Type 304 SS after accumulation of relatively high fluence, and (4) EAC of low-alloy steels. Fatigue tests were conducted on medium- sulfur-content A106-Gr B piping and A533-Gr B pressure vessel steels in simulated PWR water and in air. Additional crack growth data were obtained on fracture-mechanics specimens of cast austenitic SSs in the as-received and thermally aged conditions and chromium-nickel-plated A533-Gr B steel in simulated boiling-water reactor (BWR) water at 289 degrees C. The data were compared with predictions based on crack growth correlations for ferritic steels in oxygenated water and correlations for wrought austenitic SS in oxygenated water developed at ANL and rates in air from Section XI of the ASME Code. Microchemical and microstructural changes in high- and commercial-purity Type 304 SS specimens from control-blade absorber tubes and control-blade sheath from operating BWRs were studied by Auger electron spectroscopy and scanning electron microscopy.

NUREG/CR-4735 V08: EVALUATION AND COMPILATION OF DOE WASTE PACKAGE TEST DATA. Biannual Report,August 1989 - January 1990. INTERRANTE,C.G. Geology & Engineering Branch (Post 910506). FRAKER,A.C.; ESCALANTE,E. National Institute of Standards & Technology (formerly National Bureau of Standa. June 1993. 114pp. 9306290106. 75496:183.

This report summarizes evaluations by the National Institute of Standards and Technology (NIST) of some of the Department of Energy (DOE) activities on waste packages designed for containment of radioactive high-level nuclear waste (HLW) for the six-month period, August 1989 - January 1990. This includes reviews of related materials research and plans, information on the Yucca Mountain, Nevada disposal site activities, and other information regarding supporting research and special assistance. Short discussions are given relating to the publications reviewed and complete reviews and evaluations are included. Reports of other work are included in the Appendices.

NUREG/CR-4744 V07 N1: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,October 1991 - March 1992. CHOPRA,O.K. Argonne National Laboratory. May 1993. 152pp. 9306180315. ANL-92/42. 75387:001.

This progress report summarizes work performed by Argonne National Laboratory on long-term thermal embrittlement of cast duplex stainless steels in LWR systems during the six months from October 1991 to March 1992. Charpy-impact, tensile, and fracture toughness J-R curve data are presented for several heats of cast stainless steel that were aged 10,000-58,000 h at 290, 320, and 350 degrees C. The results indicate that thermal aging decreases the fracture toughness of cast stainless steels. In general, CF-3 steels are the least sensitive to thermal aging and CF-8M steels are the most sensitive. The values of fracture toughness J(IC) and tearing modulus for CF-8M steels can be as low as ≈90 kJ/m(m^2) and ≈60, respectively. The fracture toughness data are consistent with the Charpy-impact results, i.e., unaged and aged steels that show low impact energy also exhibit lower fracture toughness. All steels reach a minimum saturation fracture toughness after thermal aging; the time to reach saturation depends on the aging temperature. The results also indicate that low-strength cast stainless steels are generally insensitive to thermal aging.

NUREG/CR-4744 V07 N2: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,April-September 1992. CHOPRA,O.K. Argonne National Laboratory. July 1993. 54pp. 9308160133. ANL-93/11. 76120:301.

This progress report summarizes work performed by Argonne National Laboratory on long-term thermal embrittlement of cast duplex stainless steels in LWR systems during the six months from April-September 1992. A procedure and correlations are presented for predicting Charpy-impact energy tensile flow stress, fracture toughness J-R curves, tearing modulus and J(Ic) of aged cast stainless steels from known material information. The "saturation" impact strength and fracture toughness of a specific cast stainless steel, i.e., the minimum value that would be achieved for the material after long-term service, is estimated from the chemical composition of the steel. Mechanical properties as a function of time and temperature of reactor service are estimated from impact energy and flow stress of the unaged material and the kinetics of embrittlement, which are also determined from chemical composition. The J(Ic) values are determined from the estimated J-R curve and flow stress. Examples of estimating mechanical properties of cast stainless steel components during reactor service are presented. A common "lower-bound" J-R curve for cast stainless steels of unknown chemical composition is also defined for a given grade of steel, ferrite content, and temperature.

NUREG/CR-4832 V05: ANALYSIS OF THE LASALLE UNIT 2 NUCLEAR POWER PLANT: RISK METHODS INTEGRATION AND EVALUATION PROGRAM.Parameter Estimation Analysis And Screening Human Reliability Analysis. WHEELER,T.A.; SWAIN,A.D.; LAMBRIGHT,J.A.; et al. Sandia National Laboratories. March 1993. 208pp. 9304190161. SAND92-0537. 74643:007.

This volume describes the methodologies used in the data analysis, the screening human error analysis, and the common mode human error analysis performed in support of the LaSalle PRA. Selected results are presented in this volume. The remainder of the results are presented in other volumes of this report where they are actually used. The data review process used in the determination of the data used for the initial screening analysis is described and the final screening data base is given. The final data selection process is described and the final data distributions are presented. The actual implementation of the data base for the integrated accident sequence quantification is described in Volume 2 of this report on Integrated Quantification and Uncertainty Analysis. Several new methods developed for use in analyzing both pre- and post- accident human errors for the initial screening analysis are described. Most of the actual results are given in other volumes of this report under the appropriate sub-analysis descriptions. A method for determining procedural common mode analysis is described and the results presented.

NUREG/CR-4832 V08: ANALYSIS OF THE LASALLE UNIT 2 NUCLEAR POWER PLANT: RISK METHODS INTEGRATION AND EVALUATION PROGRAM (RMIEP).Seismic Analysis. WELLS,J.E.; LAPPA,D.A.; BERNREUTER,D.L.; et al. Lawrence Livermore National Laboratory. November 1993. 290pp. 9312070214. UCID-21245. 77349:002.

This report describes the methodology used and the results obtained from the application of a simplified seismic risk methodology to the LaSalle County Nuclear Generating Station Unit 2. This study is part of the Level I analysis being performed by the Risk Methods Integration and Evaluation Program (RMIEP). Using the RMIEP developed event and fault trees, the analysis resulted in a seismically induced core damage frequency point estimate of 6.0E-7/yr. This result, combined with the component importance analysis, indicated that system failures were dominated by random events. The dominant components included diesel generator failures (failure to swing, failure to start, failure to run after started), and condensate storage tank.

NUREG/CR-4832 V09: ANALYSIS OF THE LASALLE UNIT 2 NUCLEAR POWER PLANT: RISK METHODS INTEGRATION AND EVALUATION PROGRAM (RMIEP).Internal Fire Analysis. LAMBRIGHT,J.A.; BROSSEAU,D.A.; PAYNE,A.C.; et al. Sandia National Laboratories. March 1993. 600pp. 9304260106. SAND92-0537. 74711:001.

This report is a description of the internal fire analysis performed on the LaSalle County Nuclear Generating Station, Unit 2. As part of this effort, a new data base for fires was constructed (NUREG/CR-4586). This data base aided in quantification of fire initiating event frequencies. The most detailed integration between fire risk assessment and internal events analysis, to date, was also accomplished. The same system fault trees used for internal events were utilized for the fire analysis, which included modeling of components down to the contact pair level. Subsidiary equations were created to map the effects of cable failures and spurious actuations. All component and associated cable locations were traced and mapped into the fault trees. A detailed screening analysis was performed which showed most plant areas had a negligible contribution to fire-induced core damage frequency. A detailed analysis of the fire risk resulted in a total (mean) core damage frequency of 3.21E-5 per year.

NUREG/CR-5229 V05: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1992.Annual Report. MCCONNELL,J.W.; ROGERS,R.D.; JASTROW,J.D.; et al. EG&G Idaho, Inc. February 1993. 68pp. 9303120065. EGG-2577. 74235:255.

The Field Lysimeter Investigations: Low-Level Waste Data Base Development Program, funded by the U.S. Nuclear Regulatory Commission, is: (a) studying the degradation effects in EPICOR-II organic ion- exchange resins caused by radiation; (b) examining the adequacy of test procedures recommended in the Branch Technical Position on Waste Form to meet the requirements of 10 CFR 61 using solidified EPICOR-II resins; (c) obtaining performance information on solidified EPICOR-II ion-exchange resins in a disposal environment; and (d) determining the condition of EPICOR-II liners. Results of the seventh year of data acquisition from the field testing are presented and discussed. During the continuing field testing, both Portland type I-II cement and Dow vinyl ester-styrene waste forms are being tested in lysimeter arrays located at Argonne National Laboratory-East in Illinois and at Oak Ridge National Laboratory. The study is designed to provide continuous data on nuclide release and movement, as well as environmental conditions, over a 20-year period.

NUREG/CR-5247 V01 R1: RASCAL VERSION 2.0 USER'S GUIDE. ATHEY,G.F. Phoenix Associates, Inc. SJOREEN,A.L. Oak Ridge National Laboratory. RAMSDELL,J.V.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1993. 187pp. 9303150148. PNL-8454. 74242:001.

The Radiological Assessment System for Consequence AnaLysis; Version 2.0 (RASCAL 2.0) has been developed for use by the NRC personnel who report to the site of a nuclear accident at the time of radiological emergencies. It supplements assessments based on plant conditions and quick estimates based on paper methods and provides rough comparisons to EPA Protective Action Guidance and thresholds for acute health effects. The system, which can be run on any DOS system, was developed to allow consideration of the dominant aspects of source term, transport, dose, and consequences. The model that was previously the whole of RASCAL has been renamed ST-DOSE. Two new models have been added to RASCAL 2.0. The first, FM-DOSE, computes doses from environmental concentrations. The second, DECAY, computes radiologic decay and ingrowth over a selected time period. This volume includes complete instructions for system use.

NUREG/CR-5247 V02: RASCAL VERSION 2.0 WORKBOOK. ATHEY,G.F. Athey Consulting. MCKENNA,T.J. Incident Response Branch. May 1993. 105pp. 9306110031. 75321:069.

The Radiological Assessment System for Consequence Analysis, Version 2.0 (RASCAL 2.0) has been developed for use by the NRC personnel who respond to radiological emergencies. This workbook is intended to complement the RASCAL 2.0 User's Guide (NUREG/CR-5247, Vol. 1). The workbook contains exercises designed to familiarize the user with the computer based tools of RASCAL through hands-on problem solving. The workbook is composed of four major sections. The first part is a RASCAL familiarization exercise to acquaint the user with the operation of the forms, menus, on-line help, and documentation. The latter three parts contain exercises in using the three tools of RASCAL Version 2.0: DECAY, FM-DOSE, and ST-DOSE. Each section of exercises is followed by discussion on how the tools could be used to solve the problem.

NUREG/CR-5305 V02 P1: INTEGRATED RISK ASSESSMENT FOR THE LASALLE UNIT 2 NUCLEAR POWER PLANT.Phenomenology And Risk Uncertainty Evaluation Program (PRUEP) Appendices A-C. BROWN,T.D.; PAYNE,A.C.; MILLER,L.A.; et al. Sandia National Laboratories. May 1993. 200pp. 9306210243. SAND92-2765. 75408:056.

This volume contains a description of the codes and input/output files used to perform the LaSalle Level II/III Probabilistic Risk Assessment. A chart showing the process flow is presented and the relationship between the codes and the needed input and output data is discussed. Code listings for codes not documented elsewhere and complete or sample listings of the input and output files are also presented.

NUREG/CR-5305 V02 P2: INTEGRATED RISK ASSESSMENT FOR THE LASALLE UNIT 2 NUCLEAR POWER PLANT.Phenomenology And Risk Uncertainty Evaluation Program (PRUEP).Appendices D-G. BROWN,T.D.; PAYNE,A.C.; MILLER,L.A.; et al. Sandia National Laboratories. May 1993. 398pp. 9306210303. SAND92-2765. 75404:246.

See NUREG/CR-5305,V02,P1 abstract.

NUREG/CR-5358: REVIEW OF ASME CODE CRITERIA FOR CONTROL OF PRIMARY LOADS ON NUCLEAR PIPING SYSTEM BRANCH CONNECTIONS AND RECOMMENDATIONS FOR ADDITIONAL DEVELOPMENT WORK. RODABAUGH,E.C.; GWALTNEY,R.C.; MOORE,S.E. Oak Ridge National Laboratory. November 1993. 52pp. 9312070270. ORNL/TM-11572. 77352:232.

This report collects and uses available data to reexamine the criteria for controlling primary loads in nuclear piping branch connections as expressed in Section III of the ASME Boiler and Pressure Vessel Code. In particular, the primary load stress indices given in NB-3650 and NB-3683 are reexamined. The report concludes that the present usage of the stress indices in the criteria equations should be continued. However, the complex treatment of combined branch and run moments is not supported by available information. Therefore, it is recommended that this combined loading evaluation procedure be replaced for primary loads by the separate leg evaluation procedure specified in NC/ND-3653.3(c) and NC/ND-3653.3(d). No recommendation is made for fatigue or secondary load evaluations for Class 1 piping. Further work should be done on the development of better criteria for treatment of combined branch and run moment effects.

NUREG/CR-5360: XSOR CODES USERS MANUAL. JOW,H-N.; MURFIN,W.B.; JOHNSON,J.D. Sandia National Laboratories. November 1993. 318pp. 9401030178. SAND89-0943. 77651:001.

This report describes the source term estimation codes, XSORs. The codes are written for three pressurized water reactors (Surry, Sequoyah, and Zion) and two boiling water reactors (Peach Bottom and Grand Gulf). The ensemble of codes has been named "XSOR". The purpose of XSOR codes is to estimate the source terms which would be released to the atmosphere in severe accidents. A source term includes the release fractions of several radionuclide groups, the timing and duration of releases, the rates of energy release, and the elevation of releases. The codes have been developed by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission (NRC) in support of the NUREG-1150 program. The XSOR codes are fast running parametric codes and are used as surrogates for detailed mechanistic codes. The XSOR codes also provide the capability to explore the phenomena and their uncertainty which are not currently modeled by the mechanistic codes. The uncertainty distributions of input parmeters may be used by an XSOR code to estimate the uncertainty of source terms.

NUREG/CR-5404 V02: AUXILIARY FEEDWATER SYSTEM AGING STUDY.Phase I Follow-On Study. KUECK,J.D. Oak Ridge National Laboratory. July 1993. 39pp. 9308160268. ORNL-6566/V1. 76118:327.

This report documents the results of a Phase I follow-on study of the Auxiliary Feedwater (AFW) System that has been conducted for the U.S. Regulatory Commission's Nuclear Plant Aging Research Program. The Phase I study found a number of significant AFW System functions that are not being adequately tested by conventional test methods and some that are actually being degraded by conventional testing. Thus, it was decided that this follow-on study would focus on these testing omissions and equipment degradation. The deficiencies in current monitoring and operating practice are categorized and evaluated. Areas of component degradation caused by current practice are discussed. Recommendations are made for improved diagnostic methods and test procedures.

NUREG/CR-5410: STATISTICALLY BASED REEVALUATION OF PISC-II ROUND ROBIN TEST DATA. HEASLER,P.G.; TAYLOR,T.T.; DOCTOR,S.R. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1993. 150pp. 9306020025. PNL-8577. 75082:001.

This report presents a re-analysis of an international PISC-II round-robin inspection results using formal statistical techniques to account for experimental error. The analysis examines: U.S. team performance vs. other participants performance; flaw sizing performance and errors associated with flaw sizing; factors influencing flaw detection probability; and performance of all participants with respect to recently developed ASME Section XI flaw detection performance demonstration requirements, and develops conclusions concerning ultrasonic inspection capability.

NUREG/CR-5455 V01: DEVELOPMENT OF THE NRC'S HUMAN PERFORMANCE INVESTIGATION PROCESS (HPIP). PARADIES,M.; UNGER,L. System Improvements, Inc. October 1993. 25pp. 9312070146. SI-92-101. 77355:191.

The three volumes of this report detail a standard investigation process for use by Nuclear Regulatory Commission (NRC) personnel when investigating human performance related events at nuclear power plants. The process, called the Human Performance Investigation Process (HPIP), was developed to meet the special needs of NRC personnel, especially NRC resident and regional inspectors. HPIP is a systematic investigation process combining current procedures and field practices, expert experience, NRC human performance research, and applicable investigation techniques. The process is easy to learn and helps NRC personnel perform better field investigations of the root causes of human performance problems. The human performance data gathered through such investigations provides a better understanding of the human performance issues that cause events at nuclear power plants. Volume I is a concise description of the need for the human performance investigation process, the process' components, the methods used to develop the process, the methods proposed to test the process, and conclusions on the process' usefulness.

NUREG/CR-5455 V02: DEVELOPMENT OF THE NRC'S HUMAN PERFORMANCE INVESTIGATION PROCESS (HPIP). PARADIES,M.; UNGER,L. System Improvements, Inc. October 1993. 320pp. 9312070222. SI-92-101. 77360:001.

The three volumes of this report detail a standard investigation process for use by U.S. Nuclear Regulatory Commission (NRC) personnel when investigating human performance related events at nuclear power plants. The process, called the Human Performance Investigation Process (HPIP), was developed to meet the special needs of NRC personnel, especially NRC resident and regional inspectors. HPIP is a systematic investigation process combining current procedures and field practices, expert experience, NRC human performance research, and applicable investigation techniques. The process is easy to learn and helps NRC personnel perform better field investigations of the root causes of human performance problems. The human performance data gathered through such investigations provides a better understanding of the human performance issues that cause events at nuclear power plants. Volume II is a field manual for use by investigators when performing event investigations. Volume II includes the HPIP Procedure, the HPIP Modules, and Appendices that provide extensive documentation of each investigation technique.

NUREG/CR-5455 V03: DEVELOPMENT OF THE NRC'S HUMAN PERFORMANCE INVESTIGATION PROCESS (HPIP). PARADIES,M.; UNGER,L. System Improvements, Inc. October 1993. 110pp. 9312070101. SI-92-101. 77360:251.

The three volumes of this report detail a standard investigation process for use by Nuclear Regulatory Commission (NRC) personnel when investigating human performance related events at nuclear power plants. The process, called the Human Performance Investigation Process (HPIP), was developed to meet the special needs of NRC personnel especially NRC resident and regional inspectors. HPIP is a systematic investigation process combining current procedures and field practices, expert experience, NRC human performance research, and applicable investigation techniques. The process is easy to learn and helps NRC personnel perform better field investigations of the root causes of human performance problems. The human performance data gathered through such investigations provides a better understanding of the human performance issues that cause events at nuclear power plants. Volume III is a detailed documentation of the development effort and the pilot training program.

NUREG/CR-5471: ENHANCEMENTS TO DATA COLLECTION AND REPORTING OF SINGLE AND MULTIPLE FAILURE EVENTS. WHITEHEAD,D.W. Sandia National Laboratories. PAULA,H.M. JBF Associates, Inc. PARRY,G.W.; et al. NUS Corp. March 1993. 149pp. 9304190151. SAND89-2562. 74622:204.

This document presents recommendations on how the collection and documentation of failure events at nuclear power plants can be improved. These recommendations, if adopted, should enhance the reliability improvement and risk assessment programs that are dependent on such information. The report concentrates on how the recommendations should provide the information necessary to improve the parameter estimations for both independent and dependent events in a probabilistic risk assessment and alludes to the fact that this same information can be used to enhance other nuclear power plant activities. Several existing data bases are reviewed and areas where information is lacking, either because certain information is not required to be reported or because required information was simply not reported, are identified. Finally, data needs identified from recent PRAs are discussed.

NUREG/CR-5488: RISK-BASED INSPECTION GUIDE FOR THREE MILE ISLAND NUCLEAR STATION UNIT 1. HARRISON,D.G.; GORE,B.F.; VO,T.V.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1993. 132pp. 9303120073. PNL-7187. 74239:193.

The level one probabilistic risk assessment (PRA) for Three Mile Island Nuclear Station Unit 1 (TMI-1) has been analyzed to identify plant systems and components important to minimizing public risk, as measured by system contributions to the annual probability of core damage, and to identify the primary failure modes of these components. This report presents a series of tables, organized by system and prioritized by risk importance, which identify components associated with over 95% of the inspectable risk due to plant operation. The systems addressed, in descending order of importance, are: the Decay Heat Removal, High Pressure Injection, Decay Heat Cooling Water, AC Power, Nuclear Services Cooling Water, Main Steam, Emergency Feedwater, Reactor Coolant System Pressure Control, Intermediate Closed Cooling Water, Instrument Air, DC Power, and Engineered Safeguards Actuation Systems. This ranking is based on the Fussel-Vesely Importance Measure of risk importance, i.e., the fraction of the total annual probability of core damage which involves failures of the system of interest. Though not involved in the prevention of core damage and thus not ranked, containment protection systems are of fundamental importance in preventing and minimizing public risk due to a release of radionuclides, should core damage occur. Therefore, containment protection systems are included in this report, consisting of: the Reactor Building Isolation, Reactor Building Spray, and Reactor Building Emergency Cooling Systems.

NUREG/CR-5591 V01 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For April-September 1990. CORWIN,W.R. Oak Ridge National Laboratory. November 1993. 55pp. 9312070266. ORNL/TM-11568. 77352:175.

The primary goal of the Heavy-Section Steel Irradiation (HSSI) program is to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior, and in particular the fracture toughness properties, of typical pressure vessel steels as they relate to light-water reactor pressure-vessel integrity. The program includes the direct continuation of irradiation studies previously conducted within the Heavy-Section Steel Technology program augmented by enhanced examinations of the accompanying microstructural changes. Effects of specimen size, material chemistry, product form and microstructure, irradiation fluence, flux, temperature and spectrum, and post-irradiation annealing are being examined on a wide range of fracture properties. During this period detailed statistical analyses of the fracture data on K(Ic) shift of high-copper welds revealed greater shifts in fracture toughness than in Charpy transition temperatures. Testing of the duplex specimens from the second phase of the irradiated crack arrest testing on high-copper welds was initiated. Short-term aging studies were conducted on stainless steel weld-overlay cladding. Additional determinations were made of chemistry and unirradiated RT(NDT)s of the low upper-shelf weld metal from the Midland reactor and fracture toughness testing begun. An initial model describing the evolution of radiation-induced self-defect/solute clusters and other microstructures was developed and experiments initiated to examine the effects of low-energy, low-temperature neutron irradiations.

NUREG/CR-5631 R1 ADD: CONTRIBUTION OF MATERNAL RADIONUCLIDE BURDENS TO PRENATAL RADIATION DOSES.Relationships Between Annual Limits On Intake And Prenatal Doses. SIKOV,M.R.; HUI,T.E. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1993. 108pp. 9311080197. PNL-7745. 77076:001.

This addendum describes approaches for calculating and expressing radiation doses to the embryo/fetus from maternal intakes of radionuclides at levels corresponding to fractions or multiples of the Annual Limits on Intake (ALI). Information concerning metabolic or dosimetric characteristics and the placental transfer of selected, occupationally significant radionuclides was presented in NUREG/CR-5631, Revision 1. That information was used to estimate levels of radioactivity in the embryo/fetus as a function of stage of pregnancy and time after entry. Extension of MIRD methodology to accommodate gestational-stage-dependent characteristics allowed dose calculations for the simplified situation based on introduction of 1 μCi into the woman's transfer compartment (blood). The expanded scenarios in this addendum include repeated or chronic ingestion or inhalation intakes by a woman during pregnancy and body burdens at the beginning of pregnancy. Tables present dose equivalent to the embryo/fetus relative to intakes of these radionuclides in various chemical or physical forms and from pre-existing maternal burdens corresponding to ALI; complementary intake values (fraction of an ALI and μCi) that yield a dose equivalent of 0.05 rem are included. Similar tables give these measures of dose equivalency to the uterus from intakes of radionuclides for use as surrogates for embryo/fetus dose when biokinetic information is not available.

NUREG/CR-5642: LIGHT WATER REACTOR LOWER HEAD FAILURE ANALYSIS. REMPE,J.L.; CHAVEZ,S.A.; THINNES,G.L.; et al. EG&G Idaho, Inc. October 1993. 450pp. 9311080148. EGG-2618. 77082:018.

This document presents the results from a U.S. Nuclear Regulatory Commission-sponsored research program to investigate the mode and timing of vessel lower head failure. Major objectives of the analysis were to identify plausible failure mechanisms and to develop a method for determining which failure mode would occur first in different light water reactor designs and accident conditions. Failure mechanisms, such as tube ejection, tube rupture, global vessel failure, and localized vessel creep rupture, were studied. Newly developed models and existing models were applied to predict which failure mechanism would occur first in various severe accident scenarios. So that a broader range of conditions could be considered simultaneously, calculations relied heavily on models with closed-form or simplified numerical solution techniques. Finite element techniques were employed for analytical model verification and examining more detailed phenomena. High-temperature creep and tensile data were obtained for predicting vessel and penetration structural response.

NUREG/CR-5672 V03: CHARACTERISTICS OF LOW-LEVEL RADIOACTIVE DECONTAMINATION WASTE.Annual Report For Fiscal Year 1992. AKERS,D.W.; MCCONNELL,J.W.; MORCOS,N. EG&G Idaho, Inc. February 1993. 68pp. 9303120059. EGG-2635. 74241:207.

This document addresses the work performed during fiscal year 1992 at the Idaho National Engineering Laboratory by the Low-Level Radioactive Waste-Decontamination Waste Program (FIN A6359), which is funded by the U.S. Nuclear Regulatory Commission. The program evaluates the physical stability and leachability of solidified waste streams generated in the decontamination process of primary coolant systems in operating nuclear power stations. The data in this document include the chemical composition and characterization of waste streams from Peach Bottom Atomic Power Station Unit 3 and from Nine Mile Point Nuclear Plant Unit 1. The results of compressive strength testing on immersed and unimmersed solidified waste-form specimens from Peach Bottom, and the results of leachate analysis are addressed. Cumulative fractional release rates and leachability indexes of those specimens were calculated and are included in this report.

NUREG/CR-5699 V01: AGING AND SERVICE WEAR OF CONTROL ROD DRIVE MECHANISMS FOR BWR NUCLEAR PLANTS. GREENE,R.H. Oak Ridge National Laboratory. November 1992. 149pp. 9303110370. ORNL-6666. 74212:090.

This Phase I Nuclear Plant Aging Research (NPAR) study examines the aging phenomena associated with BWR control rod drive mechanisms (CRDMs) and assesses the merits of various methods of "managing" this aging. Information for this study was acquired from: (1) the results of a special CRDM aging questionnaire distributed to each U.S. BWR utility; (2) a first-of-its-kind workshop held to discuss CRDM aging and maintenance concerns; (3) an analysis of the Nuclear Plant Reliability Data System (NPRDS) failure cases attributed to the control rod drive (CRD) system; and (4) personal information exchange with nuclear industry CRDM maintenance experts. Nearly 23% of the NPRDS CRD system component failure reports were attributed to the CRDM. The CRDM components most often requiring replacement due to normal wear and aging are the Graphitar seals. The predominant causes of aging for these seals are mechanical wear and thermally induced embrittlement. More than 59% of the NPRDS CRD system failure reports were attributed to components that comprise the hydraulic control unit (HCU). The predominant HCU components experiencing the effects of service wear and aging are valve seals, discs, seats, stems, packing, and diaphragms.

NUREG/CR-5747: ESTIMATE OF RADIONUCLIDE RELEASE CHARACTERISTICS INTO CONTAINMENT UNDER SEVERE ACCIDENT CONDITIONS.Final Report. NOURBAKHSH,H.P. Brookhaven National Laboratory. November 1993. 142pp. 9401030188. BNL-NUREG-52289. 77652:001.

A detailed review of the available light water reactor source term information is presented as a technical basis for development of updated source terms into the containment under severe accident conditions. Simplified estimates of radionuclide release and transport characteristics are specified for each unique combination of the reactor coolant and containment system combinations. A quantitative uncertainty analysis in the release to the containment using NUREG-1150 methodology is also presented.

NUREG/CR-5754: BOILING-WATER REACTOR INTERNALS AGING DEGRADATION STUDY. Phase 1. LUK,K.H. Oak Ridge National Laboratory. September 1993. 56pp. 9310120363. ORNL/TM-11876. 76740:333.

This report documents the results of an aging assessment study for boiling water reactor (BWR) internals. Major stressors for BWR internals are related to unsteady hydrodynamic forces generated by the primary coolant flow in the reactor vessel. Welding and cold-working, dissolved oxygen and impurities in the coolant, applied loads and exposures to fast neutron fluxes are other important stressors. Based on results of a component failure information survey, stress corrosion cracking (SCC) and fatigue are identified as the two major aging-related degradation mechanisms for BWR internals. Significant reported failures include SCC in jet-pump holddown beams, in-core neutron flux monitor dry tubes and core spray spargers. Fatigue failures were detected in feedwater spargers. The implementation of a plant Hydrogen Water Chemistry (HWC) program is considered as a promising method for controlling SCC problems in BWR. More operating data are needed to evaluate its effectiveness for internal components. Long-term fast neutron irradiation effects and high-cycle fatigue in a corrosive environment are uncertainty factors in the aging assessment process. BWR internals are examined by visual inspections and the method is access limited. The presence of a large water gap and an absence of ex-core neutron flux monitors may handicap the use of advanced inspection methods, such as neutron noise vibration measurements, for BWR.

NUREG/CR-5755: STIFFNESS OF LOW-ASPECT RATIO, REINFORCED CONCRETE SHEAR WALLS. FARRAR,C.R. Los Alamos National Laboratory. BAKER,W.E. New Mexico, Univ. of, Albuquerque, NM. January 1993. 149pp. 9302020466. LA-12181-MS. 64729:054.

This report summarizes the information relating to stiffness of low-aspect-ratio, reinforced concrete shear walls that has been obtained from static and dynamic tests of scale-model Seismic Category 1 structures (exclusive of containment) and structural elements. Although numerous static and dynamic tests of shear wall elements are reported in the technical literature, most of these were ultimate strength tests. When these tests are examined to determine stiffness values, there is a considerable range in the results obtained. The types of structures and structural elements tested, the test procedures, the methods used to measure stiffness (both directly and indirectly), and a summary of the results are discussed. This report concludes by showing the changes in stiffness of shear walls as a function of the peak nominal base shear stress that the structure experiences during a seismic event.

NUREG/CR-5758 V03: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports,CY 1992. FLEMING,T.; WESTRA,C.; FIELD,I.; et al. Battelle Human Affairs Research Centers. July 1993. 140pp. 9308180128. PNL-8688. 76142:200.

This report summarizes the data from the semiannual reports on fitness-for-duty programs submitted to the NRC by 54 utilities for two reporting periods: January 1, 1992 to June 30, 1992, and from July 1, 1992 to December 31, 1992. During CY 1992, licensees reported that they conducted 266,551 tests for the presence of illegal drugs and alcohol. Of these tests, 1,818 (.68%) were confirmed positive. Positive test results varied by category of test and category of worker. The majority of positive test results (1,110) were obtained through pre-access testing. Of tests conducted on workers having access to the protected area, there were 461 positive tests from random testing, and 178 positive tests from for-cause testing. Followup testing of workers who had previously tested positive resulted in 69 positive tests. Positive test results also varied by category of worker. Overall, short-term and long-term contractor personnel had the highest rates of positive tests. Licensee employees had lower rates of positive test results.

NUREG/CR-5759: RISK ANALYSIS OF HIGHLY COMBUSTIBLE GAS STORAGE, SUPPLY, AND DISTRIBUTION SYSTEMS IN PRESSURIZED WATER REACTOR PLANTS. SIMION,G.P. Science Applications International Corp. (formerly Science Applications, Inc.). VANHORN,R.L.; SMITH,C.L.; et al. EG&G Idaho, Inc. June 1993. 250pp. 9306290067. EGG-2640. 75495:243.

This report presents the evaluation of the potential safety concerns for pressurized water reactors (PWRs) identified in Generic Safety Issue 106, Piping and the Use of Highly Combustible Gases in Vital Areas. A Westinghouse four-loop PWR plant was analyzed for the risk due to the use of combustible gases (predominantly hydrogen) within the plant. The analysis evaluated an actual hydrogen distribution configuration and conducted several sensitivity studies to determine the potential variability among PWRs. The sensitivity studies were based on hydrogen and safety-related equipment configurations observed at other PWRs within the United States. Several options for improving the hydrogen distribution system design were identified and evaluated for their effect on risk and core damage frequency. A cost/benefit analysis was performed to determine whether alternatives considered were justifiable based on the safety improvement and economics of each possible improvement.

NUREG/CR-5766: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE SAN ONOFRE UNIT 2 NUCLEAR POWER PLANT. PUGH,R.; GORE,B.F.; VO,T.V.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1993. 36pp. 9302230239. PNL-7609. 64963:001.

In a study sponsored by the U.S. Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory has developed and applied a methodology for deriving plant-specific risk-based inspection guidance for the auxiliary feedwater (AFW) system at pressurized water reactors that have not undergone probabilistic risk assessment (PRA). This methodology uses existing PRA results and plant operating experience information. Existing PRA-based inspection guidance information recently developed for the NRC for various plants was used to identify generic component failure modes. This information was then combined with plant-specific and industry-wide component information and failure data to identify failure modes and failure mechanisms for the AFW system at the selected plants. San Onofre-2 was selected as one of a series of plants for study. The product of this effort is a prioritized listing of AFW failures which have occurred at the plant and at other PWRs. This listing is intended for use by NRC inspectors in the preparation of inspection plans addressing AFW risk-important components at the San Onofre-2 plant.

NUREG/CR-5776: DAMPING IN LOW-ASPECT-RATIO,REINFORCED CONCRETE SHEAR WALLS. FARRAR,C.R. Los Alamos National Laboratory. BAKER,W.E. New Mexico, Univ. of, Albuquerque, NM. May 1993. 83pp. 9306210337. LA-12201-MS. 75403:242.

This report summarizes the information obtained from static and dynamic tests of scale-model Seismic Category 1 structures (exclusive of containment) on the damping of low-aspect-ratio, reinforced concrete shear walls. The report reviews experimental assessments of damping in low-aspect-ratio shear walls that have been reported in the literature and presents a summary of the types of structures and structural elements tested. It discusses the testing methods and the methods used to determine equivalent viscous damping ratios (both directly and indirectly), a numerical study that examines the accuracy of various methods for estimating damping from measured acceleration input and response data, and tabulates the damping results. The report concludes by graphically showing the changes in the damping of the shear walls as a function of the peak nominal base shear stress experienced by the structure during simulated seismic events. Also included are comparisons of the damping results obtained in this program with those obtained by other investigators.

NUREG/CR-5778 V03: NEW YORK/NEW JERSEY REGIONAL SEISMIC NETWORK.Final Report For April 1985 - September 1992. SEEBER,L.; JOHNSON,D.; ARMBRUSTER,J. Lamont-Doherty Geological Observatory. July 1993. 74pp. 9307270035. 75802:317.

For almost 20 years, Lamont-Doherty Earth Observatory has operated the primary network for monitoring earthquake activity in the New York State, northern New Jersey, and northwestern Vermont area, with support by both NRC and the USGS. The primary purpose of this research is directed toward the determination of local seismicity and the possible identification of associated geologic and tectonic features. From April 1985 to September 1992, the network recorded and located 346 regional earthquakes. Scientific activity, primarily in the form of aftershock monitoring, was concentrated upon a number of significant earthquakes: The Ardsley, NY; the Chardon, OH; the Ashtabula, OH; and the Saguenay, Canada earthquakes; in addition to the Summit, NY event. These studies involved the deployment of portable seismographs in the epicentral areas. Many of these sequences were in northeastern North America, but outside the L-DEO seismic network and were not covered by other permanent networks. Spatial correlations between structures and earthquakes were found at a wide range of scales, and systematic searches of archival material were used to improve constraints on historic sources.

NUREG/CR-5782: PRESSURIZED THERMAL SHOCK PROBABILISTIC FRACTURE MECHANICS SENSITIVITY ANALYSIS FOR YANKEE ROWE REACTOR PRESSURE VESSEL. DICKSON,T.L.; CHEVERTON,R.D.; BRYSON,J.W.; et al. Oak Ridge National Laboratory. August 1993. 114pp. 9309210205. ORNL/TM-11945. 76484:336.

The Nuclear Regulatory Commission (NRC) requested Oak Ridge National Laboratory (ORNL) to perform a pressurized-thermal-shock (PTS) probabilistic fracture mechanics (PFM) sensitivity analysis for the Yankee Rowe reactor pressure vessel, for the fluences corresponding to the end of operating cycle 22, using a specific small-break-loss-of-coolant transient as the loading condition. Regions of the vessel with distinguishing features were to be treated individually--upper axial weld, lower axial weld, circumferential weld, upper plate spot welds, upper plate regions between the spot welds, lower plate spot welds, and the lower plate regions between the spot welds. The fracture analysis methods used in the analysis of through-clad surface flaws were those contained in the established OCA-P computer code, which was developed during the Integrated Pressurized Thermal Shock (IPTS) Program. The NRC request specified that the OCA-P code be enhanced for this study to also calculate the conditional probabilities of failure for subclad flaws and embedded flaws. The results of this sensitivity analysis provide the NRC with (1) data that could be used to assess the relative influence of a number of key input parameters in the Yankee Rowe PTS analysis and (2) data that can be used for readily determining the probability of vessel failure once a more accurate indication of vessel embrittlement becomes available. This report is designated as HSST report No. 117.

NUREG/CR-5783: AGING ASSESSMENT OF THE COMBUSTION ENGINEERING AND BABCOCK & WILCOX CONTROL ROD DRIVES. GROVE,E.; GUNTHER,W. Brookhaven National Laboratory. January 1993. 219pp. 9302230172. BNL-NUREG-52299. 64978:001.

The effects of aging upon the Babcock & Wilcox (B&W) and Combustion Engineering (CE) control rod drive systems have been evaluated. For this study, the CRD system boundary included the control rod assemblies, guide tubes, control rod drive mechanism, control system components, rod position indication components, and cooling system. Detailed operation experience data for 1980 to 1990 was evaluated to identify the predominant failure modes, causes, and effects. The results of this evaluation, along with an assessment of component material and operating environment, lead to the conclusion that both the B&W and CE CRD systems are susceptible to age degradation. Failures of the CRD system have resulted in significant plant effects including power reductions, plant shutdowns, scrams, and ESF actuations. Information on current plant system inspection and maintenance practices were obtained from two B&W plants, and four CE plants through an industry survey. The results of this survey indicate that some plants have modified the system, replaced components, and established preventive maintenance programs, some of which effectively address the aging issue, while others do not. The potential application of some advanced monitoring inspection techniques are discussed.

NUREG/CR-5791: RISK EVALUATION FOR A GENERAL ELECTRIC BWR, EFFECTS OF FIRE PROTECTION SYSTEM ACTUATION ON SAFETY-RELATED EQUIPMENT. Evaluation Of Generic Issue 57. LAMBRIGHT,J.A.; ROSS,S. Sandia National Laboratories. KLAMERUS,E.; et al. Science & Engineering Associates, Inc. December 1992. 312pp. 9301220181. SAND91-1536. 64652:001.

Nuclear power plants have experienced actuations of fire protection systems (FPS) under conditions for which these systems were not intended to actuate, and also have experienced actuations with the presence of a fire. These actuations have often damaged nearby plant equipment. A review of past occurrences of both types of such events on nuclear power plant safety, and a cost-benefit analysis of potential corrective measures has been performed. Thirteen different scenarios leading to actuation of fire protection systems due to a variety of causes were identified. A quantification of these thirteen scenarios, where applicable, was performed on a BWR4/MKI. This report estimates the contribution of FPS actuations to core damage frequency, proposes physical modifications to reduce the risk from the dominant contributors, and estimates the values and impacts of the proposed modifications.

NUREG/CR-5801: PROCEDURE FOR ANALYSIS OF COMMON-CAUSE FAILURES IN PROBABILISTIC SAFETY ANALYSIS. MOSLEH,A. Maryland, Univ. of, College Park, MD. * Sandia National Laboratories. April 1993. 51pp. 9306010342. SAND91-7087. 75062:001.

This report provides practical guidelines for treatment of common-cause failures (CCF) in risk and reliability studies. The procedures outlined in this report are organized according to three phases of analysis, screening analysis, detailed qualitative analysis, and detailed quantitative analysis. The results of the screening analysis phase include conservative identification of potential common-cause vulnerabilities and determination of the scope and focus for more detailed analysis in Phases II and III. Phase II, the detailed qualitative analysis, provides a better understanding of the plant-specific susceptibilities of the systems and components to causes and coupling mechanisms of CCF. The information from this phase can then be used as a basis for a plant-specific quantitative assessment of CCF frequencies. Detailed guidelines are provided for Phase III to aid the analyst in using this qualitative information and generic data in developing a plant-specific CCF base. Depending on the overall objective of the study, CCF analysis can stop at the end of any of the three phases.

NUREG/CR-5817 V02: NRC HIGH-LEVEL RADIOACTIVE WASTE RESEARCH AT CNWRA.Calendar Year 1991. ABABOU,R.; BAGTZOGLOU,A.C.; CHOWDHURY,A.H.; et al. Center for Nuclear Waste Regulatory Analyses. May 1993. 500pp. 9306290022. CNWRA 91-01A. 75494:144.

This is an annual status report on the results of research conducted on behalf of the U.S. NRC by the Center for Nuclear Waste Regulatory Analyses in support of activities under the Nuclear Waste Policy Act, as amended. Nine specific projects are underway; eight of which are reported here. The Geochemistry project is using laboratory methods and computer calculations to assess key geochemical constraints and to evaluate sorptive properties of zeolites present at the proposed repository site. The Thermohydrology project has as its focus improved understanding of heat and fluid flow in unsaturated media. Laboratory, field, and calculational studies are combined in the Seismic Rock Mechanics project to examine the effects of repeated seismic loadings on the rock-mechanical and hydrological responses of rock masses. The Integrated Waste Package Experiments have been initiated to evaluate degradation modes of candidate waste container alloys. Three-dimensional computer analysis techniques are being used to investigate spatial variability of flow and transport in variably saturated fractured porous media in the Stochastic Flow and Transport project. The recently initiated Geochemical Analogs project seeks to investigate the role of such analogs in the licensing process, and is currently focused on characterizing and evaluating a potential site for investigation. The Sorption Modeling Project has as its objective the evaluation and eventual selection of model(s) of sorption processes which are deemed technically acceptable in the context of repository licensing. Finally, the Performance Assessment project is directed toward developing and evaluating methodologies for evaluation of the long-term performance of the proposed repository.

NUREG/CR-5817 V03 N1: NRC HIGH-LEVEL RADIOACTIVE WASTE RESEARCH AT CNWRA. January-June 1992. ABABOU,R.; AHOLA,M.P.; BACA,R.G.; et al. Center for Nuclear Waste Regulatory Analyses. May 1993. 105pp. 9306210312. CNWRA 92-01S. 75404:022.

This is a semi-annual status report on the results of research conducted on behalf of the U.S. Nuclear Regulatory Commission by the Center for Nuclear Waste Regulatory Analyses in support of activities under the Nuclear Waste Policy Act, as amended. Nine specific projects are under way as reported here. The Geochemistry Project staff is using laboratory methods and computer calculations to assess key geochemical constraints and to evaluate sorptive properties of zeolites present at the proposed repository site. The Thermohydrology Project has as its focus improved understanding of heat and fluid flow in unsaturated media. Laboratory, field, and calculational studies are combined in the Seismic Rock Mechanics Project to examine the effects of repeated seismic loadings on the rock-mechanical and hydrological responses of rock masses. The Integrated Waste Package Experiments have been initiated to evaluate degradation modes of candidate waste container alloys. Three-dimensional computer analyses techniques are being used to investigate spatial variability of flow and transport in variably saturated fractured porous media in the Stochastic Flow and Transport Project. The Geochemical Analogs Project staff seeks to investigate the role of such analogs in the licensing process, and is currently focused on characterizing and evaluating two potential sites for investigation. The Sorption Modeling Project has as its objective the evaluation and eventual selection of model(s) of sorption processes which are deemed technically acceptable in the context of repository licensing. The Performance Assessment Project is directed toward developing and evaluating methodologies for evaluation of the long-term performance of the proposed repository. Finally, in the recently initiated Volcanism Project, a comprehensive evaluation of the state of knowledge regarding volcanism in the basin and range province has been completed.

NUREG/CR-5817 V03 N2: NRC HIGH-LEVEL RADIOACTIVE WASTE RESEARCH AT CNWRA.July-December 1992. ABABOU,R.; AHOLA,M.P.; BACA,R.G.; et al. Center for Nuclear Waste Regulatory Analyses. July 1993. 200pp. 9309090005. CNWRA 92-02S. 76393:001.

Progress from July 1 to December 31, 1992 on the nine NRC-sponsored research projects conducted at the Center for Nuclear Waste Regulatory Analyses is described. Ion-exchange experiments between clinoptilolite and aqueous solutions of Na(+) and Sr(2+) and three applications of reaction-path modeling are described in the Unsaturated Mass Transport (Geochemistry) project. Numerical simulation of a laboratory-scale non-isothermal two-phase flow is discussed in the Thermohydrology chapter. Methods for estimating rock joint roughness coefficient are the focus of the Seismic Rock Mechanics project for which the Tilt Test, Tse and Cruden's equations, and fractal-based equations were tested and found to be unsatisfactory. In the Integrated Waste Package Experiments chapter, investigations of pit initiation and repassivation potential for alloys 825 and C-22 and stainless steel 304L and 316L are described. Testing of the BIGFLOW computer code and visualization of fracture topology is the theme of the Stochastic Hydrology project. Preliminary analysis of field data from the Akrotiri site in Greece is developed in the Geochemical Analogs project. Mechanistic modeling of sorption using the MINTEQA2 code is investigated as part of the Sorption project. Adaptive gridding and "modified equations" methods for solving the flow and transport equations are described in the Performance Assessment chapter. Finally, the Volcanism chapter focuses on using nonhomogeneous Poisson processes for estimating probability of volcanic events at the potential repository site.

NUREG/CR-5818: UNCERTAINTY ANALYSIS OF MINIMUM VESSEL LIQUID INVENTORY DURING A SMALL-BREAK LOCA IN A B&W PLANT-AN APPLICATION OF THE CSAU METHODOLOGY USING THE RELAP5/MOD3 COMPUTER CODE. ORTIZ,M.G.; GHAN,L.S. EG&G Idaho, Inc. December 1992. 89pp. 9302230168. EGG-2665. 64979:001.

The Nuclear Regulatory Commission (NRC) revised the emergency core cooling system licensing rule to allow the use of best estimate computer codes, provided the uncertainty of the calculations are quantified and used in the licensing and regulation process. The NRC developed a generic methodology called Code Scaling, Applicability, and Uncertainty (CSAU) to evaluate best estimate code uncertainties. The objective of this work was to adapt and demonstrate the CSAU methodology for a small-break loss-of-coolant accident (SBLOCA) in a Pressurized Water Reactor of Babcock & Wilcox Company lowered loop design using RELAP5/MOD3 as the simulation tool. The CSAU methodology was successfully demonstrated for the new set of variants defined in this project (scenario, plant design, code). However, the robustness of the reactor design to this SBLOCA scenario limits the applicability of the specific results to other plants or scenarios. Several aspects of the code were not exercised because the conditions of the transient never reached enough severity. The plant operator proved to be a determining factor in the course of the transient scenario, and steps were taken to include the operator in the model, simulation, and analyses.

NUREG/CR-5822: ANALYSIS OF THERMAL MIXING AND BORON DILUTION IN A PWR. SUN,J.G.; SHA,W.T. Argonne National Laboratory. February 1993. 94pp. 9303250056. ANL-91/43. 74358:154.

Thermal mixing and boron dilution in a pressurized water reactor were analyzed with COMMIX codes. The reactor system was a four-loop Zion reactor that was initially filled with hot boron-rich water. It was assumed that the reactor coolant pumps are tripped. Following the trip, cold unborated water from seal injection or other sources continuously flows into the reactor coolant system and dilution takes place first in the pump suction line and then in the reactor vessel. The thermal mixing and boron dilution under these conditions were analyzed. For the analysis of thermal mixing, water at room temperature (referred to as cold water) was fed into the cold leg of the reactor system at various flow rates. For the analysis of boron dilution, cold and hot unborated water was fed into the cold leg at a high flow rate. The subsequent transient thermal mixing and boron dilution that would occur in the reactor system were simulated for 1-2 h, depending on the flow rate. A third analysis was performed for the boron dilution after the start of the reactor coolant pump, which forces a slug of cold unborated water from the pump suction line into the reactor vessel. This transient was simulated for 20 sec, by which time the slug has been pushed out of the reactor core. The rates of reactivity insertion were evaluated for these analyses.

NUREG/CR-5829: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDANCE FOR THE DAVIS-BESSE NUCLEAR POWER PLANT. NICKOLAUS,J.R.; MOFFITT,N.E.; GORE,B.F.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1993. 38pp. 9310120332. PNL-7905. 76742:139.

In a study sponsored by the U.S. Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory has developed and applied a methodology for deriving plant-specific risk-based inspection guidance for the auxiliary feedwater (AFW) system at pressurized water reactors that have not undergone probabilistic risk assessment (PRA). This methodology uses existing PRA results and plant operating experience information. Existing PRA-based inspection guidance information recently developed for the NRC for various plants was used to identify generic component failure modes. This information was then combined with plant-specific and industry-wide component information and failure data to identify failure modes and failure mechanisms for the AFW system at the selected plants. Davis-Besse was selected as one of a series of plants for study. The product of this effort is a prioritized listing of AFW failures which have occurred at the plant and at other PWRs. This listing is intended for use by NRC inspectors in the preparation of inspection plans addressing AFW risk-important components at the Davis- Besse plant.

NUREG/CR-5833: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE H.B. ROBINSON NUCLEAR POWER PLANT. MOFFITT,N.E.; LLOYD,R.C.; GORE,B.F.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1993. 37pp. 9308160303. PNL-7907. 76119:293.

In a study sponsored by the U.S. Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory has developed and applied a methodology for deriving plant-specific risk-based inspection guidance for the auxiliary feedwater (AFW) system at pressurized water reactors that have not undergone probabilistic risk assessment (PRA). This methodology uses existing PRA results and plant operating experience information. Existing PRA-based inspection guidance information recently developed for the NRC for various plants was used to identify generic component failure modes. This information was then combined with plant-specific and industry-wide component information and failure data to identify failure modes and failure mechanisms for the AFW system at the selected plants. H. B. Robinson was selected as one of a series of plants for study. The product of this effort is a prioritized listing of AFW failures which have occurred at the plant and at other PWRs. This listing is intended for use by NRC inspectors in the preparation of inspection plans addressing AFW risk-important components at the H. B. Robinson plant.

NUREG/CR-5834: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE FORT CALHOUN NUCLEAR POWER PLANT. MOFFITT,N.E.; GORE,B.F.; VEHEC,T.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1993. 34pp. 9303120001. PNL-7906. 74235:324.

In a study sponsored by the U.S. Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory has developed and applied a methodology for deriving plant-specific risk-based inspection guidance for the auxiliary feedwater (AFW) system at pressurized water reactors that have not undergone probabilistic risk assessment (PRA). This methodology uses existing PRA results and plant operating experience information. Existing PRA-based inspection guidance information recently developed for the NRC for various plants was used to identify generic component failure modes. This information was then combined with plant-specific and industry-wide component information and failure data to identify failure modes and failure mechanisms for the AFW system at the selected plants. Fort Calhoun was selected as the sixth plant for study. The product of this effort is a prioritized listing of AFW failures which have occurred at the plant and at other PWRs. This listing is intended for use by NRC inspectors in the preparation of inspection plans addressing AFW risk-important components at the Fort Calhoun plant.

NUREG/CR-5835: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE BEAVER VALLEY,UNITS 1 AND 2 NUCLEAR POWER PLANTS. LLOYD,R.C.; VEHEC,T.A.; MOFFITT,N.E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1993. 40pp. 9303120101. PNL-7925. 74236:271.

In a study sponsored by the U.S. Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory has developed and applied a methodology for deriving plant-specific risk-based inspection guidance for the auxiliary feedwater (AFW) system at pressurized water reactors that have not undergone probabilistic risk assessment (PRA). This methodology uses existing PRA results and plant operating experience information. Existing PRA-based inspection guidance information recently developed for the NRC for various plants was used to identify generic component failure modes. This information was then combined with plant-specific and industry-wide component information and failure data to identify failure modes and failure mechanisms for the AFW system at the selected plants. Beaver Valley Units 1 and 2 were selected as two of a series of plants for study. The product of this effort is a prioritized listing of AFW failures which have occurred at the plant and at other PWRs. This listing is intended for use by NRC inspectors in the preparation of inspection plans addressing AFW risk-important components at Beaver Valley Units 1 and 2.

NUREG/CR-5836: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE PALO VERDE NUCLEAR POWER PLANT. BUMGARDNER,J.D.; MOFFITT,N.E.; GORE,B.F.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1993. 34pp. 9302230328. PNL-7908. 64984:128.

In a study sponsored by the U.S. Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory has developed and applied a methodology for deriving plant-specific risk-based inspection guidance for the auxiliary feedwater (AFW) system at pressurized water reactors that have not undergone probabilistic risk assessment (PRA). This methodology uses existing PRA results and plant operating experience information. Existing PRA-based inspection guidance information recently developed for the NRC for various plants was used to identify generic component failure modes. This information was then combined with plant-specific and industry-wide component information and failure data to identify failure modes and failure mechanisms for the AFW system at the selected plants. Palo Verde was selected as one of a series of plants for study. The product of this effort is a prioritized listing of AFW failures which have occurred at the plant and at other PWRs. This listing is intended for use by NRC inspectors in the preparation of inspection plans addressing AFW risk-important components at the Palo Verde plants.

NUREG/CR-5843: CORCON-MOD3:AN INTEGRATED COMPUTER MODEL FOR ANALYSIS OF MOLTEN CORE-CONCRETE INTERACTIONS.User's Manual. BRADLEY,D.R.; GARDNER,D.R.; BROCKMANN,J.E.; et al. Sandia National Laboratories. October 1993. 278pp. 9312070049. SAND92-0167. 77353:001.

The CORCON-Mod3 computer code was developed to mechanistically model the important core-concrete interaction phenomena, including those phenomena relevant to the assessment of containment failure and radionuclide release. The code can be applied to a wide range of severe accident scenarios and reactor plants. The code represents the current state of the art for simulating core debris interactions with concrete. This document comprises the user's manual and gives a brief description of the models and the assumptions and limitations in the code. Also discussed are the input parameters and the code output. Two sample problems are also given.

NUREG/CR-5844: AGING ASSESSMENT OF BISTABLES AND SWITCHES IN NUCLEAR POWER PLANTS. LEE,B.S.; VILLARAN,M.; SUBUDHI,M. Brookhaven National Laboratory. January 1993. 213pp. 9302230104. BNL-NUREG-52318. 64979:119.

Bistables and process switches play vital roles in the instrumentation and control logic of a nuclear power station. To understand the aging characteristics of these components, more than 5,000 NPRDS events and more than 1200 LERs were reviewed and analyzed. Telephone surveys were conducted and nuclear plant site visits were made to collect information on operating experience and the current status of these devices. Interaction with the equipment manufacturers provided further details on the designs of bistables and switches. The aging characteristics studied included the effects of aging on failure frequency, failure mode, and failure cause. This study found that two groups of bistables are in operation. The first group which consists of older bistables, needs attention in the near future concerning the decision on replacement or refurbishment; however, the second group, which is newer, still has time to manage aging concerns. Most of the original switches employed in reactor protection systems have been replaced with transmitters and bistables, and the trend shows that the remaining ones will be replaced in the near future. Based on the results of the analyses, recommendations for better aging management are made, and the areas for future studies are identified.

NUREG/CR-5851: LONG TERM PERFORMANCE AND AGING CHARACTERISTICS OF NUCLEAR PLANT PRESSURE TRANSMITTERS. HASHEMIAN,H.M.; MITCHELL,D.W.; FAIN,R.E.; et al. Analysis & Measurement Services Corp. March 1993. 375pp. 9304020285. 74450:001.

This report presents the results of a comprehensive research and development project conducted for the NRC to study the effects of normal aging on calibration and response time of nuclear plant pressure, level, and flow transmitters and to develop and validate new methods for testing the performance of the transmitters as installed in nuclear power plants. The project involved research in seven areas as follows: 1) Aging tests of complete transmitter assemblies; 2) Aging tests of critical components of transmitters; 3) Testing the effects of sensing line length, blockages, and voids on the response time of pressure sensing systems; 4) Oil loss phenomenon in Rosemount and other transmitters; 5) Validation of new methods for on-line testing of response times of pressure transmitters; 6) On-line detection of oil loss in Rosemount transmitters; and 7) Analysis of Licensee Event Report (LER) and Nuclear Plant Reliability Data System (NPRDS) databases for failures of pressure sensing systems in nuclear power plants.

NUREG/CR-5863: RISK ASSESSMENT OF ISOLATION DEVICES IN SAFETY SYSTEMS. CRAMOND,W.R.; MITCHELL,D.B. Sandia National Laboratories. MILLER,S.P.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). January 1993. 196pp. 9302230219. SAND92-0538. 64963:037.

Electronic isolators are used to maintain electrical separation between safety and non-safety systems in nuclear power plants. The concern is that these devices may fail allowing unwanted signals or energy to act upon safety systems, or preventing desired signals from performing their intended function. While operational history shows many isolation device problems requiring adjustments and maintenance, we could not find incidents where there was a safety implication. Even hypothesizing multiple simultaneous failures did not lead to significant contributions to core damage frequency. Although the analyses performed in this study were not extensive or detailed, there seems to be no evidence to suspect that isolation device failure is an issue which should be studied further.

NUREG/CR-5882: TRAC-B THERMAL-HYDRAULIC ANALYSIS OF THE BLACK FOX BOILING WATER REACTOR. MARTIN,R.P. EG&G Idaho, Inc. May 1993. 68pp. 9306180322. EGG-2677. 75387:169.

Thermal-hydraulic analyses of six hypothetical accident scenarios for the General Electric Black Fox Nuclear Project boiling water reactor were performed using the TRAC-BF1 computer code. This work is sponsored by the U.S Nuclear Regulatory Commission and is being done in conjunction with future analysis work at the U.S. Nuclear Regulatory Commission Technical Training Center in Chattanooga, Tennessee. These accident scenarios were chosen to assess and benchmark the thermal-hydraulic capabilities of the Black Fox Nuclear Project simulator at the Technical Training Center to model abnormal transient conditions.

NUREG/CR-5883: HEALTH RISK ASSESSMENT OF IRRADIATED TOPAZ. NELSON,K.; BAUM,J.W. Brookhaven National Laboratory. January 1993. 156pp. 9302220410. BNL-NUREG-52330. 64901:001.

Irradiated topaz gemstones are currently processed for color improvement by subjecting clear stones to neutron or high-energy electron irradiations, which leads to activation of trace elements in the stones. Assessment of the risk to consumers required the identification and quantification of the resultant radionuclides and the attendant exposure. Representative stones from Brazil, India, Nigeria, and Sri Lanka were irradiated and analyzed for gamma ray and beta particle emissions, using sodium iodide and germanium spectrometers; and Geiger-Muller, plastic and liquid scintillation, autoradiography, and thermoluminescent-dosimetry measurement techniques. Based on these studies and other information derived from published literature, dose and related risk estimates were made for typical user conditions. New criteria and methods for routine assays for acceptable release, based on gross beta and gross photon emissions from the stones, were also developed.

NUREG/CR-5884 V1 DRF: REVISED ANALYSES OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION.Effects Of Current Regulatory And Other Considerations On The Financial Assurance....Main Report.Draft Report For Comment. KONZEK,G.J.; SMITH,R.I.; BIERSCHBACH,M.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1993. 200pp. 9311080246. PNL-8742. 77078:182.

With the issuance of the Decommissioning Rule (June 27, 1988), nuclear power plant licensees are required to submit to the U.S. Nuclear Regulatory Commission (NRC) for review, decommissioning plans and cost estimates. This reevaluation study provides some of the needed bases documentation to the NRC staff that will assist them in assessing the adequacy of the licensee submittals. This report presents the results of a review and reevaluation of the PNL 1978 decommissioning study of the Trojan nuclear power plant for the DECON, SAFSTOR, and ENTOMB decommissioning alternatives. These alternatives now include an initial 5-7 year period during which the spent fuel is stored in the spent fuel pool, prior to beginning major disassembly or extended safe storage of the plant. This report also includes consideration of the NRC requirement that decommissioning activities leading to termination of the nuclear license be completed within 60 years of final reactor shutdown, consideration of packaging and disposal requirements for Greater-Than-Class C low-level waste, and reflects all costs in 1993 dollars. Sensitivity of the total license termination cost to the disposal costs at different low-level radioactive waste disposal sites, and to different depths of contaminated concrete surface removal within the facilities are also examined.

NUREG/CR-5884 V2 DRF: REVISED ANALYSES OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION.Effects Of Current Regulatory And Other Considerations On The Financial Assurance....Appendices.Draft Report For Comment. KONZEK,G.J.; SMITH,R.I.; BIERSCHBACH,M.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1993. 400pp. 9311080238. PNL-8742. 77081:001.

See NUREG/CR-5884,V01,DRF abstract.

NUREG/CR-5894: RADIONUCLIDE CHARACTERIZATION OF REACTOR DECOMMISSIONING WASTE AND NEUTRON-ACTIVATED METALS. ROBERTSON,D.E.; THOMAS,C.W.; WYNHOFF,N.L.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. June 1993. 80pp. 9307060129. PNL-8106. 75572:213.

This study is providing the NRC and licensees with a more comprehensive and defensible data base and regulatory assessment of the radiological factors associated with reactor decommissioning and disposal of wastes generated during these activities. The objectives of this study are being accomplished during a two-phase sampling, measurement, and assessment program involving the actual decommissioning of Shippingport Station and the detailed analysis of neutron-activated materials from commercial reactors. The radiological characterization studies of Shippingport decommissioning materials have now been completed, and analyses of dismantled piping and scabbled concrete have shown that neutron activation products, dominated by (60)Co, comprised the residual radionuclide inventory. Fission products and transuranic radionuclides were essentially absent. Waste classification assessments have shown that all decommissioning materials (except reactor pressure vessel internals) could be disposed of as Class A waste. Spent fuel disassembly hardware from the Shippingport Core-3 was analyzed for long-lived activation products specified in 10 CFR 61, and the hardware was classified with respect to 10 CFR 61 waste disposal rules. Niobium-94 and (63)Ni concentrations in Inconel-X750 and stainless steel components exceeded their Class C limits.

NUREG/CR-5897: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE SOUTH TEXAS PROJECT NUCLEAR POWER PLANT. BUMGARDNER,J.D.; NICHOLAUS,J.R.; MOFFITT,N.E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1993. 36pp. 9401140022. PNL-8104. 77797:301.

In a study sponsored by the U.S. Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory has developed and applied a methodology for deriving plant-specific risk-based inspection guidance for the auxiliary feedwater (AFW) system at pressurized water reactors that have not undergone probabilistic risk assessment (PRA). This methodology uses existing PRA results and plant operating experience information. Existing PRA-based inspection guidance information recently developed for the NRC for various plants was used to identify generic component failure modes. This information was then combined with plant-specific and industry-wide component information and failure data to identify failure modes and failure mechanisms for the AFW system at the selected plants. South Texas Project was selected as one of a series of plants for study. The product of this effort is a prioritized listing of AFW failures which have occurred at the plant and at other PWRs. This listing is intended for use by NRC inspectors in the preparation of inspection plans addressing AFW risk-important components at the South Texas Project plant.

NUREG/CR-5898: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE POINT BEACH NUCLEAR POWER PLANT. LLOYD,R.C.; MOFFITT,N.E.; GORE,B.F.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1993. 32pp. 9303120004. PNL-8105. 74240:225.

In a study sponsored by the U.S. Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory has developed and applied a methodology for deriving plant-specific risk-based inspection guidance for the auxiliary feedwater (AFW) system at pressurized water reactors that have not undergone probabilistic risk assessment (PRA). This methodology uses existing PRA results and plant operating experience information. Existing PRA-based inspection guidance information recently developed for the NRC for various plants was used to identify generic component failure modes. This information was then combined with plant-specific and industry-wide component information and failure data to identify failure modes and failure mechanisms for the AFW system at the selected plants. Point Beach was selected as one of a series of plants for study. The product of this effort is a prioritized listing of AFW failures which have occurred at the plant and at other PWRs. This listing is intended for use by NRC inspectors in the preparation of inspection plans addressing AFW risk-important components at the Point Beach plant.

NUREG/CR-5901: A SIMPLIFIED MODEL OF AEROSOL SCRUBBING BY A WATER POOL OVERLYING CORE DEBRIS INTERACTING WITH CONCRETE.Final Report. POWERS,D.A.; SPRUNG,J.L. Sandia National Laboratories. November 1993. 126pp. 9312160284. SAND92-1422. 77509:158.

A classic model of aerosol scrubbing from bubbles rising through water is applied to the decontamination of gases produced during core debris interactions with concrete. The model, originally developed by Fuchs, describes aerosol capture by diffusion, sedimentation, and inertial impaction. This original model for spherical bubbles is modified to account for ellipsoidal distortion of the bubbles. Eighteen uncertain variables are identified in the application of the model to the decontamination of aerosols produced during core debris interactions with concrete by a water pool of specified depth and subcooling. These uncertain variables include properties of the aerosols, the bubbles, the water and the ambient pressure. Ranges for the values of the uncertain variables are defined based on the literature and experience. Probability density functions for values of these uncertain variables are hypothesized. The model of decontamination is applied in a Monte Carlo sampling of the decontamination by pools of specified depth and subcooling. Results are analyzed using a nonparametric, order statistical analysis that allows quantitative differentiation of stochastic and phenomenological uncertainty. The sampled values of the decontamination factors are used to construct estimated probability density functions for the decontamination factor at confidence levels of 50%, 90% and 95%. The decontamination factors for pools 30, 50, 100, 200, 300, and 500 cm deep and subcooling levels of 0, 2, 5, 10, 20, 30, 50, 70 degrees C are correlated by simple polynomial regression. These polynomial equations can be used to estimate decontamination factors at prescribed confidence levels.

NUREG/CR-5903: VALIDATION OF SMART SENSOR TECHNOLOGIES FOR INSTRUMENT CALIBRATION REDUCTION IN NUCLEAR POWER PLANTS. HASHEMIAN,H.M.; MITCHELL,D.W.; PETERSEN,K.M.; et al. Analysis & Measurement Services Corp. January 1993. 168pp. 9302230183. 64964:004.

This report presents the preliminary results of a research and development project on the validation of new techniques for on-line testing of calibration drift of process instrumentation channels in nuclear power plants. These techniques generally involve a computer- based data acquisition and data analysis system to trend the output of a large number of instrument channels and identify the channels that have drifted out of tolerance. This helps limit the calibration effort to those channels which need the calibration, as opposed to the current nuclear industry practice of calibrating essentially all the safety- related instrument channels at every refueling outage.

NUREG/CR-5907: CORE-CONCRETE INTERACTIONS WITH OVERLYING WATER POOLS.The WETCOR-1 Test. BLOSE,R.E. Ktech Corp. POWERS,D.A.; COPUS,E.R.; et al. Sandia National Laboratories. November 1993. 171pp. 9312070274. SAND92-1563. 77352:001.

The WETCOR-1 test of simultaneous interactions of a high-temperature melt with water and a limestone/common-sand concrete is described. The test used a 34.1-kg melt of 76.8 w/o Al(2)O(3), 16.9 w/o CaO, and 4.0 w/o SiO(2) heated by induction using tungsten susceptors. Once quasi-steady attack on concrete by the melt was established, an attempt was made to quench the melt at 1850 K with 295 K water flowing at 57 liters per minute. Net power into the melt at the time of water addition was 0.61 ± 0.19 W/cm(3). The test configuration used in the WETCOR-1 test was designed to delay melt freezing to the walls of the test fixture. This was done to test hypotheses concerning the inherent stability of crust formation when high-temperature melts are exposed to water. No instability in crust formation was observed. The flux of heat through the crust to the water pool maintained over the melt in the test was found to be 0.52 ± 0.13 MW/m(2). Solidified crusts were found to attenuate aerosol emissions during the melt concrete interactions by factors of 1.3 to 3.5. The combination of a solidified crust and a 30-cm deep subcooled water pool was found to attenuate aerosol emissions by factors of 3 to 15.

NUREG/CR-5911: SOURCE TERM EVALUATION FOR RADIOACTIVE LOW-LEVEL WASTE DISPOSAL PERFORMANCE ASSESSMENT. COWGILL,M.G.; SULLIVAN,T.M. Brookhaven National Laboratory. January 1993. 97pp. 9302230094. BNL-NUREG-52334. 64968:001.

Information compiled on the low-level radioactive waste disposed at the three currently operating commercial disposal sites during the period 1987 - 1989 have been reviewed and processed in order to determine the total activity distribution in terms of waste stream, waste classification and waste form. The review identified deficiencies in the information currently being recorded on shipping manifests and the development of a uniform manifest is recommended. The data from waste disposed during 1989 at one of the sites (Richland, WA) were more detailed than the data available during other years and at other sites, and thus were amenable to a more in-depth treatment. This included determination of the distribution of activity for each radionuclide by waste form, and thus enabled these data to be evaluated in terms of the specific needs for improved modeling of releases from waste packages. From the results, preliminary lists have been prepared of the isotopes which might be the most significant from the aspect of the development of a source term model.

NUREG/CR-5914: CHEMICAL COMPOSITION AND RT(NDT) DETERMINATIONS FOR MIDLAND WELD WF-70. NANSTAD,R.K.; MCCABE,D.E.; SWAIN,R.L.; et al. Oak Ridge National Laboratory. December 1992. 379pp. 9302010102. ORNL-6740. 64728:001.

The Heavy-Section Steel Irradiation Program Tenth Irradiation Series has the objective to investigate the effects of radiation on the fracture toughness of the low-upper-shelf submerged-arc welds (B&W designation WF-70) in the reactor pressure vessel of the canceled Midland Unit 1 nuclear plant. This report discusses determination of variations in chemical composition and reference temperature (RT(NDT)) throughout the welds. Specimens were machined from different sections and through thickness locations in both the beltline and nozzle course welds. The nil-ductility transition temperatures ranged from -40 to -60 degrees C (-40 and -76 degrees F) while the RT(NDT)s, controlled by the Charpy behavior, varied from -20 to 37 degrees C (-4 to 99 degrees F). The upper-shelf energies varied from 77 to 108 J (57 to 80 ft-lb). The combined data revealed a mean 41-J (30-ft-lb) temperature of -8 degrees C (17 degrees F) with a mean upper-shelf energy of 88 J (65 ft-lb). The copper contents range from 0.21 to 0.34 wt % in the beltline weld and from 0.37 to 0.46 wt % in the nozzle course weld. Atom probe field ion microscope analyses indicated substantial depletion of copper in the matrix but no evidence of copper clustering. Statistical analyses of the Charpy and chemical composition results as well as interpretation of the ASME procedures for RT(NDT) determination are discussed.

NUREG/CR-5917 V01: SENSITIVITY AND UNCERTAINTY ANALYSES APPLIED TO ONE-DIMENSIONAL RADIONUCLIDE TRANSPORT IN A LAYERED FRACTURED ROCK.MULTFRAC - Analytic Solutions And Local Sensitivities. GUREGHIAN,A.B.; WU,Y.-T.; SAGAR,B.; et al. Center for Nuclear Waste Regulatory Analyses. December 1992. 139pp. 9302230456. CNWRA 91-010. 64959:001.

This report documents the derivation and verification of the closed form analytical solutions of the one-dimensional nondispersive and isothermal transport of a radionuclide in a layered system of saturated planar fractures coupled with diffusion into the adjacent saturated rock matrix. The analytical solutions are based on the Laplace transform method where the domains of radionuclide migration in both fractures and rock layers are one-dimensional and of the semi-infinite type, implying in this instance that radionuclide diffusion from the fractures wall to the rock matrix may extend to infinity. The sorption phenomena in both fracture and rock matrix layers are described by a linear equilibrium sorption isotherm. Two types of radionuclide release modes are considered: the continuously decaying, and the periodically fluctuating decaying source, which may, in turn, be subject to step and band release modes. The initial concentrations in the fracture and rock matrix layers may be assigned spatially varying values in the case of the first, whereas uniform ones may be implemented in both cases.

NUREG/CR-5917 V02: SENSITIVITY AND UNCERTAINTY ANALYSES APPLIED TO ONE-DIMENSIONAL RADIONUCLIDE TRANSPORT IN A LAYERED FRACTURED ROCK.Evaluation Of The Limit State Approach. WU,Y.-T.; GUREGHIAN,A.B.; SAGAR,B.; et al. Center for Nuclear Waste Regulatory Analyses. December 1992. 71pp. 9301220115. CNWRA91-010. 64650:021.

The Limit State approach is based on partitioning the parameter space into two parts: one in which the performance measure is smaller than a chosen value (called the limit state), and the other in which it is larger. Through a Taylor expansion at a suitable point, the partitioning surface (called the limit state surface) is approximated as either a linear or quadratic function. The success and efficiency of the limit state method depends upon choosing an optimum point for the Taylor expansion. The point in the parameter space that has the highest probability of producing the value chosen as the limit state is optimal for expansion. When the parameter space is transformed into a standard Gaussian space, the optimal expansion point, known as the Most Probable Point (MPP), has the property that its location on the Limit State surface is closest to the origin. Additionally, the projections onto the parameter axes of the vector from the origin to the MPP are the sensitivity coefficients. Once the MPP is determined and the Limit State surface approximated, formulas (see Equations 4-7 and 4-8) are available for determining the probability of the performance measure being less than the limit state. By choosing a succession of limit states, the entire cumulative distribution of the performance measure can be determined. Methods for determining the MPP and also for improving the estimate of the probability are discussed in this report.

NUREG/CR-5922: MODULAR HIGH-TEMPERATURE GAS-COOLED REACTOR SHORT-TERM THERMAL RESPONSE TO FLOW AND REACTIVITY TRANSIENTS. CLEVELAND,J.C. Oak Ridge National Laboratory. February 1993. 55pp. 9303110375. ORNL/TM-12179. 74212:239.

The research reported here has been conducted at the Oak Ridge National Laboratory for the Nuclear Regulatory Commission's Division of Regulatory Applications of the Office of Nuclear Regulatory Research. The short-term thermal response of the Modular High-Temperature Gas- Cooled Reactor (MHTGR) is analyzed for a range of flow and reactivity transients. These transients include loss of forced circulation without scram, spurious withdrawal of a control rod group, moisture ingress, control rod and control rod group ejections, and a rapid core cooling event. For each event analyzed, an event description, a discussion of the analysis approach and assumptions, and results are presented. When possible, results of these analyses are compared with those presented by the designers in the MHTGR Probabilistic Risk Assessment. The importance of inherent safety features is illustrated, and conclusions are presented regarding the safety performance of the MHTGR. Recommendations are made for a more in-depth examination of MHTGR response for some of the analyzed transients. The coupled heat transfer-neutron kinetics model is described in detail in Appendix A.

NUREG/CR-5926: SANS INVESTIGATION OF LOW ALLOY STEELS IN NEUTRON IRRADIATED, ANNEALED, AND REIRRADIATED CONDITIONS. KAMPMANN,R.; FRISIUS,F.; HACKBARTH,H.; et al. Institute for Materials Research. February 1993. 44pp. 9303120100. MEA-2490. 74240:262.

Small Angle Neutron Scattering (SANS) experiments were made on several low alloy steels and submerged-arc welds prototypic of nuclear reactor vessel construction. The objective was the characterization of radiation-enhanced and/or radiation-induced precipitation contributing to mechanical property changes observed in tensile and notch ductility tests of the materials. The materials were irradiated in the UBR Test Reactor under closely controlled conditions. A portion of the samples were examined in the 288 degrees C irradiated (I) condition; others were examined in the postirradiation annealed (IA) condition and in the 288 degrees C reirradiated (IAR) condition. Experimental variables included material composition (primarily %Cu, %P, %Ni content), postirradiation annealing temperature (454 degrees C and 399 degrees C) reirradiation fluence level, and neutron-fluence rate (~0.08, 0.7, and 9 x 10(12) n/cm(2)-s(-1), E > 1 MeV). The apparent influence of the described variables on the size, number density, and composition of copper-rich precipitates was the primary focus of the SANS analyses. SANS observations are related to measured notch ductility and tensile property changes, with a view toward mechanistic explanation of the observed mechanical property trends for I, IA, and IAR conditions.

NUREG/CR-5927 V01: EVALUATION OF A PERFORMANCE ASSESSMENT METHODOLOGY FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES.Evaluation Of Modeling Approaches. KOZAK,M.W.; OLAGUE,N.E.; RAO,R.R.; et al. Sandia National Laboratories. August 1993. 87pp. 9309210242. SAND91-2802. 76486:118.

This report represents an update to our earlier reports on low-level waste performance assessment. This update addresses needed improvements and recommended approaches to the existing state of the art in modeling, treatment of uncertainty, and use of data. Greater attention is paid to developing an integrated approach to performance assessment than was done in earlier developments of the methodology. Furthermore, insights are being developed by participating in validation exercises, and by evaluating which validation data are needed to improve confidence in the methodology. It is emphasized that the performance assessment methodology update is a work in progress; the recommendations given here will form the general directions toward which the methodology is heading, but some of the specific approaches may continue to evolve as the research progresses.

NUREG/CR-5928: ISLOCA RESEARCH PROGRAM.Final Report. GALYEAN,W.J.; KELLY,D.L.; SCHROEDER,J.A.; et al. EG&G Idaho, Inc. July 1993. 107pp. 9308160131. EGG-2685. 76120:108.

This report contains a compilation of information generated during the ISLOCA research program. Presented is a screening analysis and a procedures guide for performing an ISLOCA evaluation. This methodology has been distilled from past analyses performed for the U.S. Nuclear Regulatory Commission and documented in a series of NUREG/CR reports. The methodology comprises five distinct steps: (a) containment penetration screening; (b) interfaces for ISLOCA analysis; (c) mechanisms for failing the pressure boundary; (d) construction of event trees and estimation of rupture probabilities; and (e) quantification of the event tree. Included in the methodology are steps required for a detailed human reliability analysis. In addition, this report presents a BWR ISLOCA evaluation, a survey of PWR auxiliary building designs and identification of one design deemed most disadvantageous with respect to ISLOCA risk, and a PWR ISLOCA cost/benefit analysis.

NUREG/CR-5933: HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM RISK-BASED INSPECTION GUIDE FOR DRESDEN NUCLEAR POWER STATION UNITS 2 AND 3. SHIER,W.; VILLARAN,M.; GUNTHER,W. Brookhaven National Laboratory. February 1993. 48pp. 9303120079. BNL-NUREG-52343. 74241:280.

A review of the operating experience for the High Pressure Coolant Injection (HPCI) system at the Dresden Nuclear Power Station Units 2 and 3 is described in this report. The information for this review was obtained from Dresden Licensee Event Reports (LERs) that were generated between 1980 and 1989. These LERs have been categorized into 23 failure modes that have been prioritized based on probabilistic risk assessment considerations. In addition, the results of the Dresden operating experience review have been compared with the results of a similar, industry wide operating experience review. This comparison provides an indication of areas in the Dresden HPCI system that should be given increased attention in the prioritization of inspection resources.

NUREG/CR-5934: HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM RISK-BASED INSPECTION GUIDE FOR QUAD-CITIES STATION,UNITS 1 AND 2. VILLARAN,M.; TRAVIS,R.; GUNTHER,W. Brookhaven National Laboratory. January 1993. 55pp. 9303190108. BNL-NUREG-52344. 74314:001.

The High Pressure Coolant Injection (HPCI) system has been examined from a risk perspective. A System Risk-Based Inspection Guide (S-RIG) has been developed as an aid to HPCI system inspections at Quad Cities. Included in this S-RIG is a discussion of the role of HPCI in mitigating accidents and a presentation of PRA-based failure modes which could prevent proper operation of the system. The S-RIG uses industry operating experience, including plant-specific illustrative examples to augment the basic PRA failure modes. It is designed to be used as a reference for both routine inspections and the evaluation of the significance of component failures.

NUREG/CR-5936: ENHANCEMENTS TO THE ACCIDENT PRECURSOR METHODOLOGY. BOHNHOFF,W.J.; DINGMAN,S.E.; CAMP,A.L. Sandia National Laboratories. February 1993. 138pp. 9303120072. SAND92-2109. 74241:001.

A feasibility study for developing an improved tool and improved models for performing event assessments is described. The study indicates that the IRRAS code should become the base tool for performing event assessments, but that modifications would be needed to make it more suitable for routine use. Alternative system modeling approaches are explored and an approach is recommended that is based on improved train-level models. These models are demonstrated for Grand Gulf and Sequoyah. The insights that can be gained from importance measures are also demonstrated. The feasibility of using Individual Plant Examination (IPE) submittals as the basis for train-level models for precursor studies was also examined. The level of reported detail was found to vary widely, but in general, the submittals would not provide sufficient information to fully define the model. The feasibility of developing an industry risk profile from Accident Sequence Precursor results and of trending precursor results for individual plants is considered but not recommended because of data sparsity.

NUREG/CR-5937: INTENTIONAL DEPRESSURIZATION ACCIDENT MANAGEMENT STRATEGY FOR PRESSURIZED WATER REACTORS. BROWNSON,D.A.; HANEY,L.N.; CHIEN,N.D. EG&G Idaho, Inc. April 1993. 165pp. 9305100005. EGG-2688. 74858:270.

In a previous investigation of the Surry nuclear power station, it was concluded that intentional depressurization of the reactor coolant system (RCS) could prevent or mitigate the effects of direct containment heating (DCH) during a station blackout transient. Two strategies, early and late depressurization, were investigated as methods to mitigate DCH. The investigation concluded that since there are greater opportunities to recover plant functions before core damage occurs and operator response uncertainties are lessened, the strategy of late depressurization is preferred over early depressurization. The results of the Surry analysis were extended to other U.S. pressurized water reactors (PWRs) in order to evaluate their capability to successfully employ the late depressurization strategy to prevent or mitigate DCH. By applying appropriate scaling factors to the selected key parameters, this evaluation resulted in the categorization of four PWR groups based upon their perceived late depressurization capability. In this report, a PWR representative of each of the four PWR groups was chosen for detailed analysis of its capability to intentionally depressurize employing the late depressurization strategy. The phenomenological behavior, hardware performance, and operational performance of these PWRs during the intentional depressurization strategy were considered. The phenomenological behavior was analyzed using the SCDAP/RELAP5/MOD3 severe accident analysis code. The results of these evaluations were then extended to the remaining PWRs comprising each PWR group.

NUREG/CR-5938: NATIONAL PROFILE ON COMMERCIALLY GENERATED LOW-LEVEL RADIOACTIVE MIXED WASTE. KLEIN,J.A.; MROCHEK,J.E.; JOLLEY,R.L.; et al. Oak Ridge National Laboratory. December 1992. 468pp. 9301220128. ORNL-6731. 64650:094.

This report details the findings and conclusions drawn from a survey undertaken as part of a joint U.S. Nuclear Regulatory Commission and U.S. Environmental Protection Agency-sponsored project entitled "National Profile on Commercially Generated Low-Level Radioactive Mixed Waste." The overall objective of the work was to compile a national profile on the volumes, characteristics, and treatability of commercially generated low-level mixed waste for 1990 by five major facility categories-academic, industrial, medical, and NRC/Agreement State-licensed government facilities and nuclear utilities. Included in this report are descriptions of the methodology used to collect and collate the data, the procedures used to estimate the mixed waste generation rate for commercial facilities in the United States in 1990, and the identification of available treatment technologies to meet applicable EPA treatment standards (40 CFR Part 268) and, if possible, to render the hazardous component of specific mixed waste streams nonhazardous. The report also contains information on existing and potential commercial waste treatment facilities that may provide treatment for specific waste streams identified in the national survey.

NUREG/CR-5942: SEVERE ACCIDENT SOURCE TERM CHARACTERISTICS FOR SELECTED PEACH BOTTOM SEQUENCES PREDICTED BY THE MELCOR CODE. CARBAJO,J.J. Oak Ridge National Laboratory. September 1993. 345pp. 9310120226. ORNL/TM-12229. 76739:001.

The purpose of this report is to compare in-containment source terms developed for NUREG-1159, which used the Source Term Code Package (STCP), with those generated by MELCOR to identify significant differences. For this comparison, two short-term depressurized station blackout sequences (with a dry cavity and with a flooded cavity) and a Loss-Of-Coolant Accident (LOCA) concurrent with complete loss of the Emergency Core Cooling System (ECCS) were analyzed for the Peach Bottom Atomic Power Station (a BWR-4 with a Mark I containment). The results indicate that for the sequences analyzed, the two codes predict similar total in-containment release fractions for each of the element groups. However, the MELCOR/CORBH Package predicts significantly longer time for vessel failure and reduced energy of the released material for the station blackout sequences (when compared to the STCP results). MELCOR also calculated smaller releases into the environment than STCP for the station blackout sequences.

NUREG/CR-5943: SENSITIVITY ANALYSIS AND BENCHMARKING OF THE BLT LOW-LEVEL WASTE SOURCE TERM CODE. SUEN,C.J.; SULLIVAN,T.M. Brookhaven National Laboratory. July 1993. 81pp. 9307270012. BNL-NUREG-52346. 75802:236.

To evaluate the source term for low-level waste disposal, a comprehensive model had been developed and incorporated into a computer code, called BLT (Breach-Leach-Transport). Since the release of the original version, many new features and improvements had also been added to the Leach model of the code. This report consists of two different studies based on the new version of the BLT code: 1) a series of verification/sensitivity tests; and 2) benchmarking of the BLT code using field data. Based on the results of the verification/sensitivity tests, we concluded that the new version represents a significant improvement and it is capable of providing more realistic simulations of the leaching process. Benchmarking work was carried out to provide a reasonable level of confidence in the model predictions. In this study, the experimentally measured release curves for nitrate, technetium-99 and tritium from the saltstone lysimeters operated by Savannah River Laboratory were used. The model results are observed to be in general agreement with the experimental data, within the acceptable limits of uncertainty.

NUREG/CR-5944: A CHARACTERIZATION OF CHECK VALVE DEGRADATION AND FAILURE EXPERIENCE IN THE NUCLEAR POWER INDUSTRY. CASADA,D.A.; TODD,M.D. Oak Ridge National Laboratory. September 1993. 187pp. 9310120234. ORNL-6734. 76739:342.

Check valve operating problems in recent years have resulted in significant operating transients, increased cost and decreased system availability. As a result, additional attention has been given to check valves by utilities (resulting in the formation of the Nuclear Industry Check Valve Group), as well as the U.S. Nuclear Regulatory Commission and the American Society of Mechanical Engineers Operation and Maintenance Committee. All these organizations have the fundamental goal of ensuring reliable operation of check valves. A key ingredient to an engineering-oriented reliability improvement effort is a thorough understanding of relevant historical experience. A detailed review of historical failure data, available through the Institute of Nuclear Power Operation's Nuclear Plant Reliability Data System, has been conducted. The focus of the review is on check valve failures that have involved significant degradation of the valve internal parts. A variety of parameters are considered, including size, age, system of service, method of failure discovery, the affected valve parts, attributed causes, and corrective actions.

NUREG/CR-5949: ASSESSMENT OF THE POTENTIAL FOR HIGH PRESSURE MELT EJECTION RESULTING FROM A SURRY STATION BLACKOUT TRANSIENT. KNUDSON,D.L.; DOBBE,C.A. EG&G Idaho, Inc. November 1993. 194pp. 9312170064. EGG-2689. 77515:039.

Containment integrity could be challenged by direct heating associated with a high pressure melt ejection (HPME) of core materials following reactor vessel breach during certain severe accidents. Intentional reactor coolant system (RCS) depressurization, where operators latch pressurizer relief valves open, has been proposed as an accident management strategy to reduce risks by mitigating the severity of HPME. However, decay heat levels, valve capacities, and other plant-specific characteristics determine whether the required operator action will be effective. Without operator action, natural circulation flows could heat ex-vessel RCS pressure boundaries (surge line and hot leg piping, steam generator tubes, etc.) to the point of failure before vessel breach, providing an alternate mechanism for RCS depressurization and HPME mitigation. This report contains an assessment of the potential for HPME during a Surry station blackout transient without operator action and without recovery. The assessment included a detailed transient analysis using the SCDAP/RELAP5/MOD3 computer code to calculate the plant response with and without hot leg countercurrent natural circulation, with and without reactor coolant pump seal leakage, and with variations on selected core damage progression parameters. RCS depressurization-related probabilities were also evaluated, primarily based on the code results.

NUREG/CR-5951: THE MANAGEMENT OF ATWS BY BORON INJECTION. DIAS,M.P. Moratuwa, Univ. of, Sri Lanka. YAN,H.; THEOFANOUS,T.G. California, Univ. of, Santa Barbara, CA. March 1993. 58pp. 9304020304. 74451:037.

Experimental simulations of the multidimensional mixing/stratification phenomena in the lower plenum of a Boiling Water Reactor during operation of the Standby Liquid Control System (SLCS) are reported. The simulations both at full- and 1/2-scale allow the demarcation of the fully entraining regime, which is also interpreted in terms of an approximate consideration of flow stability criteria, based on the local Froude number. These results are combined with analyses of the subsequent dispersion (of entrained boron) throughout the primary system and in combination with neutron diffusion and natural convection (power-flow-void coupling) predictions of reactor kinetic behavior are made. On this basis the performance of SLCS during ATWS is assessed and a discussion on current Emergency Operating Procedures is offered.

NUREG/CR-5952: EVALUATION OF CRACK POP-INS AND THE DETERMINATION OF THEIR RELEVANCE TO DESIGN CONSIDERATIONS. MCCABE,D.E. Oak Ridge National Laboratory. February 1993. 30pp. 9303110379. ORNL/TM-12247. 74212:057.

The issue with regard to crack pop-ins is to determine if such events are significant to design considerations. The literature contains ample evidence of pop-in occurrences, but scant information is offered on how pop-ins should be handled as an issue for design problems. Because there are two types of cleavage crack origins, the problem was subdivided into two classes of materials, monolithic and weldments with brittle zones. The weldment situation can be analyzed as a crack-arrest toughness capability problem, following the recommendations of Sumpter et al. For monolithic materials, pop-ins are more dangerous, since they appear to be a part of the more commonly encountered full-cleavage K(Jc) instability distributions. A recommendation is made on how to determine if pop-in events lie outside of the larger body of K(Jc) instabilities. The evaluation procedure recommended by the American Society for Testing and Materials for pop-ins seems to dismiss the possibility that small crack jumps can be a safety-related issue. The present work suggests that nearly all pop-in events, regardless of the magnitude of crack jump, are relevant to safety issues.

NUREG/CR-5953: STUDIES OF HUMAN PERFORMANCE DURING OPERATING EVENTS:1990-1992. MEYER,O.R.; HILL,S.G.; STEINKE,W.F. EG&G Idaho, Inc. January 1993. 86pp. 9302010090. EGG-2690. 64727:001.

In order to better evaluate the human factors influencing operator performance during operating events at nuclear power stations, the Office for Analysis and Evaluation of Operational Data (AEOD) of the U.S. Nuclear Regulatory Commission (NRC) initiated a project to perform onsite analyses of selected events. An interim report on the results of the analysis of the first six selected events was published in May 1991. Causal factors for operator performance were identified and categorized in the interim report. Subsequently, 10 more onsite analyses have been conducted. This is a report on the analysis of the 16 events. Summaries of the 16 operating events analyzed can be found in Appendix A.

NUREG/CR-5955: MATERIALS AND DESIGN BASES ISSUES IN ASME CODE CASE N-47. HUDDLESTON,R.L.; SWINDEMAN,R.W. Oak Ridge National Laboratory. April 1993. 42pp. 9306010335. ORNL/TM-12266. 75062:284.

A preliminary evaluation of the design bases (principally ASME Code Case N-47) was conducted for design and operation of reactors at elevated temperatures where the time-dependent effects of creep, creep-fatigue, and creep ratcheting are significant. Areas where Code rules or regulatory guides may be lacking or inadequate to ensure the operation over the expected life cycles for the next-generation advanced high-temperature reactor systems, with designs to be certified by the U.S. Nuclear Regulatory Commission, have been identified as unresolved issues. Twenty-two unresolved issues were identified and brief scoping plans developed for resolving these issues.

NUREG/CR-5956: CONSIDERATION OF UNCERTAINTIES IN SOIL-STRUCTURE INTERACTION COMPUTATIONS. COSTANTINO,C.J.; MILLER,C.A. City College of New York, New York, NY. * Viking Systems International. December 1992. 140pp. 9301220172. CEERC-91-105. 64680:046.

This report presents a summary of the results obtained in a study conducted to evaluate and quantify some important effects of soil- structure interaction (SSI) on the seismic response of Category I facilities. The current procedures utilized in SSI evaluations typically use complex computer analyses to treat the various important aspects of the problem. The purpose of this study has been to provide input to the Staff for developing consistent guidelines for assessing adequacy of these SSI computations as well as clarification of some portions of the SRP relating to SSI calculations. The specific areas addressed in this study are: (a) summarizing the criteria needed when using the large computer codes used in SSI studies; (b) providing recommendations for specification of control point location for soil sites; (c) providing specific criteria to allow the Staff to judge the adequacy of fixed base structural analyses, (d) the development of expanded guidelines for inclusion of variability in soil properties in the SSI calculations, and (e) the development of estimates of radiation damping inherent in the detailed numerical analyses performed or SSI evaluations of typical Category I structures.

NUREG/CR-5957: SYSTEM 80+(TM) CONTAINMENT -- STRUCTURAL DESIGN REVIEW. GREIMANN,L.; FANOUS,F.; CHALLA,R.; et al. Iowa State Univ., Ames, IA. May 1993. 112pp. 9306010330. IS-5083. 75063:001.

A review of the structural design of the Combustion Engineering (CE) System 80+(TM) steel containment was completed. The stress analysis and the evaluation of the structure against buckling were performed by using BOSOR4 and BOSOR5 finite difference software, respectively. The CE System 80+(TM) containment was modelled as an axisymmetric shell consisting of different segments and mesh points with the additional mass of the penetrations and appurtenance being smeared around the circumference. The transition region was modelled using elastic springs with a foundation modulus of 180 lbs/in(3). The stresses due to the individual loads (dead loads, internal and external pressures and temperatures) were computed using the stress analysis option in the BOSOR4 program. The stresses from individual loads were combined according to ASME Code into stress intensities. Service Level B loadings produced a 20 percent over-stress in a small zone just above the transition region. All other stress intensities were within allowable limits. For the System 80+(TM), the perfect shell with an elastic material was initially analyzed. The calculated factor of safety values were 2.3 (Level B) and 1.59 (Levels C and D). Finally, sensitivity studies were conducted to investigate the effects of mesh size and transition zone stiffness on the controlling buckling load.

NUREG/CR-5958: TWO-PARAMETER FRACTURE MECHANICS: THEORY AND APPLICATIONS. O'DOWD,N.P. Imperial College, London, UK. SHIH,C.F. Brown Univ., Providence, RI. * Navy, Dept. of. February 1993. 45pp. 9303120046. CDNSWCSMECR1692. 74238:225.

A family of self-similar fields provides the two parameters required to characterize the full range of high- and low-triaxiality crack tip states. The two parameters, J and Q, have distinct roles: J sets the size scale of the process zone over which large stresses and strains develop, while Q scales the near-tip stress distribution relative to a high triaxiality reference stress state. An immediate consequence of the theory is this: it is the toughness values over a range of crack tip constraint that fully characterize the material's fracture resistance. It is shown that Q provides a common scale for interpreting cleavage fracture and ductile tearing data thus allowing both failure modes to be incorporated in a single toughness locus. The evolution of Q, as plasticity progresses from small scale yielding to fully yielded conditions, has been quantified for several crack geometries and for a wide range of material strain hardening properties. An indicator of the robustness of the J-Q fields is introduced; Q as a field parameter and as a pointwise measure of stress level is discussed.

NUREG/CR-5959: HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM RISK-BASED INSPECTION GUIDE FOR ENRICO FERMI ATOMIC POWER PLANT,UNIT 2. VILLARAN,M.; TRAVIS,R.; GUNTHER,W. Brookhaven National Laboratory. January 1993. 55pp. 9303190116. BNL-NUREG-52352. 74314:059.

The High Pressure Coolant Injection (HPCI) system has been examined from a risk perspective. A System Risk-Based Inspection Guide (S-RIG) has been developed as an aid to HPCI system inspections at the Enrico Fermi Unit 2 Nuclear Power Plant. Included in this S-RIG is a discussion of the role of HPCI in mitigating accidents and a presentation of PRA-based failure modes which could prevent proper operation of the system. The S-RIG uses industry operating experience, including plant-specific illustrative examples to augment the basic PRA failure modes. It is designed to be used as a reference for both routine inspections and the evaluation of the significance of component failures.

NUREG/CR-5961: POSTTEST DESTRUCTIVE EXAMINATION OF THE STEEL LINER IN A 1:6-SCALE REACTOR CONTAINMENT MODEL. LAMBERT,L.D. Sandia National Laboratories. February 1993. 40pp. 9302220439. SAND92-1721. 64899:228.

A 1:6-scale model of a nuclear reactor containment model was built and tested at Sandia National Laboratories as part of a research program sponsored by the Nuclear Regulatory Commission to investigate containment integrity. The overpressure test was terminated due to leakage from a large tear in the steel liner. A limited destructive examination of the liner and anchorage system was conducted to gain information about the failure mechanism and is described. Sections of liner were removed in areas where liner distress was evident or where large strains were indicated by instrumentation during the test. The condition of the liner, anchorage system, and concrete for each of the regions that were investigated are described. The probable cause of the observed posttest condition of the liner is discussed.

NUREG/CR-5962: HEALTH AND SAFETY IMPACTS FROM DISCRETE SOURCES OF NATURALLY-OCCURRING AND ACCELERATOR-PRODUCED RADIOACTIVE MATERIALS (NARM). NUSSBAUMER,D.; WIBLIN,C.; WELCH,L. Advanced Systems Technology, Inc. February 1993. 54pp. 9303120051. 74238:274.

This report characterizes discrete sources of naturally-occurring and accelerator-produced radioactive material (NARM) and estimates risks posed by the possession, use and disposal of them. A distinction between discrete and diffuse NARM sources is made with discrete sources being high activity, low volume and diffuse sources being low activity, high volume. Two nanocuries per gram is used as a separation guide between high and low activity, although use of this value does not impact the report's conclusions. Most NARM is under regulatory control of States that either license or register users but reporting requirements are not uniform. Use in consumer products has declined with virtually no production today; however, lack of information available concerning radiation exposures resulting form possession of ageing radium sources precludes a quantitative risk assessment in this report. The report identifies the type of information needed to permit such an assessment. Regarding accelerator-produced radioactive material (ARM), use of this material in nuclear medicine programs has recently increased. Available radiation exposure data regarding ARM handling and use indicates that the risk to workers and the public is low at this time.

NUREG/CR-5964: SAPHIRE TECHNICAL REFERENCE MANUAL:IRRAS/SARA VERSION 4.0. RUSSELL,K.D.; ATWOOD,C.L.; SATTISON,M.B.; et al. EG&G Idaho, Inc. January 1993. 75pp. 9302230366. EGG-2692. 64998:199.

This report provides information on the principles used in the construction and operation of Version 4.0 of the Integrated Reliability and Risk Analysis System (IRRAS) and the System Analysis and Risk Assessment (SARA) system. It summarizes the fundamental mathematical concepts of sets and logic, fault trees, and probability. The report then describes the algorithms that these programs use to construct a fault tree and to obtain the minimal cut sets. It gives the formulas used to obtain the probability of the top event from the minimal cut sets, and the formulas for probabilities that are appropriate under various assumptions concerning repairability and mission time. It defines the measures of basic event importance that these programs can calculate. The report gives an overview of uncertainty analysis using simple Monte Carlo sampling or Latin Hypercube sampling, and states the algorithms used by these programs to generate random basic event probabilities from various distributions. Further references are given, and a detailed example of the reduction and quantification of a simple fault tree is provided in an appendix.

NUREG/CR-5966: A SIMPLIFIED MODEL OF AEROSOL REMOVAL BY CONTAINMENT SPRAYS. POWERS,D.A. Sandia National Laboratories. BURSON,S.B. Severe Accident Issues Branch. June 1993. 180pp. 9306210236. SAND92-2689. 75407:095.

Spray system in nuclear reactor containments are described. The scrubbing of aerosols from containment atmospheres by spray droplets is discussed. Uncertainties are identified in the prediction of spray performance when the sprays are used as a means for decontaminating containment atmospheres. A mechanistic model based on current knowledge of the physical phenomena involved in spray performance is developed. With this model, a quantitative uncertainty analysis of spray performance is conducted using a Monte Carlo method to sample 20 uncertain quantities related to phenomena of spray droplet behavior as well as the initial and boundary conditions expected to be associated with severe reactor accidents. Results of the uncertainty analysis are used to construct simplified expressions for spray decontamination coefficients. Two variables that affect aerosol capture by water droplets are not treated as uncertain; they are (1) 'Q', spray water flux into the containment, and (2) 'H', the total fall distance of spray droplets. The choice of values of these variables is left to the user since they are plant and accident specific. Also, they can usually be ascertained with some degree of certainty. The spray decontamination coefficients are found to be sufficiently dependent on the extent of decontamination that the fraction of the initial aerosol remaining in the atmosphere, m(f), is explicitly treated in the simplified expressions.

NUREG/CR-5968: POTENTIAL CHANGE IN FLAW GEOMETRY OF AN INITIALLY SHALLOW FINITE-LENGTH SURFACE FLAW DURING A PRESSURIZED-THERMAL-SHOCK TRANSIENT. SHUM,D.K.; BRYSON,J.W.; MERKLE,J.G. Oak Ridge National Laboratory. September 1993. 31pp. 9311010029. ORNL/TM-12279. 76982:265.

This study presents preliminary estimates on whether an initially shallow, axially oriented, inner-surface finite-length flaw in a PWR-RPV would tend to elongate in the axial direction and/or deepen into the wall of the vessel during a postulated PTS transient. Analysis results obtained based on the assumptions of (1) linear-elastic material response, and (2) cladding with the same toughness as the base metal, indicate that a nearly semicircular flaw would likely propagate in the axial direction followed by propagation into the wall of the vessel. Note that these results correspond to initiation within the lower-shelf fracture toughness temperature range, and that their general validity within the lower-transition temperature range remains to be determined. The sensitivity of the numerical results and conclusions to the following analysis assumptions are evaluated: (1) reference flaw geometry along the entire crack front and especially within the cladding region; (2) linear-elastic vs elastic-plastic description of material response; and (3) base-material-only vs bimaterial cladding-base vessel-model assumption. The sensitivity evaluation indicates that the analysis results are very sensitive to the above assumptions. This report is designated HSST Report No. 139.

NUREG/CR-5969: J AND CTOD ESTIMATION EQUATIONS FOR SHALLOW CRACKS IN SINGLE EDGE NOTCH BEND SPECIMENS. KIRK,M.T.; DODDS,R.H. Illinois, Univ. of, Urbana, IL. * Navy, Dept. of. July 1993. 28pp. 9308160121. UILU-ENG91-2013. 76121:333.

Fracture toughness values determined using shallow cracked single edge notch bend, SE(B), specimens of structural thickness are useful for structural integrity assessments. Results from two dimensional plane strain finite-element analyses are used to develop J and CTOD estimation strategies appropriate for application to both shallow and deep crack SE(B) specimens. Crack depth to specimen width (a/W) ratios between 0.05 and 0.70 are modelled using Ramberg-Osgood strain hardening exponents (n) between 4 and 50. The estimation formulas divide J and CTOD into small scale yielding (SSY) and large scale yielding (LSY) components. For each case, the SSY component is determined by the linear elastic stress intensity factor, K(I). The formulas differ in evaluation of the LSY component. The techniques considered include: estimating J or CTOD from plastic work based on load line displacement (A(p1)/LLD), from plastic work based on crack mouth opening displacement (A(p1)/CMOD), and from the plastic component of crack mouth opening displacement (CMOD(p1)). A(p1)/CMOD provides the most accurate J estimation possible.

NUREG/CR-5970: APPROXIMATE TECHNIQUES FOR PREDICTING SIZE EFFECTS ON CLEAVAGE FRACTURE TOUGHNESS (JC). KIRK,M.T.; DODDS,R.H. Illinois, Univ. of, Urbana, IL. * Navy, Dept. of. July 1993. 38pp. 9308160116. UILU-ENG92-2016. 76122:001.

This investigation examines the ability of an elastic T-stress analysis coupled with a modified boundary layer (MBL) solution to predict stresses ahead of a crack tip in a variety of planar geometries. The approximate stresses are used as input to estimate the effective driving force for cleavage fracture (J(o)) using the micromechanically based approach introduced by Dodds and Anderson. Finite element analyses for a wide variety of planar cracked geometries are conducted which have elastic biaxiality parameters (β) ranging from -0.99 (very low constraint) to +2.96 (very high constraint). The magnitude and sign of β indicate the rate at which crack-tip constraint changes with increasing applied load. All results pertain to a moderately strain hardening material (strain hardening exponent (η) of 10). These analyses suggest that β is an effective indicator of both the accuracy of T-MBL estimates of J(o) and of applicability limits on evolving fracture analysis methodologies (i.e. T-MBL, J-Q, and J/J(o)). Specifically, when $1\beta1 > 0.4$ these analyses show that the T-MBL approximation of J(o) is accurate to within 20% of a detailed finite-element analysis. As "structural type" configurations, i.e. shallow cracks in tension, generally have $1\beta1 > 0.4$, it appears that only an elastic analysis may be needed to determine reasonably accurate J(o) values for structural conditions.

NUREG/CR-5971: CONTINUUM AND MICROMECHANICS TREATMENT OF CONSTRAINT IN FRACTURE. DODDS,R.H. Illinois, Univ. of, Urbana, IL. SHIH,C.F. Brown Univ., Providence, RI. ANDERSON,T.L. Texas A&M Univ., College Station, TX. July 1993. 47pp. 9308160294. UILU-ENG92-2014. 76119:122.

Two complementary methodologies are described to quantify the effects of crack-tip stress triaxiality (constraint) on the macroscopic measures of elastic-plastic fracture toughness, J and CTOD. In the continuum mechanics methodology, two parameters, J and Q, suffice to characterize the full range of near-tip environments at the onset of fracture. A micromechanics methodology is described which predicts the toughness locus using crack-tip stress fields and critical J-values from a few fracture toughness tests. A robust micromechanics model for cleavage fracture has evolved from the observations of a strong, spatial self-similarity of crack-tip principal stresses under increased loading and across different fracture specimens. This report explores the fundamental concepts of the J-Q description of crack-tip fields, the fracture toughness locus and micromechanics approaches to predict the variability of macroscopic fracture toughness with constraint under elastic-plastic conditions. Computational results are presented for a surface cracked plate containing a 6:1 semi-elliptical, a=t/4 flaw subjected to remote uniaxial and biaxial tension.

NUREG/CR-5972: EFFECTS OF NONSTANDARD HEAT TREATMENT TEMPERATURES ON TENSILE AND CHARPY IMPACT PROPERTIES OF CARBON-STEEL CASTING REPAIR WELDS. NANSTAD,R.K.; GOODWIN,G.M.; SWINDEMAN,M.J. Oak Ridge National Laboratory. April 1993. 118pp. 9304210258. ORNL/TM-12280. 74677:192.

Carbon steel castings are used for a number of different components in nuclear power plants, including valve bodies and bonnets. Components are often repaired by welding processes, and both welded components and the repair welds are subjected to a variety of postweld heat treatments (PWHT) with temperatures as high as 899 degrees C (1650 degrees F), well above the normal 593 to 677 degrees C (1100 to 1250 degrees F) temperature range. The temperatures noted are above the A1 transformation temperature for the materials used for these components. A test program was conducted to investigate the potential effects of such "nonstandard" PWHTs on mechanical properties of carbon steel casting welds. Four weldments were fabricated, two each with the shielded-metal-arc (SMA) and flux-cored-arc (FCA) processes, with a high-carbon and low-carbon filler metal in each case. All four welds were sectioned and given simulated PWHTs at temperatures from 621 to 899 degrees C (1150 to 1650 degrees F) in increments of 56 degrees C (100 degrees F) and for times of 5, 10, 20, and 40 h at each temperature. Hardness, tensile, and Charpy V-notch (CVN) impact tests were conducted for the as-welded and heat-treated conditions. Results were plotted versus a time-temperature relationship (tempering parameter) to enable a more direct comparison of the effects of the various PWHT conditions. Heat treatments at 621 and 677 degrees C (1150 and 1250 degrees F) gave results amenable to prediction, and regression analyses are presented for those conditions. Heat treatments at 732 to 899 degrees C (1350 to 1650 degrees F), however, resulted in substantial changes in mechanical properties of these SMA and FCA welds, with the changes not amenable to prediction and highly dependent on the weld metal. Heat treatments in that temperature range should not be applied to these materials without prior qualification for the intended use.

NUREG/CR-5973: CODES AND STANDARDS AND OTHER GUIDANCE CITED IN REGULATORY DOCUMENTS. NICKOLAUS,J.R.; VINTHER,R.W.; MAGUIRE-MOFFITT; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. January 1993. 73pp. 9302230373. PNL-8462. 64960:001.

As part of the U.S. Nuclear Regulatory Commission (NRC) Standard Review Plan Update and Development Program, Pacific Northwest Laboratory developed a listing of industry consensus codes and standards and other government and industry guidance referred to in regulatory documents. This listing identifies the version of the code or standard cited in the regulatory document, the regulatory document, and the current version of the code or standard. It also provides a summary characterization of the nature of the citation. This listing was developed from electronic searches of the Code of Federal Regulations and NRC's Bulletins, Information Notices, Circulars, Generic Letters, Policy Statements, Regulatory Guides, and Standard Review Plan (NUREG-0800).

NUREG/CR-5975: INCENTIVE REGULATION OF INVESTOR-OWNED NUCLEAR POWER PLANTS BY PUBLIC UTILITY REGULATORS. MCKINNEY,M.D.; ELLIOTT,D.B. Battelle Memorial Institute, Pacific Northwest Laboratory. January 1993. 80pp. 9302220421. PNL-8466. 64899:001.

The U.S. Nuclear Regulatory Commission (NRC) periodically surveys the Federal Energy Regulatory Commission (FERC) and state regulatory commissions that regulate utility owners of nuclear power plants. The NRC is interested in identifying states that have established economic or performance incentive programs applicable to nuclear power plants, including states with new programs, how the programs are being implemented, and in determining the financial impact of the programs on the utilities. The NRC interest stems from the fact that such programs have the potential to adversely affect the safety of nuclear power plants. The current report is an update of NUREG/CR-4911, Incentive Regulation of Nuclear Power Plants by State Regulators, published in February 1991. The information in this report was obtained from interviews conducted with each state regulatory agency that administers an incentive program and each utility that owns at least 10% of an affected nuclear power plant. The agreements, orders, and settlements that form the basis for each incentive program were reviewed as required. The interviews and supporting documentation form the basis for the individual state reports describing the structure and financial impact of each incentive program.

NUREG/CR-5976: DEVELOPMENT AND USE OF A TRAIN-LEVEL PROBABILISTIC RISK ASSESSMENT. SMITH,C.L.; FOWLER,R.D.; WOLFRAM,L.M. EG&G Idaho, Inc. April 1993. 74pp. 9306010327. EGG-2694. 75062:209.

The Idaho National Engineering Laboratory examined the potential for the development of train-level probabilistic risk assessment (PRA) databases. These train-level databases will allow the Nuclear Regulatory Commission to investigate effects on plant core damage frequency (CDF) given a train is failed or taken out of service. The intent of this task was to develop user-friendly databases that required a minimal amount of personnel involvement to be usable. It was originally intended that the train-level models would not be expanded to include basic events below the top gate of a train, with the possible exception of including some of the major train-related components (e.g., important pumps and motor-operated valves). It was found that a database similar to the original plant PRA provided the accuracy needed to measure the changes in plant CDF. The Peach Bottom Unit 2 NUREG-1150 PRA (a large fault tree model) and the Beaver Valley Unit 2 IPE (a large event tree model) were selected to demonstrate the feasibility of developing train-level databases. Five different methods for developing train-level databases were hypothesized and are examined. Ultimately, two train-level databases were developed using the Peach Bottom Unit 2 PRA and one train-level database was developed using the Beaver Valley Unit 2 IPE. The development, use, limitations, and results of these train-level databases are discussed.

NUREG/CR-5977: A PERFORMANCE INDICATOR OF THE EFFECTIVENESS OF HUMAN-MACHINE INTERFACES FOR NUCLEAR POWER PLANTS. MORAY,N.; JONES,B.J.; RASMUSSEN,J.; et al. Illinois, Univ. of, Urbana, IL. January 1993. 88pp. 9302010095. UILU-ENG92-4007. 64727:091.

Effective interfaces must call up operators' deep understanding of plant operation if operators are to deal effectively with normal operation and diagnosis of transients. The present research examines the ability of a memory recall task to indicate the ability of an interface to couple plant state to operator knowledge. Novices, people with intermediate experience, and experienced nuclear power plant operators viewed three kinds of displays. They watched nine simulated transients and tried to recall the values of variables, or the states through which the plant passed, and to detect and diagnose the nature of the transients. The displays were simulated analog instruments, simulated analog with pressure-temperature graphics, and an animated representation of the Rankine cycle. The recall tasks did not show promise as indirect performance indicators of the quality of the interfaces, but the diagnosis test detected differences in the quality of the displays and the levels of expertise.

NUREG/CR-5978: SOURCE TERM ATTENUATION BY WATER IN THE MARK I BOILING WATER REACTOR DRYWELL. POWERS,D.A. Sandia National Laboratories. September 1993. 200pp. 9311010038. SAND92-2688. 76983:001.

Mechanistic models of aerosol decontamination by an overlying water pool during core debris/concrete interactions and spray removal of aerosols from a Mark I drywell atmosphere are developed. Eighteen uncertain features of the pool decontamination model and 19 uncertain features of the model for the rate coefficient of spray removal of aerosols are identified. Ranges for values of parameters that characterize these uncertain features of the models are established. Probability density functions for values within these ranges are assigned according to a set of rules. A Monte Carlo uncertainty analysis of the decontamination factor produced by water pools 30 and 50 cm deep and subcooled 0-70 K is performed. An uncertainty analysis for the rate constant of spray removal of aerosols is done for water fluxes of 0.25, 0.01, and 0.001 cm(3) H(2)O/cm(2)-S and decontamination factors of 1.1, 2, 3.3, 10, 100, and 1000.

NUREG/CR-5980: THREE DIMENSIONAL REDISTRIBUTION OF TRITIUM FROM A POINT OF RELEASE INTO A UNIFORM UNSATURATED SOIL:A Deterministic Model For Tritium Migration In An Arid Disposal Site. SMILES,D.E.; GARDNER,W.R.; SCHULZ,R.K. California, Univ. of, Berkeley, CA. January 1993. 30pp. 9302020469. 64729:020.

This report presents a three-dimensional model for tritium migration in an arid waste disposal site. When tritiated water is released at a point in a uniform and relatively dry soil it redistributes in both the liquid and vapor phases. The flux density of tritium in each phase is of the same order of magnitude however so tritium redistribution is modeled as if transfer occurs "in parallel" in the liquid and vapor phases. The approach we describe uses the diffusion equation cast in radial (spherical) coordinates and takes into account radioactive decay. It permits calculation of radial profiles of tritium concentration, within and external to a sphere of released solution. We assume the concentration within this sphere initially to be uniform. The solution also predicts attenuation and rate of advance of the maximum of tritium concentration as it advances in the soil. With deep disposal in a desert soil, the model predicts that tritium migration will be very short range, with a maximum of a few meters.

NUREG/CR-5981: THE EFFECT OF ELECTRIC DISCHARGE MACHINED NOTCHES ON THE FRACTURE TOUGHNESS OF SEVERAL STRUCTURAL ALLOYS. JOYCE,J.A. U.S. Naval Academy, Annapolis, MD. LINK,R.E. Navy, Dept. of. September 1993. 90pp. 9310130032. 76744:290.

Recent computational studies of the stress and strain fields at the tip of very sharp notches have shown that the stress and strain fields are very weakly dependent on the initial geometry of the notch once the notch has been blunted to a radius that is 6 to 10 times the initial root radius. It follows that if the fracture toughness of a material is sufficiently high so that fracture initiation does not occur in a specimen until the crack-tip opening displacement (CTOD) reaches a value from 6 to 10 times the size of the initial notch tip diameter, then the fracture toughness will be independent of whether a fatigue crack or a machined notch served as the initial crack. In this experimental program the fracture toughness (J(Ic) and J resistance (J-R) curve, and CTOD) for several structural alloys was measured using specimens with conventional fatigue cracks and with EDM machined notches. The results of this program have shown, in fact, that most structural materials do not achieve initiation CTOD values on the order of 6 to 10 times the radius of even the smallest EDM notch tip presently achievable. It is found furthermore that tougher materials do not seem to be less dependent on the type of notch tip present. Some materials are shown to be much more dependent on the type of initial notch tip used, but no simple pattern is found that relates this observed dependence to the material strength, toughness, or strain hardening rate.

NUREG/CR-5982: EFFECTIVENESS OF CONTAINMENT SPRAYS IN CONTAINMENT MANAGEMENT. NOURBAKHSH,H.P.; PEREZ,S.E.; LEHNER,J.R. Brookhaven National Laboratory. May 1993. 54pp. 9306180261. BNL-NUREG-52354. 75427:109.

A limited study has been performed assessing the effectiveness of containment sprays to mitigate particular challenges which may occur during a severe accident. Certain aspects of three specific topics related to using sprays under severe accident conditions were investigated. The first was the effectiveness of sprays connected to an alternate water supply and pumping source because the actual containment spray pumps are inoperable. This situation could occur during a station blackout. The second topic concerned the adverse as well as beneficial effects of using containment sprays during severe accident scenarios where the containment atmosphere contains substantial quantities of hydrogen along with steam. The third topic was the feasibility of using containment sprays to moderate the consequences of DCH.

NUREG/CR-5983: SAFETY ASPECTS OF FORCED FLOW COOLDOWN TRANSIENTS IN MODULAR HIGH TEMPERATURE GAS-COOLED REACTORS. KROEGER,P.G. Brookhaven National Laboratory. May 1993. 24pp. 9306010323. BNL-NUREG-52355. 75058:297.

During some of the design basis accidents in Modular High Temperature Gas Cooled Reactors (MHTGRs), the main Heat Transport System (HTS) and the Shutdown Cooling System (SCS) are assumed to have failed. Decay heat is then removed by the passive Reactor Cavity Cooling System (RCCS) only. If either forced flow cooling system becomes available during such a transient, its restart could significantly reduce the downtime. This report used the THATCH code to examine whether such restart, during a period of elevated core temperatures, can be accomplished within safe limits for fuel and metal component temperatures. If the reactor is scrammed, either system can apparently be restarted at any time, without exceeding any safe limits. However, under unscrammed conditions a restart of forced cooling can lead to recriticality, with fuel and metal temperatures significantly exceeding the safety limits.

NUREG/CR-5984: CODE AND MODEL EXTENSIONS OF THE THATCH CODE FOR MODULAR HIGH TEMPERATURE GAS-COOLED REACTORS. KROEGER,P.G.; KENNETT,R.J. Brookhaven National Laboratory. May 1993. 46pp. 9306010318. BNL-NUREG-52356. 75062:166.

This report documents several model extensions and improvements of the THATCH code, a code to model thermal and fluid flow transients in High Temperature Gas-Cooled Reactors. A heat exchanger model was added, which can be used to represent the steam generator of the main Heat Transport System or the auxiliary Shutdown Cooling System. This addition permits the modeling of forced flow cooldown transients with the THATCH code. An enhanced upper head model, considering the actual conical and spherical shape of the upper plenum and reactor upper head was added, permitting more accurate modelling of the heat transfer in this region. The revised models are described, and the changes and addition to the input records are documented.

NUREG/CR-5987: MICROBIAL-INFLUENCED CEMENT DEGRADATION - LITERATURE REVIEW. ROGERS,R.D.; HAMILTON,M.A.; MCCONNELL,J.W. EG&G Idaho, Inc. March 1993. 34pp. 9303300171. EGG-2695. 74407:322.

The Nuclear Regulatory Commission stipulates that disposed low- level radioactive waste (LLW) be stabilized. Because of apparent ease of use and normal structural integrity, cement has been widely used as a binder to solidify LLW. However, the resulting waste forms are sometimes susceptible to failure due to the actions of waste constituents, stress, and environment. This report reviews literature which addresses the effects of microbiologically influenced chemical attack on cement-solidified LLW. Groups of microorganisms are identified, which are capable of metabolically converting organic and inorganic substrates into organic and mineral acids. Such acids aggressively react with concrete and can ultimately lead to structural failure. Mechanisms inherent in microbial-influenced degradation of cement-based material are the focus of this report. This report provides sufficient evidence of the potential for microbial-influenced deterioration of cement-solidified LLW to justify the enumeration of the conditions necessary to support the microbiological growth and population expansion, as well as the development of appropriate tests necessary to determine the resistance of cement-solidified LLW to microbiological-induced degradation that could impact the stability of the waste form.

NUREG/CR-5988: SOIL CHARACTERIZATION METHODS FOR UNSATURATED LOW-LEVEL WASTE SITES. WIERENGA,P.J.; YOUNG,M.H. Arizona, Univ. of, Tucson, AZ. GEE,G.W.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1993. 150pp. 9303120106. PNL-8480. 74236:120.

To support a license application for the disposal of low-level radioactive waste (LLW), applicants must characterize the unsaturated zone. This requires an integrated plan to be developed for sampling and analyzing the soil horizons for physical and hydraulic properties. This document provides a strategy for developing this characterization plan. It describes principles of contaminant flow and transport, site characterization and monitoring strategies, and data management. It also discusses methods and practices that are currently used to monitor properties and conditions in the soil profile, how these properties influence water and waste migration, and why they are important to the license application. The methods part of the document is divided into sections on laboratory and field-based properties, then further subdivided into the description of methods for determining 18 physical, flow, and transport properties. Because of the availability of detailed procedures in many texts and journal articles, the reader is often directed for details to the available literature. References are made to experiments performed at the Las Cruces Trench site, New Mexico, that support LLW site characterization activities. A major contribution from the Las Cruces study is the experience gained in handling data sets for site characterization and the subsequent use of these data sets in modeling studies.

NUREG/CR-5989: PERFORMANCE TESTING OF EXTREMITY DOSIMETERS--PILOT TEST. FOX,R.A.; HARTY,R.; MCDONALD,J.C. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1993. 51pp. 9308160263. PNL-8467. 76119:169.

A working group of the Health Physics Society Standards Committee (HPSSC) has issued a draft standard for extremity dosimeters. To determine the appropriateness of the proposed standard, Pacific Northwest Laboratory (PNL) has conducted three separate evaluations of the performance by processors of extremity dosimeters. The dosimeters were tested in each of the irradiation categories specified in the draft standard: high-energy photons (general and accident dosimetry), low-energy photons (general and accident dosimetry), beta particles, neutrons (first and second evaluations only), and a mixture category. In the first evaluation only about 60% of the processors met the draft standard's performance criteria for accuracy and precision. The second evaluation showed an overall improvement of 15% to 18%, but most processors were still unable to meet the performance criteria consistently in all irradiation categories. After these evaluations, PNL suggested several changes to the draft standard, including redefining some of the test categories and making the tolerance levels of the criteria more consistent with those in the standard for whole-body dosimetry. This report summarizes the third evaluation, which yielded an overall pass rate of 87%. Suggestions are given to render the draft standard generally more consistent with the criteria for whole-body dosimetry.

NUREG/CR-5991: PORFLOW: A MULTIFLUID MULTIPHASE MODEL FOR SIMULATING FLOW, HEAT TRANSFER, AND MASS TRANSPORT IN FRACTURED POROUS MEDIA. User's Manual - Version 2.41. RUNCHAL,A.K. Analytic & Computional Research, Inc. SAGAR,B. Center for Nuclear Waste Regulatory Analyses. February 1993. 221pp. 9303120062. CNWRA 92-003. 74240:004.

The PORFLOW software package is designed to simulate flow, heat transfer, and mass transport in three-dimensional heterogenous porous and fractured media. Phase change and gas phase flow is included. Radionuclide decay chains of up to four members can be included in transport analyses. The mathematical basis of the model is described in Chapters 2 and 3, the code structure is discussed in Chapters 4 and 5, detailed instructions for the user are in Chapter 6, and a few test problems are in Chapter 7. The PORFLOW is a general purpose software that can be adapted to many different problems. Analytic and Computational Research, Incorporated of Los Angeles, CA owns the copyright to the software, however, the U.S. Government has retained limited rights on its use.

NUREG/CR-5993 V01: METHODS FOR DEPENDENCY ESTIMATION AND SYSTEM UNAVAILABILITY EVALUATION BASED ON FAILURE DATA STATISTICS.Summary Report. AZARM,M.A.; HSU,F.; MARTINEZ-GURIDI; et al. Brookhaven National Laboratory. July 1993. 37pp. 9307270039. BNL-NUREG-52362. 75803:032.

This report introduces a new perspective on the basic concept of dependent failures where the definition of dependency is based on clustering in failure times of similar components. This perspective has two significant implications: firstly, it relaxes the conventional assumption that dependent failures must be simultaneous and result from a severe shock; secondly, it allows the analyst to use all the failures in a time continuum to estimate the potential for multiple failures in a window of time (e.g., a test interval), therefore arriving at a more accurate value for system unavailability. In addition, the models developed here provide a method for plant-specific analysis of dependency, reflecting the plant-specific maintenance practices that reduce or increase the contribution of dependent failures to system unavailability. The proposed methodology can be used for screening analysis of failure data to estimate the fraction of dependent failures among the failures. In addition, the proposed method can evaluate the impact of the observed dependency on system unavailability and plant risk. The formulations derived in this report have undergone various levels of validations through computer simulation studies and pilot applications. The pilot applications of these methodologies showed that the contribution of dependent failures of diesel generators in one plant was negligible, while in another plant, it was quite significant. It also showed that in the plant with significant contribution of dependency to Emergency Power System (EPS) unavailability, the contribution changed with time. Similar findings were reported for the Containment Fan Cooler breakers. Drawing such conclusions about system performance would not have been possible with any other reported dependency methodologies.

NUREG/CR-5993 V02: METHODS FOR DEPENDENCY ESTIMATION AND SYSTEM UNAVAILABILITY EVALUATION BASED ON FAILURE DATA STATISTICS.Detailed Description And Applications. AZARM,M.A.; HSU,F.; MARTINEZ-GURIDI; et al. Brookhaven National Laboratory. July 1993. 78pp. 9307270044. BNL-NUREG-52362. 75802:143.

This report introduces a new perspective on the basic concept of dependent failures where the definition of dependency is based on clustering in failure times of similar components. This perspective has two significant implications: firstly, it relaxes the conventional assumption that dependent failures must be simultaneous and result from a severe shock; secondly, it allows the analyst to use all the failures in a time continuum to estimate the potential for multiple failures in a window of time (e.g., a test interval), therefore arriving at a more accurate value for system unavailability. In addition, the models developed here provide a method for plant-specific analysis of dependency, reflecting the plant-specific maintenance practices that reduce or increase the contribution of dependent failures to system unavailability. The proposed methodology can be used for screening analysis of failure data to estimate the fraction of dependent failures among the failures. In addition, the proposed method can evaluate the impact of the observed dependency on system unavailability and plant risk. The formulations derived in this report have undergone various levels of validations through computer simulation studies and pilot applications. The pilot applications of these methodologies showed that the contribution of dependent failures of diesel generators in one plant was negligible, while in another plant, it was quite significant. It also showed that in the plant with significant contribution of dependency to Emergency Power System (EPS) unavailability, the contribution changed with time. Similar findings were reported for the Containment Fan Cooler breakers. Drawing such conclusions about system performance would not have been possible with any other reported dependency methodologies.

NUREG/CR-5995: TECHNICAL SPECIFICATION ACTION STATEMENTS REQUIRING SHUTDOWN.A Risk Perspective With Application To The RHR/SSW Systems Of A BWR. MANKAMO,T. Avaplan Oy (Finland). KIM,I.S.; SAMANTA,P.K. Brookhaven National Laboratory. November 1993. 186pp. 9312170073. BNL-NUREG-52364. 77515:215.

When safety systems fail during power operation, the limiting conditions for operation (LCOs) and associated action statements of technical specifications typically require that the plant be shut down within the limits of allowed outage time (AOT). However, when a system needed to remove decay heat, such as the residual heat removal (RHR) system, is inoperable or degraded, shutting down the plant may not necessarily be preferable, from a risk perspective, to continuing power operation over a usual repair time, giving priority to the repairs. The risk impact of the basic operational alternatives, i.e., continued operation or shutdown, was evaluated for failures in the RHR and standby service water (SSW) systems of a boiling-water reactor (BWR) nuclear power plant. A complete or partial failure of the SSW system fails or degrades not only the RHR system but other front-line safety systems supported by the SSW system. This report presents: (a) the methodology to evaluate the risk impact of LCOs and associated AOT; (b) the results of risk evaluation from its application to the RHR and SSW systems of a BWR; (c) the findings from the risk-sensitivity analyses to identify alternative operational policies; and (d) the major insights and recommendations to improve the technical specifications action statements.

NUREG/CR-5996: SUBSURFACE INJECTION OF RADIOACTIVE TRACERS:Field Experiment For Model Validation Testing. FAYER,M.J.; SISSON,J.B.; JORDAN,W.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1993. 50pp. 9303120155. PNL-8499. 74234:302.

Accurate predictions of the movement of radioactive contaminants from disposal facilities are required to evaluate effects, optimize data collection, design remediation strategies, and predict the longterm results of such strategies. A field experiment was undertaken in 1980 and 1981 to provide data to test the limits of model predictions. The purpose of this report is to provide a complete record of data generated during that field experiment for use as a model validation test case. The report combines the information in Sisson and Lu (1984) with unpublished laboratory and field data on the hydraulic properties of the sediments and core data collected at the end of the experiment. The unique features of this experiment were the documented control of the inputs, the three-dimensional nature of the experiment, the measurement of radioactive tracers in situ, and the use of multiple injections. The in situ monitoring methods were neutron moderation for water content and gamma energy analysis for tracer concentration. The data are provided on 3.5-in; diskettes. The data include observation and injection well construction details, injection solution concentrations, radioactive tracer and water content distributions in space and time, neutron probe calibration information, and sediment properties determined in both the laboratory and field.

NUREG/CR-5997: CSNI PROJECT FOR FRACTURE ANALYSES OF LARGE-SCALE INTERNATIONAL REFERENCE EXPERIMENTS (PROJECT FALSIRE). BASS,B.R.; PUGH,C.E.; KEENEY-WALKER,J.; et al. Oak Ridge National Laboratory. June 1993. 150pp. 9307020010. ORNL/TM-12307. 75585:204.

This report summarizes the recently completed Phase I of the Project for Fracture Analysis of Large-Scale International Reference Experiments (Project FALSIRE). Project FALSIRE was created by the Fracture Assessment Group (FAG) of Principal Working Group No. 3 (PWG/3) of the Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency's (NEA's) Committee on the Safety of Nuclear Installations (CSNI). Motivation for the project was derived from recognition by the CSNI-PWG/3 that inconsistencies were being revealed in predictive capabilities of a variety of fracture assessment methods, especially in ductile fracture applications. As a consequence, the CSNI/FAG was formed to evaluate fracture prediction capabilities currently used in safety assessments of nuclear components. Members are from laboratories and research organizations in Western Europe, Japan, and the United States of America (USA). On behalf of the CSNI/FAG, the U.S. Nuclear Regulatory Commission's (NRC's) Heavy-Section Steel Technology (HSST) Program at the Oak Ridge National Laboratory (ORNL) and the Gesellschaft fur Anlagen--und Reaktorsicherheit (GRS), Koln, Federal Republic of Germany (FRG) had responsibility for organization arrangements related to Project FALSIRE. The group is chaired by H. Schulz from GRS, Koln, FRG.

NUREG/CR-5998: SIMULATION OF UNSATURATED FLOW AND NONREACTIVE SOLUTE TRANSPORT IN A HETEROGENEOUS SOIL AT THE FIELD SCALE. ROCKHOLD,M.L. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1993. 65pp. 9303120158. PNL-8496. 74241:139.

A field-scale, unsaturated flow and solute transport experiment at the Las Cruces trench site in New Mexico was simulated as part of a "blind" modeling exercise to demonstrate the ability or inability of uncalibrated models to predict unsaturated flow and solute transport in spatially variable porous media. Simulations were conducted using a recently developed multiphase flow and transport simulator. Uniform and heterogeneous soil models were tested, and data from a previous experiment at the site were used with an inverse procedure to estimate water retention parameters. A spatial moment analysis was used to provide a quantitative basis for comparing the mean observed and simulated flow and transport behavior. The results of this study suggest that defensible predictions of waste migration and fate at low- level waste sites will ultimately require site-specific data for model calibration.

NUREG/CR-5999: INTERIM FATIGUE DESIGN CURVES FOR CARBON, LOW-ALLOY, AND AUSTENITIC STAINLESS STEELS IN LWR ENVIRONMENTS. MAJUMDAR,S.; CHOPRA,O.K.; SHACK,W.J. Argonne National Laboratory. April 1993. 34pp. 9305100010. ANL-93/3. 74858:189.

Existing data in the literature on fatigue of carbon, low-alloy, and austenitic stainless steels in LWR environments are reviewed. It is found that both temperature and dissolved-oxygen concentration in water significantly affect fatigue life. At the very low dissolved-oxygen levels characteristic of pressurized water reactors and boiling water reactors with hydrogen-water chemistry, environmental effects on fatigue life are modest. However, at higher dissolved-oxygen levels ($\geq$100 ppb), significant reductions in fatigue life can occur. The susceptibility of carbon and low-alloy steels to reduced fatigue life is strongly related to sulfur concentration. Although the fatigue lives of austenitic stainless steels may be reduced, the reductions are much smaller than those observed in high-sulfur carbon and low-alloy steels. In oxygenated water, fatigue life depends strongly on strain rate. Interim fatigue design curves are proposed that take into account temperature, dissolved-oxygen level in the water, the sulfur level in the steel, and strain rate. Design curves for carbon and low-alloy steels for lives up to 10(8) cycles are also proposed.

NUREG/CR-6007: STRESS ANALYSIS OF CLOSURE BOLTS FOR SHIPPING CASKS. MOK,G.C.; FISCHER,L.E. Lawrence Livermore National Laboratory. HSU,S.T. Kaiser Engineering (formerly Kaiser Engineers). January 1993. 139pp. 9302010084. UCRL-ID-110637. 64726:111.

This report specifies the requirements and criteria for stress analysis of closure bolts for shipping casks containing nuclear spent fuels or high level radioactive materials. The specification is based on existing information concerning the structural behavior, analysis, and design of bolted joints. The approach taken was to extend the ASME Boiler and Pressure Vessel Code requirements and criteria for bolting analysis of nuclear piping and pressure vessels to include the appropriate design and load characteristics of the shipping cask. The characteristics considered are large, flat, closure lids with metal-to- metal contact within the bolted joint; significant temperature and impact loads; and possible prying and bending effects. Specific formulas and procedures developed apply to the bolt stress analysis of a circular, flat, bolted closure. The report also includes critical load cases and desirable design practices for the bolted closure, an in- depth review of the structural behavior of bolted joints, and a comprehensive bibliography of current information on bolted joints.

NUREG/CR-6011: REVIEW OF STRUCTURE DAMPING VALUES FOR ELASTIC SEISMIC ANALYSIS OF NUCLEAR POWER PLANTS. HASHIMOTO,P.S.; STEELE,L.K.; JOHNSON,J.J.; et al. EQE Engineering Consultants (formerly EQE Engineering, Inc.). March 1993. 715pp. 9303300197. 74405:001.

Current U.S. Nuclear Regulatory Commission guidance on structure damping values for elastic seismic design analysis of nuclear power plants are contained in Regulatory Guide 1.61 (R.G. 1.61). The objectives of the study described in this report are to investigate the adequacy of R.G. 1.61 structure damping values based on currently available data, and to recommend revisions to R.G. 1.61 as appropriate. Measured structure damping values, and associated structure, foundation, excitation, and input/response parameters, were collected and compiled. These data were analyzed to identify the parameters that significantly influence structure damping and to quantify structure damping in terms of these parameters. Based on this study, current R.G. 1.61 damping values for structure design are either adequate, or require only minor revision, depending on the structure material. More explicit guidance on structure damping values for seismic analysis to determine input to equipment has been prepared, along with other recommendations to improve the applicability of R.G. 1.61.

NUREG/CR-6012: STIFFNESS AND DAMPING PROPERTIES OF A LOW ASPECT RATIO SHEAR WALL BUILDING BASED ON RECORDED EARTHQUAKE RESPONSES. HASHIMOTO,P.S.; TIONG,L.W.; STEELE,L.K.; et al. EQE Engineering Consultants (formerly EQE Engineering, Inc.). March 1993. 250pp. 9304020300. 74449:001.

An investigation into the structural properties and seismic responses of a low aspect ratio shear wall building, which has construction similarity to typical nuclear plant structures, has been performed using actual recorded earthquake motions. This effort used a combination of modal identification to obtain structure modal parameters directly from the recorded motions, and elastic structural analysis using methods and criteria frequently employed by the nuclear industry. Modal parameters determined by modal identification provide excellent fits to the building motions recorded during the 1984 Morgan Hill earthquake. Modal parameters identified for the 1989 Loma Prieta earthquake are more uncertain. Investigation of building stiffnesses generally confirms the adequacy of bounding estimates currently recommended for nuclear plant structure seismic analysis. Damping values identified for this building supplement the database being compiled to investigate current nuclear plant structure damping criteria.

NUREG/CR-6013: METHODS USED FOR THE TREATMENT OF NON-PROPORTIONALLY DAMPED STRUCTURAL SYSTEMS. CONOSCENTE,J.P.; MASLENIKOV,O.R.; JOHNSON,J.J. EQE Engineering Consultants (formerly EQE Engineering, Inc.). May 1993. 62pp. 9306210371. 75402:286.

Non-proportional or non-classical damping is defined as a form of viscous damping that introduces coupling between the undamped modal coordinates of motion. Such problems have practical applications in the dynamic analysis of soil-structure systems, structure-equipment systems, and structural systems made of materials with different energy dissipation capacities. Presented in this report is a review of the methods most commonly used in structural analysis for the solution of the dynamic response of systems with non-proportional damping. Both rigorous and approximate methods are described. Since rigorous methods usually require large computational efforts, approximate methods using undamped mode shapes are often preferred. In the study described here, the accuracy of three approximate methods was evaluated for three benchmark problems, with various parametric variations. Results were compared with the exact solution for different combinations of structural properties. Based on these results, conclusions and recommendations are presented for the use of the selected approximate methods.

NUREG/CR-6014: HIGH PRESSURE COOLANT INJECTION SYSTEM RISK-BASED INSPECTION GUIDE FOR HATCH NUCLEAR POWER STATION. DIBIASIO,A.M. Brookhaven National Laboratory. May 1993. 57pp. 9306110025. BNL-NUREG-52367. 75336:303.

A review of the operating experience for the High Pressure Coolant Injection (HPCI) system at the Hatch Nuclear Power Station, Units 1 and 2, is described in this report. The information for this review was obtained from Hatch Licensee Event Reports (LERs) that were generated between 1980 and 1992. These LERs have been categorized into 23 failure modes that have been prioritized based on probabilistic risk assessment considerations. In addition, the results of the Hatch operating experience review have been compared with the results of a similar, industry wide operating experience review. This comparison provides an indication of areas in the Hatch HPCI system that should be given increased attention in the prioritization of inspection resources.

NUREG/CR-6015: STRUCTURAL AGING PROGRAM TECHNICAL PROGRESS FOR PERIOD JANUARY - DECEMBER 1992. NAUS,D.J.; OLAND,C.B. Oak Ridge National Laboratory. July 1993. 164pp. 9309030224. ORNL/TM-12342. 76328:021.

The Structural Aging (SAG) Program is conducted for the Nuclear Regulatory Commission (NRC) by the Oak Ridge National Laboratory (ORNL). The program has the overall objective of preparing an expandable handbook or report which will provide potential structural safety issues and acceptance criteria for use by the NRC in nuclear power plant evaluations of continued service. Initial focus of the program is on concrete and concrete-related materials which comprise safety-related (Category I) structures in light-water reactor facilities. The SAG Program is organized into four tasks: Task S.1-Program Management, Task S.2-Materials Property Data Base, Task S.3-Structural Component Assessment/Repair Technology, and Task S.4-Quantitative Methodology for Continued Service Determinations. In meeting the individual objectives of these tasks resources are drawn from ORNL with subcontract support from universities and other research laboratories. This report provides an overview of principal developments in each of the four program tasks from January 1, 1992 to December 31, 1992. Planned activities under each of these tasks are also presented.

NUREG/CR-6018: SURVEY AND ASSESSMENT OF CONVENTIONAL SOFTWARE VERIFICATION AND VALIDATION METHODS. MILLER,L.A.; GROUNDWATER,E.; MIRSKY,S.M.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). April 1993. 186pp. 9305100042. EPRI TR-102106. 74858:001.

This report documents the results of the first (of ten) tasks being performed under a contract jointly funded by the USNRC and EPRI to develop and document guidelines for the verification and validation of expert systems in the nuclear industry. This task conducted an extensive survey of conventional software verification and validation (V&V) methods. A total of 134 different methods were identified which can be applied to either the requirements-design of implementation phases of software. These methods were classified by a sequential lifecycle model, characterized by factors of power and ease-of-use, and assessed according to their applicability to expert systems. Expert systems were decomposed into four components: knowledge base, inference engine, interfaces, and tools/utilities. The conventional software V&V methods were found to be directly or, by extension, applicable to all of the expert system techniques except the knowledge base.

NUREG/CR-6021: A LITERATURE REVIEW OF COUPLED THERMAL-HYDROLOGIC-MECHAN ICAL -CHEMICAL PROCESSES PERTINENT TO THE PROPOSED HIGH-LEVEL WASTE REPOSITORY AT YUCCA MOUNTAIN. MANTEUFEL,R.D.; AHOLA,M.P.; TURNER,D.R.; et al. Center for Nuclear Waste Regulatory Analyses. July 1993. 240pp. 9308160184. CNWRA92-011. 76117:040.

A literature review has been conducted to determine the state of knowledge available in the modeling of coupled thermal (T), hydrologic (H), mechanical (M), and chemical (C) processes relevant to the design and/or performance of the proposed high-level waste (HLW) repository at Yucca Mountain, Nevada. The review focuses on identifying coupling mechanisms between individual processes and assessing their importance (i.e., if the coupling is either important, potentially important, or negligible).

The significance of considering THMC-coupled processes lies in whether or not the processes impact the design and/or performance objectives of the repository. A review, such as reported here, is useful in identifying which coupled effects will be important, hence which coupled effects will need to be investigated by the U.S. Nuclear Regulatory Commission in order to assess the assumptions, data, analyses, and conclusions in the design and performance assessment of a geologic repository. Although this work stems from regulatory interest in the design of the geologic repository, it should be emphasized that the repository design implicitly considers all of the repository performance objectives, including those associated with the time after permanent closure. The scope of this review is considered beyond previous assessments in that it attempts (with the current state-of-knowledge) to determine which couplings are important, and identify which computer codes are currently available to model coupled processes.

NUREG/CR-6022: HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM RISK-BASED INSPECTION GUIDE FOR BROWNS FERRY NUCLEAR POWER STATION. WONG,S.; DIBIASIO,A.M.; GUNTHER,W. Brookhaven National Laboratory. September 1993. 80pp. 9311010051. BNL-NUREG-52370. 76983:216.

The High Pressure Coolant Injection (HPCI) system has been examined from a risk perspective. A System Risk-Based Inspection Guide (S-RIG) has been developed as an aid to HPCI system inspections at the Browns Ferry Nuclear Power Plant, Units 1, 2 and 3. The role of the HPCI system in mitigating accidents is discussed in this S-RIG, along with insights on identified risk-based failure modes which could prevent proper operation of the system. The S-RIG provides a review of industry-wide operating experience, including plant-specific illustrative examples to augment the PRA and operational considerations in identifying a catalogue of basic PRA failure modes for the HPCI system. It is designed to be used as a reference for routine inspections, self-initiated safety system functional inspections (SSFIs), and the evaluation of risk significance of component failures at the nuclear power plant.

NUREG/CR-6023: GENERIC ANALYSIS FOR EVALUATION OF LOW CHARPY UPPER-SHELF ENERGY EFFECTS ON SAFETY MARGINS AGAINST FRACTURE OF REACTOR PRESSURE VESSEL MATERIALS. DICKSON,T.L. Oak Ridge National Laboratory. July 1993. 82pp. 9308160110. ORNL/TM-12340. 76120:215.

Appendix G to 10 CFR Part 50 requires that reactor pressure vessel beltline materials maintain Charpy upper-shelf energies of no less than 50 ft-lb during the plant operating life, unless it is demonstrated in a manner approved by the Nuclear Regulatory Commission (NRC), that lower values of Charpy upper-shelf energy provide margins of safety against fracture equivalent to those in Appendix G to Section XI of the ASME Code. Analyses based on acceptance criteria and analysis methods adopted in the ASME Code Case N-512 are described herein. Additional information on material properties was provided by the NRC, Office of Nuclear Regulatory Research, Materials Engineering Branch. These cases, specified by the NRC, represent generic applications to boiling water reactor and pressurized water reactor vessels. This report is designated as HSST Report No. 140.

NUREG/CR-6025: THE PROBABILITY OF MARK-I CONTAINMENT FAILURE BY MELT-ATTACK OF THE LINER. THEOFANOUS,T.G.; YAN,H. California, Univ. of, Santa Barbara, CA. PODOWSKI,M.Z.; et al. Rensselaer Polytechnic Institute, Troy, NY. November 1993. 350pp. 9312160292. 77508:150.

This report is a followup to the work presented in NUREG/CR-5423 addressing early failure of a BWR Mark I containment by melt atta[illegible]e liner, and it constitutes a part of the implementatio[illegible]isk-Oriented Accident Analysis Methodology (ROAAM) [illegible] therein. In particular, it expands the quantification to inclu[illegible]our independent evaluations carried out at Rensselaer Polytechnic Institute, Argonne National Laboratories, Sandia National Laboratories and ANATECH, Inc. on the various portions of the phenomenology involved. These independent evaluations are included here as Parts II through V. The results, and their integration in Part 1, demonstrate the substantial synergism and convergence necessary to recognize that the issue has been resolved.

NUREG/CR-6026: THEORETICAL AND EXPERIMENTAL INVESTIGATION OF THERMOHYDROLOGIC PROCESSES IN A PARTIALLY SATURATED, FRACTURED POROUS MEDIUM. GREEN,R.T.; MANTEUFEL,R.D. Center for Nuclear Waste Regulatory Analyses. DODGE,F.T.; et al. Southwest Research Institute. July 1993. 225pp. 9309090010. CNWRA 92-006. 76390:064.

The performance of a geologic repository for high-level nuclear waste will be influenced to a large degree by thermoh[illegible]logic phenomena created by the emplacement of heat-generating radioactive waste. The importance of these phenomena is manifest in that they can greatly affect the movement of moisture and the resulting transport of radionuclides from the repository. Thus, these phenomena must be well understood prior to a definitive assessment of a potential repository site. An investigation has been undertaken along three separate avenues of analysis: (1) laboratory experiments; (2) mathematical models; and (3) similitude analysis. A summary of accomplishments to date is as follows: (1) A review of the literature on the theory of heat and mass transfer in partially saturated porous medium; (2) A development of the governing conservation and constitutive equations; (3) A development of a dimensionless form of the governing equation; (4) A numerical study of the importance and sensitivity of flow to a set of dimensionless groups; (5) A survey and evaluation of experimental measurement techniques; (6) Execution of laboratory experiments of nonisothermal flow in a porous medium with a simulated fracture.

NUREG/CR-6027: PRELIMINARY EVALUATION OF SNUBBER SINGLE FAILURES. WARE,A.G.; BLANDFORD,R.K.; KELLY,D.L.; et al. EG&G Idaho, Inc. April 1993. 49pp. 9305100059. EGG-2697. 74857:316.

The United States Nuclear Regulatory Commission developed FIN L2430, Preliminary Evaluation of Snubber Single Failures, with the objective of performing a preliminary evaluation of the safety implication of a potential single failure of a snubber used to support safety-related piping or equipment. The Idaho National Engineering Laboratory staff conducted a qualitative review of a large number of light water reactor systems, and a quantitative stress analysis of four systems. A candidate list was developed that ranked the systems as having a high, medium, or low probability of causing a significant increase in the core damage frequency should a single snubber fail to function. Two systems were ranked high, a PWR ice condenser main steam containment penetration and a BWR Mark I torus. Six systems were ranked medium and the remaining 30 were ranked as low or low-to-medium. The two systems ranked high and two systems ranked medium, a PWR ice condenser auxiliary feedwater line and a PWR reactor coolant system loop drain line, were chosen for a quantitative stress analysis. Of the four systems analyzed, only the PWR ice condenser main steam containment penetration is judged to be significantly susceptible to the failure of a single snubber.

NUREG/CR-6028: BIGFLOW: A NUMERICAL CODE FOR SIMULATING FLOW IN VARIABLY SATURATED, HETEROGENEOUS GEOLOGIC MEDIA.Theory And User's Manual -- Version 1.1. ABABOU,R. France. BAGTZOGLOU,A.C. Center for Nuclear Waste Regulatory Analyses. June 1993. 160pp. 9307060141. CNWRA 92-026. 75572:290.

This report documents BIGFLOW 1.1, a numerical code for simulating flow in variably saturated heterogeneous geologic media. It contains the underlying mathematical and numerical models, test problems, benchmarks, and applications of the BIGFLOW code. The BIGFLOW software package is composed of a simulation and an interactive data processing code (DATAFLOW). The simulation code solves linear and nonlinear porous media flow equations based on Darcy's law, appropriately generalized to account for 3D, deterministic, or random heterogeneity. A modified Picard Scheme is used for linearizing unsaturated flow equations, and preconditioned iterative methods are used for solving the resulting matrix systems. The data processor (DATAFLOW) allows interactive data entry, manipulation, and analysis of 3D datasets. The report contains analyses of computational performance carried out using Cray-2 and Cray-Y/MP8 supercomputers. Benchmark tests include comparisons with other independently developed codes, such as PORFLOW and CMVSFS, and with analytical or semi-analytical solutions.

NUREG/CR-6029 V01: AGING ASSESSMENT OF NUCLEAR AIR-TREATMENT SYSTEM HEPA FILTERS AND ADSORBERS.Phase I. WINEGARDNER,W. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1993. 48pp. 9309210238. PNL-8594. 76486:075.

A Phase I aging assessment of high-efficiency particulate air (HEPA) filters and activated carbon gas adsorption units (adsorbers) was performed by the Pacific Northwest Laboratory (PNL) as part of the U.S. Nuclear Regulatory Commission's (NRC) Nuclear Plant Aging Research (NPAR) Program. Information concerning design features; failure experience; aging mechanisms, effects, and stressors; and surveillance and monitoring methods for these key air-treatment system components was compiled. Over 1100 failures, or 12 percent of the filter installations, were reported as part of a Department of Energy (DOE) survey. Investigators from other national laboratories have suggested that aging effects could have contributed to over 80 percent of these failures. Tensile strength tests on aged filter media specimens indicated a decrease in strength. Filter aging mechanisms range from those associated with particle loading to reactions that alter properties of sealants and gaskets. Low radioiodine decontamination factors associated with the Three Mile Island (TMI) accident were attributed to the premature aging of the carbon in the adsorbers. Mechanisms that can lead to impaired adsorber performance include oxidation as well as the loss of potentially available active sites as a result of the adsorption of pollutants. Stressors include heat, moisture, radiation, and airborne particles and contaminants.

NUREG/CR-6031: CAVITATION GUIDE FOR CONTROL VALVES. TULLIS,J.P. Tullis Engineering Consultants. April 1993. 119pp. 9306010310. 75062:052.

This guide teaches the basic fundamentals of cavitation to provide the reader with an understanding of what causes cavitation when it occurs and the potential problems cavitation can cause to a valve and piping system. The document provides guidelines for understanding how to reduce the cavitation and/or select control valves for a cavitating system. The guide provides a method for predicting the intensity of control valves and how the effect of cavitation on a system will vary with valve type, valve size, valve function, operating pressure, duration of operation and details of the piping installation. The guide defines six cavitation limits identifying cavitation intensities ranging from inception to the maximum intensity possible. The intensity of the cavitation at each limit is described including a brief discussion of how each level of cavitation influences the valve and system. Examples are included to demonstrate how to apply the method, including making both size and pressure scale effects corrections. Methods of controlling cavitation are discussed providing information on various techniques which can be used to design a new system or modify an existing one so it can operate at a desired level of cavitation.

NUREG/CR-6032: SOLIDUS AND LIQUIDUS TEMPERATURES OF CORE-CONCRETE MIXTURES. ROCHE,M.F.; LEIBOWITZ,L.; FINK,J.K.; et al. Argonne National Laboratory. June 1993. 55pp. 9306290008. ANL-93/9. 75494:001.

Solidus and liquidus temperatures were measured by a combination of differential thermal analysis and rotational viscometry for four types of concrete (limestone, limestone sand, basalt, and siliceous) and for their mixtures with urania and zirconia. The measured solidus temperatures for the urania-zirconia-concrete mixtures were significantly lower (hundreds of degrees) than those employed in the CORCON-Mod 2 thermal hydraulic code, and the measured liquidus temperatures were significantly higher (also hundreds of degrees). The liquidus temperatures for urania-zirconia-concrete mixtures containing limestone or limestone-sand concrete were generally above 2850 K, which was the upper temperature limit of our experiments. The revised solidus and liquidus temperatures are to be incorporated in the CORCON-Mod 3 thermal hydraulic code which is an integral part of the U.S. Nuclear Regulatory Commission's MELCOR Code. DTA was also employed to redetermine the calcia-urania (CaO-UO(2)) phase diagram which is required in computer programs that calculate the phase diagrams (and solidus and liquidus temperatures) of urania-zirconia-concrete systems from the phase diagrams of simpler systems.

NUREG/CR-6034: OKLAHOMA SEISMIC NETWORK.Final Report. LUZA,K.V.; LAWSON,J.E. Oklahoma, Univ. of, Norman, OK. July 1993. 42pp. 9308160178. 76122:314.

The Nemaha uplift is composed of a number of crustal blocks typically 3 to 5 miles (5 to 8 km) wide and 5 to 20 miles (8 to 32 km) long. In Oklahoma, several discontinuous uplifts, such as the Oklahoma City, Lovell, Garber, and Crescent uplifts, occur along the main axis of the Nemaha uplift. A statewide network of 12 stations records seismological data in Oklahoma. Six semipermanent seismograph stations, four radiotelemetry stations, the Oklahoma Geophysical Observatory's seismograph station, and a borehole seismograph station at the Observatory comprise the Oklahoma Geological Survey's seismic network. From January 1, 1987, through December 31, 1992, 373 earthquakes were located by the Oklahoma seismic network. The distribution of the earthquakes by state is as follows: 315 in Oklahoma, 28 in Texas, 14 in Kansas, 8 in Arkansas, 7 in Missouri, and 1 in Nebraska. Of the 352 earthquakes, 23 were reported felt. The earthquake epicentral data produces at least three seismic trends. These trends are located in north-central Oklahoma, at the eastern margin of the Anadarko basin, and in the Arkoma basin-Ouachita Mountains area.

NUREG/CR-6035: FEASIBILITY STUDY FOR IMPROVED STEADY-STATE INITIALIZATION ALGORITHMS FOR THE RELAP5 COMPUTER CODE. PAULSEN,M.P.; PETERSON,C.E.; KATSMA,K.R. Computer Simulation & Analysis, Inc. April 1993. 75pp. 9306010277. 75058:225.

A design for a new steady-state initialization method is presented that represents an improvement over the current method used in RELAP5. Current initialization methods for RELAP5 solve the transient fluid flow balance equations simulating a transient to achieve steady-state conditions. Because the transient solution is used, the initial conditions may change from the desired values requiring the use of controllers and long transient running times to obtain steady-state conditions for system problems. The new initialization method allows the user to fix thermal-hydraulic values in volumes and junctions where the conditions are best known and have the code compute the initial conditions in other areas of the system. The steady-state balance equations and solution methods are presented. The constitutive, component, and special purpose models are reviewed with respect to modifications required for the new steady-state initialization method. The requirements for user input are defined and the feasibility of the method is demonstrated with a testbed code by initializing some simple channel problems. The initialization of the sample problems using the old and the new methods are compared.

NUREG/CR-6036: INITIAL RESULTS OF THE INFLUENCE OF BIAXIAL LOADING ON FRACTURE TOUGHNESS. THEISS,T.J.; BASS,B.R.; BRYSON,J.W.; et al. Oak Ridge National Laboratory. June 1993. 94pp. 9306290012. ORNL/TM-12349. 75494:056.

A testing program to examine the influence of biaxial loads on the fracture toughness of shallow-flaw specimens under conditions prototypic of a reactor pressure vessel was begun. Existing data suggest that shallow-flaw specimens under biaxial loading will exhibit a toughness reduction compared to comparable uniaxial specimens. Quantification of this toughness reduction is the main goal of the biaxial fracture toughness program. A cruciform specimen with a two-dimensional shallow through-thickness flaw under a biaxial load ratio of 0.6:1 was used for biaxial fracture toughness testing. The critical fracture load for each specimen was approximately the same, but the uniaxial specimen withstood substantially more deformation at failure than did the biaxial specimens. Three-dimensional, elastic-plastic, finite-element posttest analyses were necessary to estimate fracture toughness. In all cases, agreement between the measured and computed load vs deformation responses was excellent. Toughness values for the cruciform specimens were compared with data from previously tested, deep- and shallow-crack specimens. Results from these tests indicate that the shallow-crack toughness increase is partially, but not totally, removed by the application of biaxial loading. However, additional data are required to solidify these conclusions. A proposed test matrix for additional uniaxial and biaxial testing is described. This report has been designated HSST Report No. 138.

NUREG/CR-6041: DISPOSAL UNIT SOURCE TERM (DUST) DATA INPUT GUIDE. SULLIVAN,T.M. Brookhaven National Laboratory. May 1993. 92pp. 9306180317. BNL-NUREG-52375. 75388:001.

Performance assessment of a low-level waste (LLW) disposal facility begins with an estimation of the rate at which radionuclides migrate out of the facility (i.e., the source term). The focus of this work is to develop a methodology for calculating the source term. In general, the source term is influenced by the radionuclide inventory, the wasteforms and containers used to dispose of the inventory, and the physical processes that lead to release from the facility (fluid flow, container degradation, wasteform leaching, and radionuclide transport). The computer code DUST (Disposal Unit Source Term) has been developed to model these processes. This document presents the models used to calculate release from a disposal facility, verification of the model, and instructions on the use of the DUST code. In addition to DUST, a preprocessor, DUSTIN, which helps the code user create input decks for DUST and a postprocessor, GRAFXT, which takes selected output files and plots them on the computer terminal have been written. Use of these codes is also described.

NUREG/CR-6043 V01: AGING ASSESSMENT OF ESSENTIAL HVAC CHILLERS USED IN NUCLEAR POWER PLANTS.Phase I. BLAHNIK,D.E.; KLEIN,R.F. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1993. 101pp. 9310120252. PNL-8614. 76741:146.

The Pacific Northwest Laboratory conducted a Phase I aging assessment of chillers used in the essential safety air-conditioning systems of nuclear power plants. Centrifugal chillers in the 75- to 750-ton refrigeration capacity range are the predominant type used. The chillers used, and air-conditioning systems served, vary in design from plant-to-plant. It is crucial to keep chiller internals very clean and to prevent the leakage of water, air, and other contaminants into the refrigerant containment system. Periodic operation on a weekly or monthly basis is necessary to remove moisture and non-condensable gases that gradually build up inside the chiller. This is especially desirable if a chiller is required to operate only as an emergency standby unit. The primary stressors and aging mechanisms that affect chillers include vibration, excessive temperatures and pressures, thermal cycling, chemical attack, and poor quality cooling water. Aging is accelerated by moisture, non-condensable gases (e.g., air), dirt, and other contamination within the refrigerant containment system, excessive start/stop cycling, and operating below the rated capacity. Aging is also accelerated by corrosion and fouling of the condenser and evaporator tubes. The principal cause of chiller failures is lack of adequate monitoring. Lack of performing scheduled maintenance and human errors also contribute to failures.

NUREG/CR-6047: CONTINUOUS SPECTROSCOPIC ANALYSIS OF VANADOUS AND VANADIC IONS. BISHOP,J.V.; DUTCHER,R.A.; FISHER,M.S.; et al. Omni Tech International, Ltd. October 1993. 28pp. 9311080119. 77072:304.

Spectroscopic methods were investigated for the determination of vanadium ions in aqueous solutions arising in the production of vanadium (II) formate and its use in the LOMI (Low Oxidation-state Metal Ion) process for the chemical decontamination of systems in nuclear power plants. In the LOMI process, a dilute solution of vanadous formate and picolinic acid is used. The vanadous formate reduces metal oxides in the scale on the equipment, causing the scale to break up and become suspended. The picolinic acid chelates these materials and makes them soluble. During the decontamination the progress is followed by analyses of the metal ions and of the radioactivity. When the values stop increasing, the decontamination is terminated. At present, it cannot be determined if the values are no longer changing due to all the scale being removed or due to the vanadous ion being spent. Infrared and ultraviolet-visible analysis were investigated as the means of analyzing for vanadium species. It was found that the complex formed by V(II) with picolinic acid could be used for colorimetric analysis for V(II) in the range of 0 - 0.011 moles/liter, which encompasses the concentration range used in the LOMI process. The findings will be used to develop an on-line instrument for continuously monitoring V(II) during decontamination.

NUREG/CR-6048: PRESSURIZED-WATER REACTOR INTERNALS AGING DEGRADATION STUDY. Phase 1. LUK,K.H. Oak Ridge National Laboratory. September 1993. 66pp. 9310120328. ORNL/TM-12371. 76740:266.

This report documents the results of a Phase 1 study on the effects of aging degradations on pressurized-water reactor (PWR) internals. Primary stressors for internals are generated by the primary coolant flow in the reactor vessel, and they include unsteady hydrodynamic forces and pump- generated pressure pulsations. Other stressors are applied loads, manufacturing processes, impurities in the coolant and exposures to fast neutron fluxes. A survey of reported aging-related failure information indicates that fatigue, stress corrosion cracking (SCC) and mechanical wear are the three major aging-related degradation mechanisms for PWR internals. Significant reported failures include thermal shield flow-induced vibration problems, SCC in guide tube support pins and core support structure bolts, fatigue-induced core baffle water-jet impingement problems and excess wear in flux thimbles. Many of the reported problems have been resolved by accepted engineering practices. Uncertainties remain in the assessment of long-term neutron irradiation effects and environmental factors in high-cycle fatigue failures. Reactor internals are examined by visual inspections and the technique is access limited. Improved inspection methods, especially one with an early failure detection capability, can enhance the safety and efficiency of reactor operations.

NUREG/CR-6049: PIPING BENCHMARK PROBLEMS FOR THE GENERAL ELECTRIC ADVANCED BOILING WATER REACTOR. BEZLER,P.; DEGRASSI,G.; BRAVERMAN,J.; et al. Brookhaven National Laboratory. August 1993. 178pp. 9309030213. BNL-NUREG-52377. 76327:127.

To satisfy the need for verification of the computer programs and modeling techniques that will be used to perform the final piping analyses for an advanced boiling water reactor standard design, three benchmark problems were developed. The problems are representative piping systems subjected to representative dynamic loads with solutions developed using the methods being proposed for analysis for the advanced reactor standard design. It will be required that the combined license holders demonstrate that their solutions to these problems are in agreement with the benchmark problem set.

NUREG/CR-6050: RADIATION EXPOSURE MONITORING AND INFORMATION TRANSMITTAL (REMIT) SYSTEM.User's Manual. CALE,R.; CLARK,T.; DIXSON,R.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). June 1993. 300pp. 9307220201. SAIC-93/1310-01. 75742:001.

The Radiation Exposure Monitoring and Information Transmittal (REMIT) system is designed to assist U.S. Nuclear Regulatory Commission (NRC) licensees in meeting the reporting requirements of the revised 10CFR20 and in agreement with the guidance contained in R.G. 8.7, Rev. 1, "Instructions for Recording and Reporting Occupational Exposure Data." REMIT is a personal computer (PC) based menu driven system that facilitates the manipulation of data base files to record and report radiation exposure information. REMIT is designed to be user-friendly and contains the full text of R.G. 8.7, Rev. 1, on-line as well as context-sensitive help throughout the program. The user can enter data directly from NRC Forms 4 or 5, REMIT allows the user to view the individual's exposure in relation to regulatory or administrative limits and alerts the user to exposures in excess of these limits. The system also provides for the calculation and summation of dose from intakes and the determination of the dose to the maximally exposed extremity for the monitoring year. REMIT can produce NRC Forms 4 and 5 in paper and electronic format and can import/export data from ASCII and data base files.

NUREG/CR-6052: METHODOLOGY FOR RELIABILITY BASED CONDITION ASSESSMENT. Application To Concrete Structures In Nuclear Plants. MORI,Y.; ELLINGWOOD,B. Johns Hopkins Univ., Baltimore, MD. * Oak Ridge National Laboratory. August 1993. 164pp. 9308200285. ORNLSUB93-SD684. 76159:048.

Structures in nuclear power plants may be exposed to aggressive environmental effects that cause their strength to decrease over an extended period of service. A major concern in evaluating the continued service of such structures is to ensure that in their current condition they are able to withstand future extreme load events during the intended service life with a level of reliability sufficient for public safety. This report describes a methodology to facilitate quantitative assessments of current and future structural reliability and performance of structures in nuclear power plants. This methodology takes into account the nature of past and future loads, and randomness in strength and in degradation resulting from environmental factors. An adaptive Monte Carlo simulation procedure is used to evaluate time-dependent system reliability. The time-dependent reliability is sensitive to the time-varying load characteristics and to the choice of initial strength and strength degradation models but not to correlation in component strengths within a system. Inspection/maintenance strategies are identified that minimize the expected future costs of keeping the failure probability of a structure at or below an established target failure probability during its anticipated service period.

NUREG/CR-6054 DRF FC: ESTIMATING PRESSURIZED WATER REACTOR DECOMMISSIONING COSTS.A User's Manual For The PWR Cost Estimating Computer Program (CECP) Software.Draft Report For Comment. BIERSCHBACH,M. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1993. 150pp. 9311080114. PNL-8497. 77072:163.

With the issuance of the Decommissioning Rule ,(July 27, 1988), nuclear power plant licensees are required to submit to the U.S. Nuclear Regulatory Commission (NRC) for review, decommissioning plans and cost estimates. This user's manual and the accompanying Cost Estimating Computer Program (CECP) software provide a cost-calculating methodology to the NRC staff that will assist them in assessing the adequacy of the licensee submittals. The CECP, designed to be used on a personal computer, provides estimates for the cost of decommissioning PWR power stations to the point of license termination. Such cost estimates include component, piping, and equipment removal costs; packaging costs; decontamination costs; transportation costs; burial costs; and manpower costs. In addition to costs, the CECP also calculates burial volumes, person-hours, crew-hours, and exposure person-hours associated with decommissioning.

NUREG/CR-6056: A FRAMEWORK FOR THE ASSESSMENT OF SEVERE ACCIDENT MANAGEMENT STRATEGIES. KASTENBERG,W.E.; APOSTOLAKIS,G.; DHIR,V.K.; et al. California, Univ. of, Los Angeles, CA. September 1993. 350pp. 9310130038. 76743:231.

Severe accident management can be defined as the use of existing and/or alternative resources, systems and actors to prevent or mitigate a core-melt accident. For each accident sequence and each combination of severe accident management strategies, there may be several options available to the operator, and each involves phenomenological and operational considerations regarding uncertainty. Operational uncertainties include operator, system and instrumentation behavior during an accident. A framework based on decision trees and influence diagrams has been developed which incorporates such criteria as feasibility, effectiveness, and adverse effects, for evaluating potential severe accident management strategies. The framework is also capable of propagating both data and model uncertainty. It is applied to several potential strategies including PWR cavity flooding, BWR drywell flooding, PWR depressurization, and PWR feed and bleed.

NUREG/CR-6058: VIRGINIA REGIONAL SEISMIC NETWORK.Final Report (1986 -1992). BOLLINGER,G.A.; SIBOL,M.S.; CHAPMAN,M.C.; et al. Virginia Polytechnic Institute & State Univ., Blacksburg, VA. July 1993. 115pp. 9308160175. 76122:084.

In 1986, the Virginia Regional Seismic Network was one of the few fully calibrated digital seismic networks in the United States. Continued operation has resulted in the archival of signals from 2000+ local, regional and teleseismic sources. Seismotectonic studies of the central Virginia seismic zone showed the activity in the western part to be related to a large antiformal structure while seismicity in the eastern portion is associated spatially with dike swarms. The eastern Tennessee seismic zone extends over a 300x50 km area and is the result of a compressive stress field acting at the intersection between two large crustal blocks. Hydroseismicity, which proposes a significant role for meteoric water in intraplate seismogenesis, found support in the observation of common cyclicities between streamflow and earthquake strain data. Seismic hazard studies have provided the following results: 1) Damage areas in the eastern United States are three to five times larger than those observed in the west; 2) Judged solely on the basis of cataloged earthquake recurrence rates, the next major shock in the southeast region will probably occur outside the Charleston, South Carolina, area; and 3) Investigations yielded necessary hazard parameters (for example, maximum magnitudes) for several sites in the southeast. Basic to these investigations was the development and maintenance of several seismological data bases.

NUREG/CR-6059: MACCS VERSION 1.5.11.1: A MAINTENANCE RELEASE OF THE CODE. CHANIN,D.; FOSTER,J.; et al. Technadyne Engineering Consultants, Inc. ROLLSTIN,J. GRAM, Inc. October 1993. 46pp. 9311150030. SAND92-2146. 77210:190.

A new version of the MACCS code (version 1.5.11.1) has been developed by Sandia National Laboratories under sponsorship of the U.S. Nuclear Regulatory Commission. MACCS was developed to support evaluations of the off-site consequences from hypothetical severe accidents at commercial power plants. MACCS is the only current public domain code in the U.S. which embodies all of the following modeling capabilities: (1) weather sampling using a year of recorded weather data; (2) mitigative actions such as evacuation, sheltering, relocation, decontamination, and interdiction; (3) economic costs of mitigative actions; (4) cloudshine, groundshine, and inhalation pathways as well as food and water ingestion; (5) calculation of both individual and societal doses to various organs; and (6) calculation of both acute (non-stochastic) and latent (stochastic) health effects and risks of health effects. All of the consequence measures may be generated in the form of a complementary cumulative distribution function (CCDF). The current version implements a revised cancer model consistent with recent reports such as BEIR V and ICRP 60. In addition, a number of error corrections and portability enhancements have been implemented. This report describes only the changes made in creating the new version. Users of the code will need to obtain the code's original documentation, NUREG/CR-4691.

NUREG/CR-6060: HYDROGEN MIXING STUDIES (HMS) ASSESSMENT MANUAL. LAM,K.L.; WILSON,T.L. Los Alamos National Laboratory. TRAVIS,J.R. Science Applications International Corp. (formerly Science Applications, Inc.). June 1993. 94pp. 9308160049. LA-12593-M. 76121:148.

This report documents some calculations performed to assess the Hydrogen Mixing Studies (HMS) code. Results are presented first for some analytical test problems, including laminar flow and mass diffusion. The von Karman vortex street problem and the Sandia FLAME Facility and Heis. Dampf Reaktor (HDR) containment facility test problems are then discussed. For the analytical problems, the code gave results that agree exceptionally well with the analytical solutions Calculations for the von Karman vortex street problem were performed at selected Reynolds numbers for several obstacle types. The computed flow patterns agree well with experimental observations--specifically the occurrence of a vortex street (double row of vortices) above a critical Reynolds number. Calculations for the von Karman vortex street problem were performed at selected Reynolds numbers for several obstacle types. The computed flow patterns agree well with experimental observations--specifically the occurrence of a vortex street (double row of vortices) above a critical Reynolds number The last assessment problem involves modeling the experiment T31.5. The experiment was carried out in the HDR containment building which is a large, multi-compartment facility (11 300 m(3) free volume in 72 compartments). In the experiment, a steam-water mixture was first injected into the containment to simulate a large-break blowdown of a pressure vessel, and then superheated steam was injected that was followed by a release of helium-hydrogen light gas. The calculated results (pressure, temperature, and gas concentrations) agree reasonably well with the experimental data.

NUREG/CR-6061: DETERMINATION OF THE BIAS IN LOFT FUEL PEAK CLADDING TEMPERATURE DATA FROM THE BLOWDOWN PHASE OF LARGE-BREAK LOCA EXPERIMENTS. BERTA,V.T.; HANSON,R.G.; JOHNSEN,G.W.; et al. Idaho National Engineering Laboratory. May 1993. 82pp. 9306210362. EGG-2610. 75403:140.

Data from the Loss-of-Fluid Test (LOFT) Program help quantify the margin of safety inherent in pressurized water reactors during postulated loss-of-coolant accidents (LOCAs). This report analyzes how well externally-mounted fuel rod cladding surface thermocouples in LOFT accurately reflected actual cladding temperature during large-break LOCA experiments. The analysis shows that there can be a significant difference (referred to as bias) between the surface-mounted thermocouple reading and the actual cladding temperature, and that the magnitude of this bias depends on the rate of heat transfer between the fuel rod cladding and coolant. Further, it is shown that, in terms of peak cladding temperature recorded during LOFT large-break LOCA experiments, the mean bias is 11.4 $\pm$ 16.2 K (20.5 $\pm$ 29.2 degrees F). The best-estimate value of peak cladding temperature for LOFT LP-02-6 is 1104.8 K. The best-estimate peak cladding temperature for LOFT LP-LB-1 is 1284.0 K.

for LOFT LP-02-6 is 1104.8 K. The best-estimate peak cladding temperature for LOFT LP-LB-1 is 1284.0 K.

NUREG/CR-6062: PERFORMANCE OF PORTABLE RADIATION SURVEY INSTRUMENTS. EISENHOWER,E.; WELCH,L.; WIBLIN,C. Advanced Systems Technology, Inc. December 1993. 35pp. 9401060236. 77686:271.

This report examines alleged and documented deficiencies in the performance and the calibration of existing portable radiation survey instruments. This report also examines a limited number of reported overexposures and excessive exposures attributed to instrumentation or calibration problems. The high failure rates in performance testing of a limited number of instruments indicate further testing is needed to demonstrate which instruments are acceptable and for what application. Further, the adequacy of calibration is not demonstrated at this time as many calibrations are performed by nonaccredited calibration laboratories. A review of the regulatory requirements and practices of the NRC and Agreement States regarding the use of existing performance standards such as ANSI N42-17A-1988 and the use of accredited calibration laboratories demonstrates that (1) the regulatory programs do not require compliance with existing industry standards; and (2) instruments are generally not required to be calibrated by accredited laboratories. Options are recommended that might encourage the use of industry performance standards and calibration techniques.

NUREG/CR-6065: SYSTEMS ANALYSIS OF THE CANDU 3 REACTOR. WOLFGONG,J.R.; LINN,M.A.; WRIGHT,A.L.; et al. Oak Ridge National Laboratory. July 1993. 334pp. 9308160298. ORNL/TM-12396. 76118:001.

This report presents the results of a systems failure analysis study of the CANDU 3 reactor design; the study was performed for the U.S. Nuclear Regulatory Commission. As part of the study a review of the CANDU 3 design documentation was performed, a plant assessment methodology was developed, representative plant initiating events were identified for detailed analysis, and a plant assessment was performed. The results of the plant assessment included classification of the CANDU 3 event sequences that were analyzed, determination of CANDU 3 systems that are "significant to safety," and identification of key operator actions for the analyzed events.

NUREG/CR-6070: MODELING APPROACHES FOR CONCRETE BARRIERS USED IN LOW-LEVEL WASTE DISPOSAL. SEITZ,R.R.; WALTON,J.C. EG&G Idaho, Inc. November 1993. 35pp. 9312220140. EGG-2701. 77543:208.

A series of three NUREGs and several papers addressing different aspects of modeling performance of concrete barriers for low-level radioactive waste disposal have been prepared previously for the Concrete Barriers Research Project. This document integrates the information from the previous documents into a general summary of models and approaches that can be used in performance assessments of concrete barriers. Models for concrete degradation, flow, and transport through cracked concrete barriers are discussed. The models for flow and transport assume that cracks have occurred and thus should only be used for later times in simulations after fully penetrating cracks are formed. Most of the models have been implemented in a computer code, CEMENT, that was developed concurrently with this document. User documentation for CEMENT is provided separate from this report. To avoid duplication, the reader is referred to the three previous NUREGs for detailed discussions of each of the mathematical models. Some additional information that was not presented in the previous documents is also included. Sections discussing lessons learned from applications to actual performance assessments of low-level waste disposal facilities are provided. Sensitive design parameters are emphasized to identify critical areas of performance for concrete barriers, and potential problems in performance assessments are also identified and discussed.

NUREG/CR-6071: IMPACT OF ENDF/B-VI CROSS-SECTION DATA ON H.B. ROBINSON CYCLE 9 DOSIMETRY CALCULATIONS. WILLIAMS,M.L.; ASGARI,M. Louisiana State Univ., Baton Rouge, LA. KAM,F.B. Oak Ridge National Laboratory. October 1993. 31pp. 9311080105. ORNL/TM-12406. 77069:275.

Dosimeters that were removed from the H.B. Robinson reactor following Cycle 9 were analyzed and compared with calculated results in an earlier study. This work updates the calculation using recently available ENDF/B-VI data in order to assess advantages to using the newer cross sections in reactor pressure vessel fluence calculations. A comparison is also made to determine the impact of various cross-section libraries on computed dosimeter activities. Significant improvements are obtained with the ENDF/B-VI cross sections. Other factors, such as differences in group structures of multigroup libraries, may also affect the calculated dosimeter activities.

NUREG/CR-6072: EXPERIMENTAL STUDY ON THE COMBUSTION BEHAVIOR OF HYDROGEN-AIR MIXTURES WITH TURBULENT JET IGNITION AT LARGE SCALE. DOROFEEV,S.B.; BEZMELNITSIN,A.; EFIMENKO,A.A.; et al. Russian Research Center (Kurchatov Institute). June 1993. 83pp. 9308160272. RRCKI-80-05/3. 76117:272.

This report describes research carried out in the KOPER facility on spontaneous detonation ignition in hydrogen-air mixtures by turbulent jet ignition. The KOPER facility is a semi-confined volume of 47.7 m(3) and consists of a steel canyon and a robust frame placed above it. The frame sides are sealed with thin polyethylene sheet. A "jet" chamber of 0.55 m(3), located on the bottom of the canyon was used to produce a jet of hot gases, which was vented into the hydrogen-air mixture. The effects of three variables were investigated: hydrogen concentration (18-30% vol.); jet orifice diameter (100-400 mm); and the composition of combustion products in the turbulent jet (by varying the hydrogen mole fraction in the "jet"-chamber from 25 to 50% vol.). The possibility of initiation of turbulent combustion and local detonation was demonstrated. Local detonation develops after a delay of 10-25 ms from ignition. For spontaneous detonation initiation, the minimum hydrogen concentration is within the range of 20 to 25% vol., and the minimum jet orifice diameter lies in the range of 100 to 200 mm for the KOPER facility. A minimum ratio of turbulent jet size L and mixture detonation cell λ, L/λ = 12-13 is required for detonation initiation which is supported by other type of turbulent jet initiation experiments (closed volume and continuous venting) and by theoretical analysis.

NUREG/CR-6073: LYSIMETER LITERATURE REVIEW. ROGERS,R.D.; MCCONNELL,J.W. EG&G Idaho, Inc. August 1993. 75pp. 9309210201. EGG-2706. 76484:257.

Many reports have been published concerning the use of lysimeters to obtain data on the performance of buried radioactive waste. This document presents a review of some of those reports. This review includes lysimeter studies using radioactive waste forms at Savannah River Site, Hanford Site, Argonne National Laboratory, and Oak Ridge National Laboratory; radionuclide tracer studies at Whiteshell Nuclear Research Establishment and Los Alamos National Laboratory; and water movement studies at the Nuclear Regulatory Commission's Beltsville, Maryland site, at the Hanford Site, and at New Mexico State University. The tests, results, and conclusions of each report are summarized, and conclusions concerning lysimeter technology are presented from an overall analysis of the literature.

NUREG/CR-6078: ANALYSIS OF CRACK INITIATION AND GROWTH IN THE HIGH LEVEL VIBRATION TEST AT TADOTSU. KASSIR,M.K.; PARK,Y.J.; HOFMAYER,C.H.; et al. Brookhaven National Laboratory. August 1993. 82pp. 9309210185. BNL-NUREG-52383. 76484:075.

The High Level Vibration Test data are used to assess the accuracy and usefulness of current engineering methodologies for predicting crack initiation and growth in a cast stainless steel pipe elbow under complex, large amplitude loading. The data were obtained by testing at room temperature a large scale modified model of one loop of a PWR primary coolant system at the Tadotsu Engineering Laboratory in Japan. Fatigue crack initiation time is reasonably predicted by applying a modified local strain approach (Coffin-Mason-Goodman equation) in conjunction with Miner's rule of cumulative damage. Three fracture mechanics methodologies are applied to investigate the crack growth behavior observed in the hot leg of the model. These are: the ΔK methodology (Paris law), ΔJ concepts and a recently developed limit load stress-range criterion. The report includes a discussion on the pros and cons of the analysis involved in each of the methods, the role played by the key parameters influencing the formulation and a comparison of the results with the actual crack growth behavior observed in the vibration test program. Some conclusions and recommendations for improvement of the methodologies are also provided.

NUREG/CR-6079: SEISMOLOGICAL INVESTIGATION OF EARTHQUAKES IN THE NEW MADRID SEISMIC ZONE.Final Report.September 1986 - December 1992. HERRMANN,R.B.; NGUYEN,B. St. Louis Univ., St. Louis, MO. August 1993. 75pp. 9309030148. 76325:084.

Earthquake activity in the New Madrid Seismic Zone had been monitored by regional seismic networks since 1975. During this time period, over 3700 earthquakes have been located within the region bounded by latitudes 35 degrees-39 degrees N and longitudes 87 degrees-92 degrees W. Most of these earthquakes occur within a 1.5 degree x 2 degree zone centered on the Missouri Bootheel. Source parameters of larger earthquakes in the zone and in eastern North America are determined using surface-wave spectral amplitudes and broadband waveforms for the purpose of determining the focal mechanism, source depth and seismic moment. Waveform modeling of broadband data is shown to be a powerful tool in defining these source parameters when used complementary with regional seismic network data, and in addition, in verifying the correctness of previously published focal mechanism solutions.

NUREG/CR-6080: REPLACEMENT ENERGY,CAPACITY, AND RELIABILITY COSTS FOR PERMANENT NUCLEAR REACTOR SHUTDOWNS. VANKUIKEN,J.C.; BUEHRING,W.A.; HAMILTON,S.; et al. Argonne National Laboratory. October 1993. 37pp. 9311180062. ANL-93/19. 77231:267.

Average replacement power costs are estimated for potential permanent shutdowns of nuclear electricity-generating units. Replacement power costs are considered to include replacement energy, capacity, and reliability cost components. These estimates were developed to assist the U.S. Nuclear Regulatory Commission in evaluating regulatory issues that potentially affect changes in serious reactor accident frequencies. Cost estimates were derived from long-term production-cost and capacity expansion simulations of pooled utility-system operations. Factors that affect replacement power cost, such as load growth, replacement sources of generation, and capital costs for replacement capacity, were treated in the analysis. Costs are presented for a representative reactor and for selected subcategories of reactors, based on estimates for 112 individual reactors.

NUREG/CR-6081: ENHANCED REMOVAL OF RADIOACTIVE PARTICLES BY FLUOROCARBON SURFACTANT SOLUTIONS. KAISER,R.; HARLING,O.K. Entropic Systems, Inc. August 1993. 97pp. 9309210192. 76484:156.

The proposed research addressed the application of ESI's particle removal process to the non-destructive decontamination of nuclear equipment. The cleaning medium used in this process is a solution of a high molecular weight fluorocarbon surfactant in an inert perfluorinated liquid which results in enhanced particle removal. The perfluorinated liquids of interest, which are recycled in the process, are nontoxic, nonflammable, and environmentally compatible, and do not present a hazard to the ozone layer. The information obtained in the Phase I program indicated that the proposed ESI process is technically effective and economically attractive. The fluorocarbon surfactant solutions used as working media in the ESI process survived exposure of up to 10 Mrad doses of gamma rays, and are considered sufficiently radiation resistant for the proposed process. Ultrasonic cleaning in perfluorinated surfactant solutions was found to be an effective method of removing radioactive iron (Fe 59) oxide particles from contaminated test pieces. Radioactive particles suspended in the process liquids could be quantitatively removed by filtration through a 0.1 um membrane filter. Projected economics indicate a pre-tax pay back time of 1 month for a commercial scale system.

NUREG/CR-6082: DATA COMMUNICATIONS. PRECKSHOT,G.G. Lawrence Livermore National Laboratory. August 1993. 96pp. 9309030171. UCRL-ID-114567. 76326:219.

The purpose of this paper is to recommend regulatory guidance for reviewers examining computer communicatic 1 systems used in nuclear power plants. The recommendatic .s cover three areas important to these communications systems: system design, communication protocols, and communication media. The first area, system design, considers three aspects of system design--questions about architecture, specific risky design elements or omissions to look f r in designs being reviewed, and recommendations for multiplexed data communication systems used in safety systems. The second area reviews pertinent aspects of communication protocol design and makes recommendations for newly designed protocols or the selection of existing protocols for safety system, information display, and non-safety control system use. The third area covers communication media selection, which differs significantly from traditional wire and cable. The recommendations for communication media extend or enhance the concerns of published IEEE standards about three subjects: data rate, imported hazards and maintainability.

NUREG/CR-6083: REVIEWING REAL-TIME PERFORMANCE OF NUCLEAR REACTOR SAFETY SYSTEMS. PRECKSHOT,G.G. Lawrence Livermore National Laboratory. August 1993. 88pp. 9309030112. UCRL-ID-114565. 76325:001.

The purpose of this paper is to recommend regulatory guidance for reviewers examining real-time performance of computer-based safety systems used in nuclear power plants. Three areas of guidance are covered in this report. The first area covers how to determine if, when, and what prototypes should be required of developers to make a convincing demonstration that specific problems have been solved or that performance goals have been met. The second area has recommendations for timing analyses that will prove that the real-time system will meet its safety-imposed deadlines. The third area has descriptions of means for assessing expected or actual real-time performance before, during, and after development is completed. To ensure that the delivered real-time software product meets performance goals, the paper recommends certain types of code-execution and communications scheduling. Technical background is provided in the appendix on methods of timing analysis, scheduling real-time computations, prototyping, real-time software development approaches, modeling and measurement, and real-time operating systems.

NUREG/CR-6084: VALUE-IMPACT ANALYSIS OF GENERIC ISSUE 143, "AVAILABILITY OF HEATING, VENTILATION, AIR CONDITIONING (HVAC) AND CHILLED WATER SYSTEMS." DALING,P.M.; MARLER,J.E.; VO,T.V. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1993. 700pp. 9312170083. PNL-8750. 77513:001.

This study evaluates the values and impacts associated with potential resolutions to Generic Issue 143, "Availability of HVAC and Chilled Water Systems." The study identifies vulnerabilities related to failures of HVAC, chilled water, and room cooling systems, develops estimates of room heatup rates and safety-related equipment vulnerabilities, develops estimates of the core damage frequencies and public risks associated with failures of these systems, develops three proposed resolution strategies to this generic issue, and performs a value/impact analysis of the proposed resolutions. Existing probabilistic risk assessments (PRAs) for four representative plants, including one plant from each vendor, form the basis for the core damage frequency and public risk calculations. Both internal and external events were considered. It was concluded that all three proposed resolution strategies exceed the $1,000/person-rem cost-effectiveness ratio. Additional evaluations were performed to develop "generic" insights on potential design- and configuration-related vulnerabilities and potential high-frequency accident sequences. It was concluded that some high-frequency sequences exist but are plant-specific in nature or have been resolved through hardware and/or operational changes. The plant-specific Individual Plant Examinations (IPEs) are an effective vehicle for identification and resolution of these plant-specific anomalies and hardware configurations.

NUREG/CR-6085: UNITED STATES SEISMOGRAPHIC NETWORK. BULAND,R. Interior, Dept. of, Geological Survey. September 1993. 83pp. 9310120339. 76741:028.

The concept of a United States National Seismograph Network (USNSN) dates back nearly 30 years. The idea was revived several times over the decades, but never funded. For example, a national network was proposed and discussed at great length in the so called Bolt Report (U.S. Earthquake Observatories: Recommendations for a New National Network, National Academy Press, Washington, D.C., 1980, 122 pp). From the beginning, a national network was viewed as augmenting and complementing the relatively dense, predominantly short-period vertical coverage of selected areas provided by the Regional Seismograph Networks (RSN's) with a sparse, well-distributed network of three-component, observatory quality, permanent stations. The opportunity finally to begin developing a national network arose in 1986 with discussions between the U.S. Geological Survey (USGS) and the Nuclear Regulatory Commission (NRC). Under the agreement signed in 1987, the NRC has provided $5 M in new funding for capital equipment (over the period 1987-1992) and the USGS has provided personnel and facilities to develop, deploy, and operate the network. Because the NRC funding was earmarked for the eastern United States, new USNSN station deployments are mostly east of 105 degrees West longitude while the network in the western United States is mostly made up of cooperating stations (stations meeting USNSN design goals, but deployed and operated by other institutions which provide a logical extension to the USNSN).

NUREG/CR-6090: THE PROGRAMMABLE LOGIC CONTROLLER AND ITS APPLICATION IN NUCLEAR REACTOR SYSTEMS. PALOMAR,J.; WYMAN,R. Lawrence Livermore National Laboratory. September 1993. 96pp. 9310120250. UCRL-ID-112900. 76740:170.

This document provides recommendations to guide reviewers in the application of Programmable Logic Controllers (PLCs) to the control, monitoring and protection of nuclear reactors. The first topics addressed are system-level design issues, specifically including safety. The document then discusses concerns about the PLC manufacturing organization and the protection system engineering organization. Supplementing this document are two appendices. Appendix A summarizes PLC characteristics. Specifically addressed are those characteristics that make the PLC more suitable for emergency shutdown systems than other electrical/electronic-based systems, as well as characteristics that improve reliability of a system. Also covered are PLC characteristics that may create an unsafe operating environment. Appendix B provides an overview of the use of programmable logic controllers in emergency shutdown systems. The intent is to familiarize the reader with the design, development, test, and maintenance phases of applying a PLC to an ESD system. Each phase is described in detail and information pertinent to the application of a PLC is pointed out.

NUREG/CR-6098: LOADING RATE EFFECTS ON STRENGTH AND FRACTURE TOUGHNESS OF PIPE STEELS USED IN TASK 1 OF THE IPIRG PROGRAM. MARSCHALL,C.W.; LANDOW,M.P.; WILKOWSKI,G.M. Battelle Memorial Institute, Columbus Laboratories. October 1993. 129pp. 9311080142. BMI-2175. 77084:041.

Material characterization tests were conducted on laboratory specimens machined from pipes to determine the effect of dynamic loading (i.e., rates comparable to those for high amplitude seismic events) on tensile properties and fracture resistance at 288 degrees C (550 degrees F). Specimens were fabricated from seven different pipes, including carbon steels and stainless steels (both base metal and weld metal), which were to be subjected to full-scale pipe tests in IPIRG Task 1.0. For the stainless steels tested at 288 degrees C (550 degrees F), tensile strength was unchanged, while yield strength and fracture resistance were increased. The increase in fracture resistance was modest for the wrought base metals and substantial for the weld metal and the cast base metal. The carbon steels tested were sensitive to dynamic strain aging, and hence the strength and toughness was affected by both temperature and strain rate effects. The carbon steel base metal and welds exhibited ultimate tensile strength values at 288 degrees C (550 degrees F) that were greater than at room temperature. Furthermore, the ultimate tensile strength at 288 degrees C (550 degrees F) was lowered significantly by increased strain rate and, in the carbon steel base metals, increased strain rate also lowered the fracture resistance, substantially in the base metal of one pipe.

NUREG/CR-6101: SOFTWARE RELIABILITY AND SAFETY IN NUCLEAR REACTOR PROTECTION SYSTEMS. LAWRENCE,J.D. Lawrence Livermore National Laboratory. November 1993. 150pp. 9312220167. 77544:011.

Planning the development, use and regulation of computer systems in nuclear reactor protection systems in such a way as to enhance reliability and safety is a complex issue. This report is one of a series of reports from the Computer Safety and Reliability Group, Lawrence Livermore National Laboratory, that investigates different aspects of computer software in reactor protection systems. There are two central themes in the report. First software considerations cannot be fully understood in isolation from computer hardware and application considerations. Second, the process of engineering reliability and safety into a computer system requires activities to be carried out throughout the software life cycle. The report discusses the many activities that can be carried out during the software life cycle to improve the safety and reliability of the resulting product. The viewpoint is primarily that of the assessor, or auditor.

NUREG/CR-6108: SPHERICAL DIFFUSION OF TRITIUM FROM A POINT OF RELEASE IN A UNIFORM UNSATURATED SOIL.A Deterministic Model For Tritium Migration In An Arid Disposal Site. SMILES,D.E.; GARDNER,W.R.; SCHULZ,R.K. California, Univ. of, Berkeley, CA. October 1993. 38pp. 9311080128. 77084:001.

This report presents a three-dimensional model for tritium migration in an arid waste disposal site. When tritiated water is released at a point in a uniform and relatively dry soil it redistributes in both the liquid and vapor phases. The flux density of tritium in each phase is of the same order of magnitude however so tritium redistribution is modeled as if transfer occurs "in parallel" in the liquid and vapor phases. The approach we describe uses the diffusion equation cast in radial (spherical) coordinates and takes into account radioactive decay. It permits calculation of radial profiles of tritium concentration, within and external to a sphere of released solution. We assume the concentration within this sphere initially to be uniform. The solution also predicts attenuation and rate of advance of the maximum of tritium concentration as it advances in the soil. With deep disposal in a desert soil, the model predicts that tritium migration will be short range, with a maximum of a few meters.

NUREG/CR-6111: INTEGRATED SYSTEMS ANALYSIS OF THE PIUS REACTOR. FULLWOOD,F.; KROEGER,P.G.; HIGGINS,J.; et al. Brookhaven National Laboratory. November 1993. 450pp. 9312160278. BNL-NUREG-52393. 77505:198.

Results are presented of a systems failure analysis of the PIUS plant systems that are used during normal reactor operation and postulated accidents. This study was performed to provide the NRC with an understanding of the behavior of the plant. The study applied two diverse failure identification methods, Failure Modes Effects & Criticality Analysis (FMECA) and Hazards & Operability (HAZOP) to the plant systems, supported by several deterministic analyses. Conventional PRA methods were also used along with a scheme for classifying events by initiator frequency and combinations of failures. Principal results of this study are: (a) an extensive listing of potential event sequences, grouped in categories that can be used by the NRC; (b) identification of support systems that are important to safety; and (c) identification of key operator actions.

NUREG/CR-6113: CLASS 1E DIGITAL SYSTEMS STUDIES. HECHT,H.; TAI,A.T.; TSO,K.S. SoHaR, Inc. October 1993. 220pp. 9311080138. 77083:139.

This document is furnished as part of the effort to develop NRC Class 1E Digital Computer Systems Guidelines which is Task 8 of USAF Rome Laboratories Contract F30602-89-D-0100. The report addresses four major topics, namely, computer programming languages, software design and development, software testing and fault tolerance and fault avoidance. The topics are intended as stepping stones leading to a Draft Regulatory Guide document. As part of this task a small scale survey of software fault avoidance and fault tolerance practices was conducted among vendors of nuclear safety related systems and among agencies that develop software for other applications demanding very high reliability. The findings of the present report are in part based on the survey and in part on review of software literature relating to nuclear and other critical installations, as well as on the authors' experience in these areas.

NUREG/CR-6114 V01: APPLICATION OF AN INFILTRATION EVALUATION METHODOLOGY TO A HYPOTHETICAL LOW-LEVEL WASTE DISPOSAL FACILITY. MEYER,P.D. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1993. 42pp. 9401140021. PNL-8842. 77797:138.

An analysis of infiltration and percolation at a hypothetical low-level waste (LLW) disposal facility was carried out. The analysis was intended to illustrate general issues of concern assessing the performance of LLW disposal facilities. Among the processes considered in the analysis were precipitation, runoff, infiltration, evaporation, transpiration, and redistribution. The hypothetical facility was located in a humid environment characterized by frequent and often intense precipitation events. The facility consisted of a series of concrete vaults topped by a multilayer cover. Cover features included a sloping soil surface to promote runoff, plant growth to minimize erosion and promote transpiration, a sloping clay layer, and a sloping capillary barrier. The analysis within the root zone was carried out using a one-dimensional, transient simulation of water flow. Below the root zone, the analysis was primarily two-dimensional and steady-state. Results of the situations illustrated the limited value of daily precipitation data. For the humid site studied, hourly rainfall data provided significantly better estimates of the water balance. Results also demonstrated the importance of transpiration in removing water from the soil column, implying a need for accurate models of plant growth and water utilization. In addition, the amount of water predicted to percolate below the root zone was often less than the amount required to keep the clay barrier layer fully saturated, even in the relatively wet environment studied. This could be a concern if the clay were subject to shrinking under unsaturated conditions. The two-dimensional simulations showed that the sloping clay barrier diverted 75 percent of the water reaching it. The sloping capillary barrier, in contrast, diverted more than 99.99 percent of the water reaching it. Performance of the capillary barrier, however, was shown to vary significantly with the hydraulic properties of the two materials of which it is composed. Predicting performance simply by inspecting the water retention and hydraulic conductivity functions was difficult. An analytical expression was presented that can be used to estimate capillary barrier performance and to determine appropriate materials for construction.

NUREG/CR-6117: NEUTRON SPECTRA AT DIFFERENT HIGH FLUX ISOTOPE REACTOR (HFIR) PRESSURE VESSEL SURVEILLANCE LOCATIONS. REMEC,I. Josef Sefan Institute. KAM,F.B. Oak Ridge National Laboratory. December 1993. 137pp. 9401140025. ORNL/TM-12484. 77797:177.

This project addresses the potential problem of radiation embrittlement of reactor pressure vessel (RPV) supports. Surveillance specimens irradiated at the High Flux Isotope Reactor (HFIR) at relatively low neutron flux levels (about 1.5E+8 cm(-2).s(-1)) and low temperature (about 50 degrees C) showed embrittlement more rapidly than expected. Commercial power reactors have similar flux levels and temperatures at the vessel support structures. The purposes of this work are to provide the neutron fluence spectra data that are needed to evaluate previously measured mechanical property changes in the HFIR, to explain the discrepancies in neutron flux levels between the nickel dosimeters and two other dosimeters, neptunium and beryllium, and to address any questions or peculiarities of the HFIR reactor environment.

NUREG/CR-6118: ASSESSMENT OF THE EFFECTIVENESS OF THE LEU REFORM RULE AND ITS IMPLEMENTATION. MORAN,B.W.; NATIONS,J.O. Martin Marietta Energy Systems, Inc. HAMMOND,G.A. 21st Century Industries, Inc. November 1993. 34pp. 9312070227. K/NSP-117. 77349:295.

The U.S. Nuclear Regulatory Commission (NRC) amended its material control and accounting (MC&A) requirements in 1985 for licensees possessing and using special nuclear material (SNM) of low strategic significance in quantities larger than one effective kilogram (kg). The goal of the Low-Enriched Uranium (LEU) Reform Rule (i.E., 10CFR 74.31) was to establish MC&A requirements for the LEU licensees at a level consistent with the safeguards risk associated with the relatively low strategic importance of such material. The amended requirements were written in a performance-oriented manner, rather than a prescriptive one, in an effort to allow the licensees the opportunity to choose the most cost-effective means of satisfying the requirements. The LEU Reform Rule was implemented in January 1988 and the fuel cycle facilities have had sufficient experience in implementing the rule to allow a meaningful review of its effectiveness. This document provides technical analysis and recommendations to assist the NRC in making a determination if the rule is achieving its intended purpose, and if not, to make the necessary changes to accomplish this.

NUREG/GR-0005 V02 P1: RISK-BASED INSPECTION-DEVELOPMENT OF GUIDELINES.Light Water Reactor (LWR) Nuclear Power Plant Components. * American Society of Mechanical Engineers. July 1993. 173pp. 9308200281. CRTD-VOL.20-2. 76159:209.

Effective inservice inspection programs can play a significant role in minimizing equipment and structural failures. Most of the current inservice inspection programs for light water reactor (LWR) nuclear power plant components are based on experience and engineers' qualitative judgment. These programs include only an implicit consideration of risk, which combines the probability of failure of a component under its operation and loading conditions and the consequences of such failure, if it occurs. This document recommends appropriate methods for establishing a risk-based inspection program for LWR nuclear power plant components. The process has been built from a general methodology (Volume 1) and has been expanded to involve five major steps: defining the system; evaluating qualitative risk assessment results; using this and information from plant probabilistic risk assessments to perform a quantitative risk analysis; selecting target failure probabilities; and developing an inspection program for components using economic decision analysis and structural reliability assessment methods. Included: extensive bibliography. Companion Volume 2 - Part 2 document will recommend risk-based inspection program for consideration by Section XI of the ASME Boiler and Pressure Vessel Code.

NUREG/GR-0006: DEPOSITION: SOFTWARE TO CALCULATE PARTICLE PENETRATION THROUGH AEROSOL TRANSPORT SYSTEMS.Final Report. ANAND,N.K.; MCFARLAND,A.R.; WONG,F.S.; et al. Texas A&M Univ., College Station, TX. April 1993. 45pp. 9305100008. 74858:223.

User-friendly software (DEPOSITION 2.0) has been developed which permits characterization of aerosol particle losses in transport systems. The sub-models which comprise the DEPOSITION code are presented and the limitations of these sub-models are noted. These sub-models have all been previously published in the peer-reviewed literature. The software can be used to determine the penetration of aerosol through existing transport systems; it will provide the optimal tube diameter for a transport system operated at a given flow rate and at a given particle size; it will provide a value for the maximum penetration for a transport system that would connect two points in three-dimensional space; and, it will provide tables of data and create output files for parametric studies on the effects of varying particle size, flow rate and tube diameter. Use of this software for specific examples is given herewith in an Appendix. Reference to this software is included in NRC Regulatory Guide 8.25 (1992) where it is considered to be an acceptable method for calculating the penetration of particles through sampling systems.

NUREG/GR-0009: STEPWISE INTEGRAL SCALING METHOD AND ITS APPLICATION TO SEVERE ACCIDENT PHENOMENA. ISHII,M.; ZHANG,G. Purdue Univ., West Lafayette, IN. NO,H.C. Korea Advanced Institute of Science and Technology. October 1993. 109pp. 9311080286. 77077:109.

Severe accidents in light water reactors are characterized by an occurrence of multiphase flow with complicated phase changes, chemical reaction and various bifurcation phenomena. Because of the inherent difficulties associated with fullscale testing, scaled down and simulation experiments are an essential part of the severe accident analyses. However, one of the most significant shortcomings in the area is the lack of well-established and reliable scaling method and scaling criteria. In view of this, the stepwise integral scaling method is developed for severe accident analyses. This new scaling method is quite different from the conventional approach. However, its focus on dominant transport mechanisms and use of the integral response of the system make this method relatively simple to apply to very complicated multi-phase flow problems. In order to demonstrate its applicability and usefulness, three case studies have been made. The phenomena considered are: 1) corium dispersion in DCH; 2) corium spreading in BWR MARK-I containment; and 3) incore boil-off and heating process. The results of these studies clearly indicate the effectiveness of their stepwise integral scaling method. Such a simple and systematic scaling method has not been previously available to severe accident analysis.

NUREG/GR-0010: HYBRID DIGITAL SIGNAL PROCESSING AND NEURAL NETWORKS FOR AUTOMATED DIAGNOSTICS USING NDE METHODS. UPADHYAYA,B.R.; YAN,W. Tennessee, Univ. of, Knoxville, TN. November 1993. 150pp. 9312220131. 77543:001.

The primary purpose of the current research was to develop an integrated approach by combining information compression methods and artificial neural networks for the monitoring of plant components using nondestructive examination data. Specifically, data from eddy current inspection of heat exchanger tubing were utilized to evaluate this technology. The focus of the research was to develop and test various data compression methods (for eddy current data) and the performance of different neural network paradigms for defect classification and defect parameter estimation. Feedforward, fully-connected neural networks, that use the back-propagation algorithm for network training, were implemented for defect classification and defect parameter estimation using a modular network architecture. A large eddy current tube inspection database was acquired from the Metals and Ceramics Division of ORNL. These data were used to study the performance of artificial neural networks for defect type classification and for estimating defect parameters. A PC-based data preprocessing and display program was also developed as part of an expert system for data management and decision making. The results of the analysis showed that for effective (low-error) defect classification and estimation of parameters, it is necessary to identify proper feature vectors using different data representation methods. The integration of data compression and artificial neural networks for information processing was established as an effective technique for automation of diagnostics using nondestructive examination methods.

NUREG/IA-0085: ASSESSMENT OF FULL POWER TURBINE TRIP START-UP TEST FOR C. TRILLO I WITH RELAP5/MOD2. LOZANO,M.F.; MORENO,P.; DE LA CAL,C.; et al. Spain, Govt. of. July 1993. 73pp. 9308160158. ICSP-TR-TTRIP-R. 76119:218.

C. Trillo I has developed a model of the plant with RELAP5/MOD2/36.04. This model will be validated against a selected set of start-up tests. One of the transients selected to that aim is the turbine trip, which presents very specific characteristics that make it significantly different from the same transient in other PWRs of different design, the main difference being that the reactor is not tripped: a reduction in primary power is carried out instead. Pre-test calculations were done of the Turbine Trip Test and compared against the actual test. Minor problems in the first model, specially in the Control and Limitation Systems, were identified and post-test calculations had been carried out. The results show a good agreement with data for all the compared variables.

NUREG/IA-0090: ASSESSMENT OF RELAP5/MOD2 USING THE TEST DATA OF REWET-II REFLOODING EXPERIMENT SGI/R. HAMALAINEN,A. Technical Research Centre of Finland (VTT). May 1993. 44pp. 9306230098. 75462:278.

An analyses of a reflooding experiment with RELAP5/MOD2 cycle 36.04 is presented. The experiment had been carried out in the REWET-II facility simulating the reactor core with a bundle of 19 electrically heated rods. On the basis of the results of two calculations recommendations for the core nodalization are presented, and a modification to the code is proposed.

NUREG/IA-0091: ASSESSMENT OF RELAP5/MOD2 AGAINST A NATURAL CIRCULATION EXPERIMENT IN NUCLEAR POWER PLANT BORSSELE. WINTERS,L. Netherlands Energy Research Foundation ECN. July 1993. 66pp. 9308160307. ECN-89-91. 76122:247.

As part of the ICAP (International Code Assessment and Applications Program) agreement between ECN (Netherlands Energy Research Foundation) and USNRC, ECN has performed a number of assessment calculations for the thermohydraulic system analysis code RELAP5/MOD2/36.05. This document describes the assessment of this computer program versus a natural circulation experiment as conducted at the Borssele Nuclear Power Plant. The results of this comparison show that the code RELAP5/MOD2 predicts well the natural circulation behavior of Nuclear Power Plant Borssele. The work has been sponsored by the Dutch Licensing Authority and ECN.

NUREG/IA-0092: ASSESSMENT OF RELAP5/MOD2 COMPUTER CODE AGAINST THE NET LOAD TRIP TEST DATA FROM YONG-GWANG,UNIT 2. ARNE,N.; CHO,S. Korea Electric Power Corp. LEE,S.H. Korea Institute of Nuclear Safety. June 1993. 75pp. 9306290172. 75498:139.

The results of the RELAP5/MOD2 computer code simulation for the 100% Net Load Trip Test in Yong-Gwang Unit 2 are analyzed here and compared with the plant operation data. The control systems for the control rod, feedwater, steam generator level, steam dump, pressurizer level and pressure are modeled to be functioned automatically until the power level decreases below 30% nuclear power. A sensitivity study on control rod worth was carried out and it was found that variable rod worth should be used to achieve good prediction of neutron power. The results obtained from RELAP5/MOD2 simulation agree well with the plant operating data and it can be concluded that this code has the capability in analyzing the transient of this type in a best estimate means.

NUREG/IA-0094: ASSESSMENT OF RELAP5/MOD3 AGAINST TWENTY-FIVE POST-DRYOUT EXPERIMENTS PERFORMED AT THE ROYAL INSTITUTE OF TECHNOLOGY. NILSSON,L. Swedish Nuclear Power Inspectorate (Statens Karnkraftinspektion). May 1993. 92pp. 9306110035. STUDSVIKNS90/93. 75377:080.

Assessment of RELAP5/MOD2 has been made against various experimental data, among other data from twenty-five post-dryout experiments conducted at the Royal Institute of Technology (RIT) in Stockholm. As the MOD3 version of RELAP5 has now been released, incorporating a different method of calculating critical heat flux compared to RELAP5/MOD2, it seemed justified to make another assessment against the same RIT-data. The results show that the axial dryout position is generally better predicted by the MOD3 than by the MOD2 version. The prediction is, however, still nonconservative, i.e. the calculated dryout position falls in most cases downstream the actual measured point. While the pre-dryout heat transfer seems to be equal for MOD2 and MOD3, both versions giving slightly higher wall temperatures than the experiments, there is a considerable difference in the post-dryout heat transfer. The results of the RIT data comparison indicate that MOD3 underpredicts the post-dryout wall temperatures remarkably while MOD2 gave reasonable agreement. In this respect RELAP/MOD3 shows no improvement over RELAP5/MOD2.

NUREG/IA-009[illegible] RELAP5 ASSESSMENT USING LSTF TEST DATA SB-CL-18. LEE,S.H.; CHUNG,B-D.; KIM,H-J. Korea Institute of Nuclear Safety. May 1993. 100pp. 9306210225. 75402:065.

5% cold leg break test, run SB-CL-18, conducted at the Large Scale Test Facility (LSTF) was analyzed using RELAP5/MOD2 Cycle 36.04 and RELAP5/MOD3 Version 5m5 codes. The test was conducted with the main objective being the investigation of thermal-hydraulic mechanisms responsible for early core uncovery, including manometric effect due to an asymmetric coolant holdup in the steam generator upflow and downflow side. The present analysis, carried out with RELAP5/MOD2 and MOD3 codes, demonstrates the code's capability to predict, with sufficient accuracy, the main phenomena occurring in the depressurization transient, both from a qualitative and quantitative point of view. Nevertheless, several differences regarding the evolution of phenomena and affecting the timing order have to be pointed out in the base calculations. The sensitivity study on the break flow and the nodalization study in the components of the steam generator U-tubes and the cross-over legs were also carried out. The RELAP5/MOD3 calculation with the nodalization change resulted in good predictions of the major thermal-hydraulic phenomena and their timing order.

NUREG/IA-0096: NUMERICS AND IMPLEMENTATION OF THE UK HORIZONTAL STRATIFICATION ENTRAINMENT OFF-TAKE MODEL INTO RELAP5/MOD3. BRYCE,W.M. Winfrith Technology Centre (United Kingdom). June 1993. 47pp. 9307220307. AEA-TRS-1050. 75743:253.

This report presents the numerics and implementation details to add the same improved discharge quality correlations into RELAP5/MOD3. In the light of experience with the modified RELAP5/MOD2 code, some of the numerics have been slightly changed for RELAP5/MOD3. The description is quite detailed in order to facilitate change by some future code developer. A simple test calculation was performed to confirm the coding of the correlations implemented in RELAP5/MOD3.

NUREG/IA-0099: RELAP5 ASSESSMENT USING SEMISCALE SBLOCA TEST S-NH-1. LEE,E-J.; CHUNG,B-D.; KIM,H-J. Korea Institute of Nuclear Safety. June 1993. 250pp. 9306290162. 75498:090.

2-inch cold leg break test S-NH-1, conducted at the 1/1705 volume scaled facility Semiscale, was analyzed using RELAP5/MOD2 Cycle 36.04 and MOD3 Version 5m5. Loss of HPIS was assumed, and reactor trip occurred on a low PZR pressure signal (13.1 MPa), and pumps began an unpowered coastdown on SI signal (12.5 MPa). The system was recovered by opening ADV's when the PCT became higher than 811 K. Accumulator was finally injected into the system when the primary system pressure was less than 4.0 MPa. The experiment was terminated when the pressure reached the LPIS actuation set point. RELAP5/MOD2 analysis demonstrated its capability to predict, with a sufficient accuracy, the main phenomena occurring in the depressurization transient, both from a qualitative and quantitative points of view. Nevertheless, several differences were noted regarding the break flow rate and inventory distribution due to deficiencies in two-phase choked flow model, horizontal stratification interfacial drag, and a CCFL model. The main reason for the core to remain nearly fully covered with the liquid was the under-prediction of the break flow by the code. Several sensitivity calculations were tried using the MOD2 to improve the results by using the different options of break flow modeling (downward, homogeneous, and area increase). The break area compensating concept based on "the integrated break flow matching" gave the best results than downward junction and homogeneous options. And the MOD3 showed improvement in predicting a CCFL in SG and a heatup in the core.

NUREG/IA-0100: ASSESSMENT OF CCFL MODEL OF RELAP5/MOD3 AGAINST SIMPLE VERTICAL TUBES AND ROD BUNDLE TESTS. CHO,S.; ARNE,N. Korea Electric Power Corp. CHUNG,B-D.; et al. Korea Institute of Nuclear Safety. June 1993. 122pp. 9307220299. 75744:001.

The CCFL model used in RELAP5/MOD3 version 5m5 has been assessed against simple vertical tubes and rod bundle tests performed at a facility of Korea Atomic Energy Research Institute. The effect of changes in tube diameter and nodalization of tube section were investigated. The roles of interfacial drags on the flooding characteristics are discussed. Differences between the calculation and the experiment are also discussed. A comparison between model assessment results and the test data showed that the calculated value lay well on the experimental flooding curve specified by user, but the pressure jump before onset of flooding was not calculated.

NUREG/IA-0103: ASSESSMENT OF BETHSY TEST 9.1.B USING RELAP5/MOD3. LEE,S.H.; CHUNG,B-D.; KIM,H-J. Korea Institute of Nuclear Safety. June 1993. 250pp. 9307220272. 75743:001.

2" cold leg break test 9.1.b, conducted at the BETHSY facility was analyzed using the RELAP5/MOD3 Version 5m5 code. The test 9.1.b was conducted with the main objective being the investigation of the thermal-hydraulic mechanisms responsible for the large core uncovery and fuel heat-up, requiring the implementation of an ultimate procedure. The present analysis demonstrates the code's capability to predict, with sufficient accuracy, the main phenomena occurring in the depressurization transient, both from a qualitative and quantitative point of view. Nevertheless, several differences regarding the evolution of phenomena and affecting the timing order have to be pointed out in the base calculation. Three calculations were carried out to study the sensitivity to change of the nodalization in the components of the loop seal cross-over legs, and of the auxiliary feedwater control logics, and of the break discharge coefficient.

NUREG/IA-0104: RELAP5/MOD3 ASSESSMENT USING THE SEMISCALE 50% FEED LINE BREAK TEST S-FS-11. LEE,E-J.; CHUNG,B-D.; KIM,H-J. Korea Institute of Nuclear Safety. June 1993. 200pp. 9306290153. 75497:272.

The RELAP5/MOD3 5m5 code was assessed using the 1/1705 volume scaled Semiscale 50% Feed Line Break (FLB) test S-FS-11. Test S-FS-11 was designed in three phases: (a) blowdown phase, (b) stabilization phase, and (c) refill phase. The fist objective was to assess the code applicability to 50% FLB situation, the second was to evaluate the FSAR conservatisms regarding SG heat transfer degradation, steam line check valve failure, break flow state, and peak primary system pressure, and the third was to validate the EOP effectiveness. The code was able to simulate the major T/H parameters except for the two-phase break flow and the secondary convective heat transfer rate. The two-phase break flow had still deficiencies. The current boiling heat transfer rate was developed from the data for flow inside of a heated tube, not for flow around heated tubes in a tube bundle. Results indicated that the assumption of 100% heat transfer until the liquid inventory depletion was not conservative, the failed affected steam generator main steam line check valve assumption was not either conservative, the measured break flow experienced all types of flow conditions, the relative proximity to the 110% design pressure limit was conservative. The automatic actions during the blowdown phase were effective in mitigating the consequences. The stabilization operation performed by operator actions were effective to permit natural circulation cooldown and depressurization. The voided secondary refill operations also verified the effectiveness of the operations while recovering the inventory in a voided steam generator.

NUREG/IA-0105: ASSESSMENT OF RELAP5/MOD3 VERSION 5M5 USING INADVERTENT SAFETY INJECTION INCIDENT DATA OF KORI UNIT 3 PLANT. KIM,K.T.; CHUNG,B-D.; KIM,I.G.; et al. Korea Institute of Nuclear Safety. May 1993. 67pp. 9306180330. 75387:232.

An inadvertent safety injection incident occurred at Kori Unit 3 in September 6, 1990. It was analyzed using the RELAP5/MOD3 code. The event was initiated by a closure of main feedwater control valve of one of three steam generators. High pressure safety injection system was actuated by the low pressure signal of main steam line. The actual sequence of plant transient with the proper estimations of operator actions was investigated in the present calculation. The asymmetric loop behaviors of the plant were also considered by nodalizing the loops of the plant into three. The calculational results are compared with the plant transient data. It is shown that the overall plant transient depends strongly on the auxiliary feedwater flowrate controlled by the operator and that the code gives an acceptable prediction of the plant behavior with the proper assumptions of the operator actions. The results also show that the solidification of pressurizer does not occur and the liquid-vapor mixture does not flow out through pressurizer PORV. The behavior of primary pressure during pressurizer PORV actuation is poorly predicted because the actual behavior of pressurizer PORV could not be modelled in the present simulation.

NUREG/IA-0106: ASSESSMENT OF PWR STEAM GENERATOR MODELLING IN RELAP5/MOD2. PUTNEY,J.M.; PREECE,R.J. National Power (United Kingdom). June 1993. 123pp. 9307120157. TEC/L/0471/R91. 75623:025.

An assessment of Steam Generator (SG) modelling in the PWR thermal-hydraulic code RELAP5/MOD2 is presented. The assessment is based on a review of code assessment calculations performed in the UK and elsewhere, detailed calculations against a series of commissioning tests carried out on the Wolf Creek PWR and analytical investigations of the phenomena involved in normal and abnormal SG operation. A number of modelling deficiencies are identified and their implications for PWR safety analysis are discussed - including methods for compensating for the deficiencies through changes to the input deck. Consideration is also given as to whether the deficiencies will still be present in the successor code RELAP/MOD3.

NUREG/IA-0107: ASSESSMENT OF RELAP5/MOD2 AGAINST A LOAD REJECTION FROM 100% TO 50% POWER IN THE VANDELLOS II NUCLEAR POWER PLANT. LLOPIS,C.; MENDIZABAL,R.; PEREZ,J. Spain, Govt. of. June 1993. 53pp. 9307220264. ICSP-V2-R50-R. 75742:241.

An assessment of RELAP5/MOD2 cycle 36.04 against a load rejection from 100% to 50% power in Vandellos II NPP (Spain) is presented. The work is inscribed in the framework of the Spanish contribution to ICAP Project. The model used in the simulation consists of a single loop, a steam generator and a steam line up to the steam header, all of them enlarged on a scale of 3:1; and full-scaled reactor vessel and pressurizer. The results of the calculations have been in reasonable agreement with plant measurements.

NUREG/IA-0108: ASSESSMENT OF RELAP5/MOD2 AGAINST A TURBINE TRIP FROM 100% POWER IN THE VANDELLOS II NUCLEAR POWER PLANT. LLOPIS,C.; PEREZ,J.; MENDIZABAL,R. Spain, Govt. of. June 1993. 58pp. 9306210328. ICSP-V2-R100-R. 75403:325.

An assessment of RELAP5/MOD2 cycle 36.04 against a turbine trip from 100% power in Vandellos II NPP (Spain) is presented. The work is inscribed in the framework of the Spanish contribution to ICAP Project. The model used in the simulation consists of a single loop, a steam generator and a steam line up to the steam header all of them enlarged on a scale of 3:1; and full-scaled reactor vessel and pressurizer. The results of the calculations have been in reasonable agreement with plant measurements. An additional study has been performed, to check the ability of a model in which all the plant components are full-scaled to reproduce the transient. A second study has been performed using the Homogeneous Equilibrium Model in the pressurizer trying to elucidate the influence of the velocity slip in the primary depressurization rate.

NUREG/IA-0109: ASSESSMENT OF RELAP5/MOD2 AGAINST A 10% LOAD REJECTION TRANSIENT FROM 75% STEADY STATE IN THE VANDELLOS II NUCLEAR POWER PLANT. LLOPIS,C.; PEREZ,J.; CASALS,A.; et al. Spain, Govt. of. May 1993. 59pp. 9306210175. UNID-91-08. 75401:065.

The Consejo de Seguridad Nuclear (CSN) and the Asociacion Nuclear Vandellos (ANV) have developed a model of Vandellos II Nuclear Power Plant. The ANV collaboration consisted in the supply of design and actual data, the cooperation in the simulation of the control systems and other model components, as well as in the results analysis. The obtained model has been assessed against the following transients occurring in the plant: a trip from the 100% power level (CSN); a load rejection from 100% to 50% (CSN); a load rejection from 75% to 65% (ANV), and a feedwater turbopump trip (ANV). This copy is a report of the load rejection from 75% to 65% transient simulation. This transient was one of the tests carried out in Vandellos II NPP during the startup tests.

NUREG/IA-0110: ASSESSMENT OF RELAP5/MOD2 AGAINST A MAIN FEEDWATER TURBOPUMP TRIP TRANSIENT IN THE VANDELLOS II NUCLEAR POWER PLANT. LLOPIS,C.; PEREZ,J.; CASALS,A.; et al. Spain, Govt. of. December 1993. 54pp. 9401060229. ICAP-00219. 77686:306.

The Consejo de Seguridad Nuclear (CSN) and the Asociacion Nuclear Vandellos (ANV) have developed a model of Vandellos II Nuclear Power Plant. The ANV collaboration consisted in the supply of design and actual data, the cooperation in the simulation of the control systems and other model components, as well as in the results analysis. The obtained model has been assessed against the following transients occurred in plant: A trip from the 100% power level (CSN); a load rejection from 100% to 50% (CSN); a load rejection from 75% to 65% (ANV); a feedwater turbopump trip (ANV). This copy is a report of the feedwater turbopump trip transient simulation. This transient occurred actually in plant on June 19, 1989.

NUREG/IA-0112: ASSESSMENT OF RELAP5/MOD2 AGAINST ECN-REFLOOD EXPERIMENTS. WOUDSTRA,A.; VANDEBOGAARD,J.; STOOP,P.M. Netherlands Energy Research Foundation ECN. July 1993. 87pp. 9308160054. ECN-C-92-008. 76121:243.

As part of the ICAP (International Code Assessment and Applications Program) agreement between ECN (Netherlands Energy Research Foundation) and USNRC, ECN has performed a number of assessment calculations with the computer program RELAP5. This report describes the results as obtained by ECN from the assessment of the thermohydraulic computer program RELAP5/MOD2/CY 36.05 versus a series of reflood experiments in a bundle geometry. A total number of seven selected experiments have been analyzed, from the reflood experimental program as previously conducted by ECN under contract of the Commission of the European Communities (CEC). In this document, the results of the analyses are presented and a comparison with the experimental data is provided.

NUREG/IA-0113: PRELIMINARY ASSESSMENT OF PWR STEAM GENERATOR MODELLING IN RELAP5/MOD3. PREECE,R.J.; PUTNEY,J.M. National Power (United Kingdom). July 1993. 26pp. 9309090021. 76381:154.

A preliminary assessment of Steam Generator (SG) modelling in the PWR thermal-hydraulic code RELAP5/MOD3 is presented. The study is based on calculations against a series of steady-state commissioning tests carried out on the Wolf Creek PWR over a range of load conditions. Data from the tests are used to assess the modelling of primary to secondary side heat transfer and, in particular, to examine the effect of reverting to the standard form of the Chen heat transfer correlation in place of the modified form applied in RELAP5/MOD2. Comparisons between the two versions of the code are also used to show how the new interphase drag model in RELAP5/MOD3 affects the calculation of SG liquid inventory and the void fraction profile in the riser.

NUREG/IA-0116: ASSESSMENT OF RELAP5/MOD3/V5M5 AGAINST THE UPTF TEST NUMBER 11 (COUNTERCURRENT FLOW IN PWR HOT LEG). CURCA-TIVIG,F. Siemens AG - KWU Group (formerly Siemens AG - Bereich Energieerzeugung (KWU)). May 1993. 100pp. 9306210180. KWU E412/91/E10. 75401:122.

Analysis of the UPTF Test No. 11 using the "best-estimate" computer code RELAP5/MOD3/Version 5m5 is presented. Test No. 11 was a quasi-steady state, separate effect test designed to investigate the conditions for countercurrent flow of steam and saturated water in the hot leg of a PWR. An unphysical result was received using a CCFL correlation of the Wallis type with the intercept C = 0.644 and the slope m = 0.8. The unphysical prediction is an indication of possible programming errors in the CCFL model of the RELAP5/MOD3/V5m5 computer code.

NUREG/IA-0118: ANALYSIS OF LOFT TEST L5-1 USING RELAP5/MOD2. COOPER,S. United Kingdom. May 1993. 43pp. 9306210192. TD/SPB/REP/0130. 75401:234.

The RELAP5/MOD2 code, Reference 1, is being used by Nuclear Electric for the calculation of Small Break Loss of Coolant Accidents (SBLOCA) and pressurized transient sequences in the Sizewell "B" PWR. To validate the code for this purpose, it has been used to model experiments of this type of transient carried out in various integral test facilities. A number of these studies have been for experiments carried out in the LOFT experimental reactor, Reference 2, and are described in References 3, 4, 5, 6, and 7. To assist in assessing the capability of RELAP5/MOD2, the LOFT test L5-1 has been selected for analysis. This test was designed to simulate the rupture of a single 14 inch diameter accumulator injection line in a commercial PWR, equivalent to a 25% area break in the broken loop cold leg. Early in the transient the pumps were tripped and the HPIS injection initiated; towards the end of the transient, accumulator and LPIS injection began. It should be noted that for Sizewell "B" analyses a 25% break is classified as large, whereas in this report, as in the external literature, this break size is referred to as intermediate.

NUREG/IA-0119: ASSESSMENT AND APPLICATION OF BLACK-OUT TRANSIENTS AT ASCO NUCLEAR POWER PLANT WITH RELAP5/MOD2. REVENTOS,F.; BAPTISTA,J.S.; NAVAS,A.P.; et al. Spain, Govt. of. June 1993. 63pp. 9306290137. ICSP-AS-BOUT-R. 75497:140.

The Asociacion Nuclear Asco has prepared a model of Asco NPP using RELAP5/MOD2. This model, which include thermal-hydraulics, kinetics and protection and controls, has been qualified in previous calculations of several actual plant transients. The first part of the transient presented in this report is an actual black-out and one of the transients of the qualification process. The results are in agreement with plant data. The second part of the transient is a hypothetical case. It consists in re-starting a primary pump and assume a new black-out. The phenomenology prediction of this second part has been useful from the operation and safety point of view.

NUREG/IA-0120: ASSESSMENT OF THE TURBINE TRIP TRANSIENT IN COFRENTES NPP WITH TRAC-BF1. CASTRILLO,F. Hidroelectrica Espanola. GOMEZ,A.; GALLEGO,I.; et al. Union Iberoamericana De Tecnologia. June 1993. 74pp. 9306290147. EST-SIAN-22. 75497:203.

This report presents the results of the assessment of TRAC-BF1 (G1-J1) code with the model of C. N. Cofrentes for simulation of the transient originated by the manual trip of the main turbine. C. N. Cofrentes is a General Electric designed BWR/6 plant, with a nominal core thermal power of 2894 Mwt, in commercial operation since 1985, owned and operated by Hidroelectrica Espanola, S. A. The plant incorporates all the characteristics of BWR/6 reactors, with two turbine driven FW pumps. As a result of this assessment a model of C. N. Cofrentes has been developed for TRAC-BF1 that fairly reproduces operational transient behavior of the plant. A special purpose code was generated to obtain reactivity coefficients, as required by TRAC-BF1, from the 3D simulator.

NUREG/IA-0121: ASSESSMENT OF A PRESSURIZER SPRAY VALVE FAULTY OPENING TRANSIENT AT ASCO NUCLEAR POWER PLANT WITH RELAP5/MOD2. REVENTOS,F.; BAPTISTA,J.S.; NAVAS,A.P.; et al. Spain, Govt. of. December 1993. 58pp. 9401140031. ICSP-AS-SPR-R. 77797:080.

The Association Nuclear Asco has prepared a model of Asco Nuclear Power Plant using RELAP5/MOD2. This model, which includes thermalhydraulics, kinetics and protection and controls, has been qualified in previous calculations of several actual plant transients. One of the transients of the qualification process is a "pressurized spray valve faulty opening" presented in this report. It consists in a primary coolant depressurization that causes the reactor trip by overtemperature and later on the actuation of the safety injection. The results are in close agreement with plant data.

NUREG/IA-0122: ASSESSMENT OF MSIV FULL CLOSURE FOR SANTA MARIA DE GARONA NUCLEAR POWER PLANT USING TRAC-BF1 (G1J1). CRESPO,J.L.; FERNANDEZ,R.A. Cantabria, Univ. of, Spain. June 1993. 46pp. 9307060112. ICSP-GA-MSIV-T. 75572:167.

An assessment of the first 60 seconds of a spurious Main Steam Insolation Valve (MSIV's) closure for Santa Maria Garona Nuclear Power Plant using TRAC-BF1 code is presented. Reasonable and realistic adjustments have been made in the model to improve its performance. This work is part of the validation set for the TRAC model that is being developed for wider use and to test the code capabilities. As a result of the analysis, it is felt that TRAC-BF1 is capable of reproducing the plant behavior with an acceptable degree of accuracy although better models are clearly needed, in addition [illegible]ther noding work and code improvements. The code took almost 14000 sec. which makes a 1/230 calculation time to real time ratio. For this transient a mechanistic separator model is needed. It will also help to cut down running costs if the vessel noding could have a different number of cells at different heights. Though not very important for this transient, the critical flow model will allow for realistic RV flow assumptions. There are not guidelines available for separator modelling in transients. It has been found that a detailed noding in the separator region may be needed to represent steam-water interaction.

NUREG/IA-0123: APPLICATION OF FULL POWER BLACKOUT FOR C.N. ALMARAZ WITH RELAP5/MOD2. LECHAS,A.L. Spain, Govt. of. June 1993. 100pp. 9306290128. ICSP-AL-BOUT-R. 75496:293.

The analysis group of Almaraz Nuclear Power Plant has developed a model of the plant with RELAP5/MOD2/36.04. This model is the result of the work-experience on the code RELAP5/MOD1 that was the standard code during the period 1984/1989. Different solutions were adopted in the network to adequate the model to RELAP5/MOD2 Computer Code. This transient was selected for ICAP because it presents an experience with the same transient calculated with RELAP5/MOD1/CY 29 Computer Code. The comparison between both analysis will be interesting.

NUREG/IA-0124: ASSESSMENT OF RELAP5/MOD2 AGAINST A PRESSURIZER SPRAY VALVE INADVERTED FULLY OPENING TRANSIENT AND RECOVERY BY NATURAL CIRCULATION IN JOSE CABRERA NUCLEAR STATION. ARROYO,R.; REBOLLO,L. Union Electrica Fenosa, S.A. June 1993. 126pp. 9307060121. ICSP-JC-SPR-R. 75572:049.

This document presents the comparison between the simulation results and the plant measurements of a real event that took place in Jose Cabrera nuclear power plant in August 30, 1984. The event was originated by the local, continuous and inadverted opening of the pressurizer spray valve PCV-400A. Jose Cabrera power plant is a single loop Westinghouse PWR belonging to UNION ELECTRICA FENOSA, S.A. (UNION FENOSA), a Spanish utility which participates in the International Code Assessment and Applications Program (ICAP) as a member of UNIDAD ELECTRICA, S.A. (UNESA). This is the second of its two contributions to the Program: The first one was an application case and this is an assessment one. The simulation has been performed using the RELAP5/MOD2 cycle 36.04 code, running on a CDC CYBER 180/830 computer under NOS 2.5. operating system. The main phenomena have been calculated correctly and some conclusions about the 3D characteristics of the condensation due to the spray and its simulation with a 1D tool have been reached.

NUREG/IA-0125: ASSESSMENT OF RELAP5/MOD2 COMPUTER CODE AGAINST THE NATURAL CIRCULATION TEST DATA FROM YONG-GWANG UNIT 2. ARNE,N.; CHO,S. Korea Electric Power Corp. KIM,H-J. Korea Institute of Nuclear Safety. June 1993. 110pp. 9306290132. 75497:032.

The results of the RELAP5/MOD2 computer code simulation for the Natural Circulation Test in Yong-Gwang Unit 2 are analyzed here and compared with the plant operation data. The result of comparison reveals that the code calculation does present well the overall macroscopic behaviors of thermalhydraulic parameters in primary and secondary system compared with the plant operating data. The sensitivity study is performed to find out the effect of steam dump flow rate on the primary temperatures and it is found that the primary temperatures are very sensitive to the steam dump flow rate during the Natural Circulation. Because of the inherent uncertainties in the plant data, the assessment work is focussed on phenomena whereby the comparison between plant data and calculated data is based more on trends than on absolute values.

NUREG/IA-0126: 2D/3D PROGRAM WORK SUMMARY REPORT. DAMERELL,P.S.; SIMONS,J.W. MPR Associates, Inc. *; et al. Japan Atomic Energy Research Institute. June 1993. 400pp. 9307220220. GRS-100. 75745:042.

The 2D/3D Program was carried out by Germany, Japan and the United States to investigate the thermal-hydraulics of a PWR large-break LOCA. A contributory approach was utilized in which each country contributed significant effort to the program and all three countries shared the research results. Germany constructed and operated the Upper Plenum Test Facility (UPTF), and Japan constructed and operated the Cylindrical Core Test Facility (CCTF) and the Slab Core Test Facility (SCTF). The US contribution consisted of provision of advanced instrumentation to each of the three test facilities, and assessment of the TRAC computer code against the test results. Evaluations of the test results were carried out in all three countries. This report summarizes the 2D/3D Program in terms of the contributing efforts of the participants.

NUREG/IA-0127: REACTOR SAFETY ISSUES RESOLVED BY THE 2D/3D PROGRAM. DAMERELL,P.S.; SIMONS,J.W. MPR Associates, Inc. * Japan Atomic Energy Research Institute. July 1993. 400pp. 9308160173. GRS-101. 76116:001.

The 2D/3D Program studied multidimensional thermal-hydraulics in a PWR core and primary system during the end-of-blowdown and post-blowdown phases of a large-break LOCA (LBLOCA), and during selected small-break LOCA (SBLOCA) transients. The program included tests at the Cylindrical Core Test Facility (CCTF), the Slab Core Test Facility (SCTF), and the Upper Plenum Test Facility (UPTF), and computer analyses using TRAC. Tests at CCTF investigated core thermal-hydraulics and overall system behavior while tests at SCTF concentrated on multidimensional core thermal-hydraulics. The UPTF tests investigated two-phase flow behavior in the downcomer, upper plenum, tie plate region, and primary loops. TRAC analyses evaluated thermal-hydraulic behavior throughout the primary system in tests as well as in PWRs. This report summarizes the test and analysis results in each of the main areas where improved information was obtained in the 2D/3D Program. The discussion is organized in terms of the reactor safety issues investigated.

NUREG/IA-0128: INTERNATIONAL CODE ASSESSMENT AND APPLICATIONS PROGRAM: SUMMARY OF CODE ASSESSMENT STUDIES CONCERNING RELAP5/MOD2, RELAP5/MOD3, AND TRAC-B. SCHULTZ,R.R. EG&G Idaho, Inc. December 1993. 268pp. 9401060241. EGG-EAST-8719. 77686:001.

Members of the International Code Assessment Program (ICAP) have assessed the U. S. Nuclear Regulatory Commission (USNRC) advanced thermal-hydraulic codes over the past few years in a concerted effort to identify deficiencies, to define user guidelines, and to determine the state of each code. The results of sixty-two code assessment reviews, conducted at INEL, are summarized. Code deficiencies are discussed and user recommended nodalizations investigated during the course of conducting the assessment studies and reviews are listed. All the work that is summarized was done using the RELAP5/MOD2, RELAP5/MOD3, and TRAC-B codes.

2

Secondary Report Number Index

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

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Subject Index

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

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NRC Originating Organization Index (Staff Reports)

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

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NUREG-1488 DRFT FC: REVISED LIVERMORE SEISMIC HAZARD ESTIMATES FOR 69 NUCLEAR POWER PLANT SITES EAST OF THE ROCKY MOUNTAINS.Draft Report For Comment.

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NRC Originating Organization Index (International Agreements)

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

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NRC Contract Sponsor Index (Contractor Reports)

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

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Contractor Index

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

NUREG/CR-5836: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE PALO VERDE NUCLEAR POWER PLANT.
NUREG/CR-5884 V1 DRF: REVISED ANALYSES OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION.Effects Of Current Regulatory And Other Considerations On The Financial Assurance....Main Report.Draft Report For Comment.
NUREG/CR-5884 V2 DRF: REVISED ANALYSES OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION.Effects Of Current Regulatory And Other Considerations On The Financial Assurance....Appendices.Draft Report For Comment.
NUREG/CR-5894: RADIONUCLIDE CHARACTERIZATION OF REACTOR DECOMMISSIONING WASTE AND NEUTRON-ACTIVATED METALS.
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NUREG/CR-5898: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE POINT BEACH NUCLEAR POWER PLANT.
NUREG/CR-5973: CODES AND STANDARDS AND OTHER GUIDANCE CITED IN REGULATORY DOCUMENTS.
NUREG/CR-5975: INCENTIVE REGULATION OF INVESTOR-OWNED NUCLEAR POWER PLANTS BY PUBLIC UTILITY REGULATORS.
NUREG/CR-5988: SOIL CHARACTERIZATION METHODS FOR UNSATURATED LOW-LEVEL WASTE SITES.
NUREG/CR-5989: PERFORMANCE TESTING OF EXTREMITY DOSIMETERS--PILOT TEST.
NUREG/CR-5996: SUBSURFACE INJECTION OF RADIOACTIVE TRACERS.Field Experiment For Model Validation Testing.
NUREG/CR-5998: SIMULATION OF UNSATURATED FLOW AND NONREACTIVE SOLUTE TRANSPORT IN A HETEROGENEOUS SOIL AT THE FIELD SCALE.
NUREG/CR-6029 V01: AGING ASSESSMENT OF NUCLEAR AIR-TREATMENT SYSTEM HEPA FILTERS AND ADSORBERS.Phase I.
NUREG/CR-6043 V01: AGING ASSESSMENT OF ESSENTIAL HVAC CHILLERS USED IN NUCLEAR POWER PLANTS.Phase I.
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NUREG/CR-6084: VALUE-IMPACT ANALYSIS OF GENERIC ISSUE 143, "AVAILABILITY OF HEATING, VENTILATION, AIR CONDITIONING (HVAC) AND CHILLED WATER SYSTEMS."
NUREG/CR-6114 V01: APPLICATION OF AN INFILTRATION EVALUATION METHODOLOGY TO A HYPOTHETICAL LOW-LEVEL WASTE DISPOSAL FACILITY.

BROOKHAVEN NATIONAL LABORATORY
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NUREG/CR-2907 V11: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1990.
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NUREG/CR-5844: AGING ASSESSMENT OF BISTABLES AND SWITCHES IN NUCLEAR POWER PLANTS.
NUREG/CR-5883: HEALTH RISK ASSESSMENT OF IRRADIATED TOPAZ.
NUREG/CR-5911: SOURCE TERM EVALUATION FOR RADIOACTIVE LOW-LEVEL WASTE DISPOSAL PERFORMANCE ASSESSMENT.
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NUREG/CR-5934: HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM RISK-BASED INSPECTION GUIDE FOR QUAD-CITIES STATION,UNITS 1 AND 2.
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NUREG/CR-5959: HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM RISK-BASED INSPECTION GUIDE FOR ENRICO FERMI ATOMIC POWER PLANT,UNIT 2.
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NUREG/CR-6049: PIPING BENCHMARK PROBLEMS FOR THE GENERAL ELECTRIC ADVANCED BOILING WATER REACTOR.
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NUREG/CR-6111: INTEGRATED SYSTEMS ANALYSIS OF THE PIUS REACTOR.

BROWN UNIV., PROVIDENCE, RI
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