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# Regulatory and Technical Reports (Abstract Index Journal)

Annual Compilation for 1991

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**Division of Freedom of Information and Publications Services**
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## PREFACE

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

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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
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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 eleme    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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<u>International Agreement Report</u>

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

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 report number of the originating organization (if given), and (9) the microfiche address (for NRC internal use).

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

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

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

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<u>NRC Report Codes</u>

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

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

# 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 V15:** LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of December 31, 1990.(Gray Book I) HARTFIELD,R.A. Division of Computer & Telecommunications Services (Post 890205). August 1991. 352pp. 9109050304. 58987:142.

The Nuclear Re[illegible]atory Commission's annual summary of licensed nuclear [illegible] reactor data is based primarily on the report of operatin[illegible] 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 1990) 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 V14 N04:** LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1990.(White Book) * Division of Reactor Inspection & Safeguards (Post 870411). February 1991. 274pp. 9103200057. 57062:125.

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 1990 through December 1990.

**NUREG-0040 V15 N01:** LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1991.(White Book) * Division of Reactor Inspection & Safeguards (Post 870411). May 1991. 105pp. 9105300232. 57867:156.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organization during the period from January 1991 through March 1991.

**NUREG-0040 V15 N02:** LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,April-June 1991.(White Book) * Division of Reactor Inspection & Safeguards (Post 870411). September 1991. 292pp. 9110080408. 59310:281.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organization during the period from April through June 1991.

**NUREG-0040 V15 N03:** LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,July-September 1991.(White Book) * Division of Reactor Inspection & Safeguards (Post 870411). October 1991. 143pp. 9111110275. 59575:187.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organization during the period from July 1991 through September 1991.

**NUREG-0090 V13 N03:** REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1990. * Office for Analysis & Evaluation of Operational Data, Director. January 1991. 35pp. 9102250192. 56789:018.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health and safety and requires a quarterly report of such events to be made to Congress. This report covers the period July 1 through September 30, 1990. The report discusses six abnormal occurrences, none of which involved a nuclear power plant. There were five abnormal occurrences at NRC-licensed facilities: one involved a medical therapy misadministration; three involved medical diagnostic misadministrations; and one involved a significant breakdown in management and procedural controls at a medical facility. The sixth abnormal occurrence was reported by an Agreement State (Arizona); the event involved a medical therapy misadministration.

**NUREG-0090 V13 N04:** REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1990. * Office for Analysis & Evaluation of Operational Data, Director. March 1991. 24pp. 9104040281. 57260:302.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health and safety and requires a quarterly report of such events to be made to Congress. This report covers the period October 1 through December 31, 1990. The report discusses five abnormal occurrences, none of which involved a nuclear power plant. Two involved significant overexposures to the hands of two radiographers, two involved medical therapy misadministrations, and one involved a medical diagnostic misadministration. No abnormal occurrences were reported by the Agreement States. The report also contains information that updates a previously reported abnormal occurrence.

**NUREG-0090 V14 N01:** REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.January-March 1991. * Office for Analysis & Evaluation of Operational Data, Director. June 1991. 21pp. 9107220285. 58489:273.

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 th[illegible] period January through March 1991. The report discusses six abnormal occurrences, none of which involved a nuclear power plant. Five of the events occurred at NRC-licensed facilities: one involved a significant degradation of plant safety at a nuclear fuel cycle facility, one involved a medical diagnostic misadministration, and three involved medical therapy misadministrations. An Agreement State

(Arizona) reported one abnormal occurrence that involved medical therapy misadministrations.

**NUREG-0090 V14 N02:** REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1991. * Office for Analysis & Evaluation of Operational Data, Director. September 1991. 31pp. 9110290332. 59455:243.

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 1991. The report discussed five abnormal occurrences, none of which involved a nuclear power plant. Two of the events occurred at NRC-licensed facilities: one involved a potential criticality accident at a nuclear fuel cycle facility, and one involved multiple medical teletherapy misadministrations. The Agreement States reported three abnormal occurrences, all involving radiation overexposures. The report also contains information that updates some previously reported abnormal occurrences.

**NUREG-0090 V14 N03:** REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1991. * Office for Analysis & Evaluation of Operational Data, Director. December 1991. 24pp. 9201140012. 60299:187.

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 July through September 1991. The report discusses two abnormal occurrences at NRC-licensed facilities, neither involving a nuclear power plant. One involved radiation exposures to members of the public from a lost radioactive source and the other involved a medical diagnostic misadministration. The Agreement States reported no abnormal occurrences. The report also contains information that updates some previously reported abnormal occurrences.

**NUREG-0304 V15 N04:** REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1990. * Division of Freedom of Information & Publications Services (Post 890205). March 1991. 137pp. 9104040288. 57261:003.

This journal includes all formal reports in the NUREG series prepared by the NRC staff and contractors; proceedings of conferences and workshops; as well as 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 V16 N01:** REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1991,January-March. * Division of Freedom of Information & Publications Services (Post 890205). June 1991. 43pp. 9107220276. 58489:294.

See NUREG-0304,V15,N04 abstract.

**NUREG-0304 V16 N02:** REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Second Quarter 1991,April-June. * Division of Freedom of Information & Publications Services (Post 890205). November 1991. 46pp. 9112310208. 60177:124.

See NUREG-0304,V15,N04 abstract.

**NUREG-0304 V16 N03:** REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Third Quarter 1991,July-September. * Division of Freedom of Information & Publications Services (Post 890205). December 1991. 47pp. 9201080097. 60232:275.

See NUREG-0304,V15,N04 abstract.

**NUREG-0327 R05:** OWNERS OF NUCLEAR POWER PLANTS. WOOD,R.S. Office of Nuclear Reactor Regulation, Director (Post 870411). July 1991. 29pp. 9108130276. 58765:077.

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

**NUREG-0383 V01 R14:** DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Report Of NRC Approved Packages. * Division of Safeguards & Transportation (Post 870413). October 1991. 466pp. 9112310211. 60154:132.

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 cource 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 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 R14:** DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Certificates Of Compliance. * Division of Safeguards & Transportation (Post 870413). October 1991. 610pp. 9201060296. 60203:122.

See NUREG-0383,V01,R14 abstract.

**NUREG-0383 V03 R11:** DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Report Of NRC Approved Quality Assurance Programs For Radioactive Materials Packages. * Division of Safeguards & Transportation (Post 870413). October 1991. 153pp. 9112310225. 60153:039.

See NUREG-0383,V01,R14 abstract.

**NUREG-0386 D05 R09:** UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission, Appeal Board And Licensing Board Decisions.July 1972 - September 1990. * Office of the General Counsel (Post 860701). February 1991. 573pp. 9103200069. 57064:001.

This Revision 9 of the fifth 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, 1990, interpreting the NRC's Rules Practice in 10 CFR Part 2.

**NUREG-0386 D06:** UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission, Appeal Board And Licensing Decisions.July 1972 - December 1990. * Office of the General Counsel (Post 860701). December 1991. 696pp. 9201140351. 60299:246.

This sixth edition of the NRC Staff Practice and Procedure Digest contains a digest of a number of Commission, Atomic Safety and Licensing Appeal Board, and Atomic Safety and Licensing Board decisions issued during the period from July 1, 1972 to December 31, 1990 interpreting the NRC's Rules of Practice in 10 CFR Part 2. This sixth edition replaces in part earlier editions and revisions and includes appropriate changes reflecting the amendments to the Rules of Practice effective through December 31, 1990.

**NUREG-0430 V10:** LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.July 1989 - June 1990.(Gray Book II) BROWN,C.; JOY,D. Office of Nuclear Material Safety & Safeguards. February 1991. 18pp. 9103200016. 57066:152.

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 R17:** SAFEGUARDS SUMMARY EVENT LIST (SSEL).Pre-NRC Through December 31, 1990. * Division of Safeguards & Transportation (Post 870413). July 1991. 123pp. 9108130182. 58764:182.

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, alcohol and drug related (through 1989), and miscellaneous. Because of public interest, the miscellaneous section 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 reports.

**NUREG-0540 V12 N11:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30,1990. * Division of Freedom of Information & Publications Services (Post 890205). January 1991. 293pp. 9102040312. 56570:279.

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 to Principal Documents.

**NUREG-0540 V12 N12:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. December 1-31, 1990. * Division of Freedom of Information & Publications Services (Post 890205). February 1991. 337pp. 9102280245. 56835:269.

See NUREG-0540,V12,N11 abstract.

**NUREG-0540 V13 N01:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31, 1991. * Division of Freedom of Information & Publications Services (Post 890205). March 1991. 330pp. 9103260124. 57153:035.

See NUREG-0540,V12,N11 abstract.

**NUREG-0540 V13 N02:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. February 1-28, 1991. * Division of Freedom of Information & Publications Services (Post 890205). April 1991. 336pp. 9104250053. 57489:113.

See NUREG-0540,V12,N11 abstract.

**NUREG-0540 V13 N03:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. March 1-31, 1991. * Division of Freedom of Information & Publications Services (Post 890205). May 1991. 388pp. 9105210080. 57808:012.

See NUREG-0540,V12,N11 abstract.

**NUREG-0540 V13 N04:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30, 1991. * Division of Freedom of Information & Publications Services (Post 890205). June 1991. 386pp. 9107010133. 58252:076.

See NUREG-0540,V12,N11 abstract.

**NUREG-0540 V13 N05:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. May 1-31, 1991. * Division of Freedom of Information & Publications Services (Post 890205). July 1991. 335pp. 9108130341. 58766:177.

See NUREG-0540,V12,N11 abstract.

**NUREG-0540 V13 N06:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. June 1-30, 1991. * Division of Freedom of Information & Publications Services (Post 890205). August 1991. 315pp. 9109050321. 58986:187.

See NUREG-0540,V12,N11 abstract.

**NUREG-0540 V13 N07:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. July 1-31, 1991. * Division of Freedom of Information & Publications Services (Post 890205). September 1991. 335pp. 9110080399. 59309:306.

See NUREG-0540,V12,N11 abstract.

**NUREG-0540 V13 N08:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. August 1-31, 1991. * Division of Freedom of Information & Publications Services (Post 890205). October 1991. 314pp. 9110280064. 59449:036.

See NUREG-0540,V12,N11 abstract.

**NUREG-0540 V13 N09:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. September 1-30, 1991. * Division of Freedom of Information & Publications Services (Post 890205). November 1991. 299pp. 9112310213. 60165:223.

See NUREG-0540,V12,N11 abstract.

**NUREG-0540 V13 N10:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. October 1-31, 1991. * Division of Freedom of Information & Publications Services (Post 890205). December 1991. 358pp. 9201090195. 60244:285.

See NUREG-0540,V12,N11 abstract.

**NUREG-0675 S34:** SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company) ROOD,H.; CHOKSHI,N.; MCMULLEN,R.; et al. Division of Reactor Projects - III,IV,V (Post 901216). June 1991. 354pp. 9107100057. 58383:001.

Supplement 34 to the Safety Evaluation Report for the application by Pacific Gas and Electric Company (PG&E) for licenses to operate Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 (Docket Nos. 50-275 and 50-323, respectively) has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement documents the NRC staff review of the Long-Term Seismic Program conducted by PG&E in response to License Condition 2.C.(7) of Facility Operating License DPR-80, the Diablo Canyon Unit 1 operating license.

**NUREG-0713 V10:** OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES,1988.Twenty First Annual Report. RADDATZ,C.T. Division of Regulatory Applications (Post 870413). HAGEMEYER,D. Science Applications International Corp. (formerly Science Applications, Inc.). July 1991. 284pp. 9108190270. 58827:261.

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 1988. 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 1988 annual reports submitted by about 429 licensees indicated that approximately 220,048 individuals were monitored, 113,000 of whom were monitored by nuclear power facilities. They incurred an average individual dose of 0.20 rem (cSv) and an average measurable dose of 0.41 (cSv). Termination radiation exposure reports were analyzed to reveal that about 113,072 individuals completed their employment with one or more of the 429 covered licensees during 1988. Some 80,211 of these individuals terminated from power reactor facilities, and about 8,760 of them were considered to be transient workers who received an average dose of 0.27 rem (cSv).

**NUREG-0725 R07:** PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL. * Division of Safeguards & Transportation (Post 870413). January 1991. 32pp. 9102060142. 56593:209.

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 1989. It also lists detailed highway and railway segments used within each state from October 1, 1987 through December 31, 1989.

**NUREG-0750 V32 I02:** INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-December 1990. * Division of Freedom of Information & Publications Services (Post 890205). March 1991. 75pp. 9104250060. 57488:326.

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

**NUREG-0750 V32 N05:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1990. Pages 333-393. * Division of Freedom of Information & Publications Services (Post 890205). January 1991. 69pp. 9102040314. 56571:212.

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

**NUREG-0750 V32 N06:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1990. Pages 395-496. * Division of Freedom of Information & Publications Services (Post 890205). February 1991. 109pp. 9102280236. 56837:023.

See NUREG-0750,V32,N05 abstract.

**NUREG-0750 V33 I01:** INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January-March 1991. * Division of Freedom of Information & Publications Services (Post 890205). June 1991. 50pp. 9107220263. 58489:153.

See NUREG-0750,V32,I02 abstract.

**NUREG-0750 V33 I02:** INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January-June 1991. * Division of Freedom of Information & Publications Services (Post 890205). October 1991. 87pp. 9110280071. 59448:208.

See NUREG-0750,V32,I02 abstract.

**NUREG-0750 V33 N01:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1991. Pages 1-60. * Division of Freedom of Information & Publications Services (Post 890205). March 1991. 67pp. 9103270034. 57158:181.

See NUREG-0750,V32,N05 abstract.

**NUREG-0750 V33 N02:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1991. Pages 61-173. * Division of Freedom of Information & Publications Services (Post 890205). April 1991. 121pp. 9105160092. 57727:282.

See NUREG-0750,V32,N05 abstract.

**NUREG-0750 V33 N03:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1991.Pages 175-232. * Division of Freedom of Information & Publications Services (Post 890205). May 1991. 64pp. 9105300258. 57869:308.

See NUREG-0750,V32,N05 abstract.

**NUREG-0750 V33 N04:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1991.Pages 233-293. * Division of Freedom of Information & Publications Services (Post 890205). June 1991. 69pp. 9107010138. 58250:114.

See NUREG-0750,V32,N05 abstract.

**NUREG-0750 V33 N05:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1991.Pages 295-459. * Division of Freedom of Information & Publications Services (Post 890205). August 1991. 174pp. 9109050275. 58989:011.

See NUREG-0750,V32,N05 abstract.

**NUREG-0750 V33 N06:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1991.Pages 461-619. * Division of Freedom of Information & Publications Services (Post 890205). August 1991. 168pp. 9110110237. 59358:124.

See NUREG-0750,V32,N05 abstract.

**NUREG-0750 V34 N01:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1991.Pages 1-148. * Division of Freedom of Information & Publications Services (Post 890205). September 1991. 156pp. 9110100235. 59345:039.

See NUREG-0750,V32,N05 abstract.

**NUREG-0750 V34 N02:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1991. Pages 149-183. * Division of Freedom of Information & Publications Services (Post 890205). October 1991. 41pp. 9110290328. 59455:202.

See NUREG-0750,V32,N05 abstract.

**NUREG-0750 V34 N03:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1991. Pages 185-228. * Division of Freedom of Information & Publications Services (Post 890205). November 1991. 52pp. 9201060237. 60202:206.

See NUREG-0750,V32,N05 abstract.

**NUREG-0750 V34 N04:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1991. Pages 229-260. * Division of Freedom of Information & Publications Services (Post 890205). December 1991. 38pp. 9201060349. 60202:168.

See NUREG-0750,V32,N05 abstract.

**NUREG-0837 V10 N04:** NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. October-December 1990. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). April 1991. 325pp. 9104290263. 57529:208.

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 of 1990.

**NUREG-0837 V11 N01:** NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. January-March 1991. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). July 1991. 239pp. 9107220305. 58494:064.

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 first quarter of 1991.

**NUREG-0837 V11 N02:** NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. April-June 1991. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). September 1991. 231pp. 9110080396. 59309:075.

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 1991.

**NUREG-0837 V11 N03:** NRC TLD DIR... RADIATION MONITORING NETWORK.Progress Report. July-September,1991. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). December 1991. 231pp. 9201090199. 60259:292.

This report provides the status and results of the NRC Thermoluminescent Dosimeter 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 1991.

**NUREG-0847 S06:** SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT,UNITS 1 AND 2.Docket Nos. 50-390 And 50-391.(Tennessee Valley Authority). TAM,P.S. Division of Reactor Projects - I/II (Post 870411). April 1991. 159pp. 9105150345. 57717:273.

Supplement No. 6 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. 5 was issued, and (2) matters that the staff had under review when Supplement No. 5 was issued.

**NUREG-0847 S07:** SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT,UNITS 1 AND 2.Docket Nos. 50-390 And 50-391.(Tennessee Valley Authority) TAM,P.S. Division of Reactor Projects - I/II (Post 870411). September 1991. 76pp. 9110090253. 59329:001.

Supplement No. 7 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. 6 was issued, and (2) matters that the staff had under review when Supplement No. 6 was issued.

**NUREG-0933 S01-12:** A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R.; RIGGS,R.; MILSTEAD,W.; et al. Division of Safety Issue Resolution (Post 880717). July 1991. 1,512pp. 9108130354. 58769:001.

See NUREG-0933,S12 abstract.

**NUREG-0933 S12:** A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R.; RIGGS,R.; MILSTEAD,W.; et al. Division of Regulatory Applications (Post 870413). January 1991. 265pp. 9101300169. 56536:018.

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 S13:** A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R.; RIGGS,R.; MILSTEAD,W.; et al. Division of Regulatory Applications (Post 870413). December 1991. 386pp. 9201140354. 60298:072.

See NUREG-0933,S12 abstract.

**NUREG-0936 V09 N04:** NRC REGULATORY AGENDA.Quarterly Report,October-December 1990. * Division of Freedom of Information & Publications Services (Post 890205). January 1991. 149pp. 9102250195. 56789:215.

The NRC Regulatory Agenda is a compilation of all rules on which the NRC has proposed or is considering action and all petitions for rulemaking which have been received by the Commission and are pending disposition by the Commission. The Regulatory Agenda is updated and issued each quarter.

**NUREG-0936 V10 N01:** NRC REGULATORY AGENDA.Quarterly Report,January-March 1991. * Division of Freedom of Information & Publications Services (Post 890205). April 1991. 156pp. 9105150343. 57717:117.

See NUREG-0936,V09,N04 abstract.

**NUREG-0936 V10 N02:** NRC REGULATORY AGENDA.Quarterly Report,April-June 1991. * Division of Freedom of Information & Publications Services (Post 890205). August 1991. 154pp. 9109050327. 58986:033.

See NUREG-0936,V09,N04 abstract.

**NUREG-0936 V10 N03:** NRC REGULATORY AGENDA.Quarterly Report,July-September 1991. * Division of Freedom of Information & Publications Services (Post 890205). October 1991. 150pp. 9201060291. 60201:111.

See NUREG-0936,V09,N04 abstract.

**NUREG-0940 V09 N04:** ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,October-December 1990. * Ofc of Enforcement (Post 870413). February 1991. 389pp. 9103050496. 56876:354.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (October - December 1990) 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 V10 N01:** ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,January-March 1991. * Ofc of Enforcement (Post 870413). May 1991. 255pp. 9105210085. 57807:117.

This compilation summarize significant enforcement actions that have been resolved during one quarterly period (January - March 1991) 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 V10 N02:** ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June 1991. * Ofc of Enforcement (Post 870413). July 1991. 392pp. 9109050330. 58985:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1991) 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 V10 N03:** ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,July-September 1991. * Ofc of Enforcement (Post 870413). November 1991. 224pp. 9201060294. 60202:258.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (July - September 1991) 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-0975 V08:** COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING.Annual Report For FY 1990. * Division of Engineering (Post 870413). March 1991. 335pp. 9104220205. 57450:295.

This report presents summaries of the research work performed during Fiscal Year 1990 by laboratories and organizations under contracts administered by the NRC's Materials Engineering Branch, Office of Nuclear Regulatory Research. Each contractor has written a more complete and detailed annual report of its work which can be obtained by writing to NRC; however, we believe it is useful to have a summary of each contractor's efforts for the year combined into one volume.

**NUREG-0980 V01 N01:** NUCLEAR REGULATORY LEGISLATION.101st Congress. * Office of the General Counsel (Post 860701). June 1991. 495pp. 9108190294. 58829:028.

This document is a compilation of nuclear regulatory legislation and other relevant material through the 101st Congress, 2nd 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 N01:** NUCLEAR REGULATORY LEGISLATION.101st Congress. * Office of the General Counsel (Post 860701). June 1991. 443pp. 9109100333. 59052:065.

See NUREG-0980,V01,N01 abstract.

**NUREG-1022 R01 DR FC:** EVENT REPORTING SYSTEMS 10 CFR 50.72 AND 50.73.Clarification Of NRC Systems And Guidelines for Reporting.Draft Report For Comment. BOARDMAN,J.R.; BOBE,P.E.; CROOKS,J.L.; et al. Office for Analysis & Evaluation of Operational Data, Director. September 1991. 229pp. 9110100246. 59336:078.

Revision 1 to NUREG-1022 provides clarification of the immediate notification requirements of Title 10 of the Code of Federal Regulations, Part 50, Section 50.72 (10 CFR 50.72), and the 30-day written licensee event report (LER) requirements of 10 CFR 50.73 for nuclear power plants. This revision was initiated to ensure events are reported as required by improving 10 CFR 50.72 and 50.73 reporting guidelines and to consolidate these guidelines into a single reference document. This document updates and supersedes NUREG-1022 and its Supplements 1 and 2. This document does not change the reporting requirements of 10 CFR 50.72 and 50.73.

**NUREG-1100 V07:** BUDGET ESTIMATES.Fiscal Years 1992-1993. * Division of Budget & Analysis (Post 890205). February 1991. 185pp. 9102110174. 56653:037.

This report contains the fiscal year budget justifications to Congress. The budget provides estimates for salaries and expenses and for the Office of the Inspector General for fiscal years 1992 - 1993.

**NUREG-1125 V12:** A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS.1990 Annual. * ACRS - Advisory Committee on Reactor Safeguards. April 1991. 142pp. 9105150335. 57718:072.

This compilation contains 31 ACRS reports submitted to the Commission or to the Executive Director for Operations during calendar year 1990. 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-1144 R02:** NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.Status And Accomplishments. * Division of Engineering (Post 870413). June 1991. 176pp. 9107220249. 58490:200.

A comprehensive Nuclear Plant Aging Research (NPAR) Program was implemented by the U.S. NRC Office of Nuclear Regulatory Research in 1985 to identify and resolve technical safety issues related to the aging of systems, structures, and components in operating nuclear power plants. This is Revision 2 to the NPAR Program Plan. This plan defines the goals of the program, the current status of research, and summarizes the utilization of the research results in the regulatory process. The plan also describes major milestones and schedules for coordinating research within the agency and with organizations and institutions outside the agency, both domestic and foreign. Currently, the NPAR Program comprises seven major areas: (1) hardware-oriented engineering research involving components and structures; (2) system-oriented aging interaction studies; (3) development of technical bases for license renewal rulemaking; (4) determining risk significance of aging phenomena; (5) development of technical bases for resolving generic safety issues; (6) recommendations for field inspection and maintenance addressing aging concerns; and (7) residual lifetime evaluations of major LWR components and structures. The NPAR technical database comprises approximately 100 NUREG/CR reports by June 1991, plus numerous published papers and proceedings that offer regulators and industry important insights to aging characteristics and aging management of safety-related equipment. Regulatory applications include revisions to and development of regulatory guides and technical specifications; support to resolve generic safety issues; development of codes and standards; evaluation of diagnostic techniques (e.g., for cables and valves); and technical support for development of the license renewal rule.

**NUREG-1145 V07:** U.S. NUCLEAR REGULATORY COMMISSION 1990 ANNUAL REPORT. * Office of Administration (Post 890205). July 1991. 254pp. 9108290260. 58910:354.

This report covers the major activities, events, decisions, and planning that took place during fiscal year 1990 within the U.S. Nuclear Regulatory Commission (NRC) or involving the NRC.

**NUREG-1150 V03:** SEVERE ACCIDENT RISKS: AN ASSESSMENT FOR FIVE U.S. NUCLEAR POWER PLANTS.Appendices D And E.Final Report. * Division of Systems Research (Post 880717). January 1991. 93pp. 9102200273. 56744:039.

This report summarizes an assessment of the risks from severe accidents in five commercial nuclear power plants in the United States. These risks are measured in a number of ways, including: the estimated frequencies of core damage accidents from internally initiated accidents, and externally initiated accidents for two of the plants; the performance of containment structures under severe accident loadings; the potential magnitude of radionuclide releases and offsite consequences of such accidents; and the overall risk (the product of accident frequencies and consequences). Supporting this summary report are a large number of reports written under contract to NRC which provide the detailed discussion of the methods used and results obtained in these risk studies. Volume 3 of this report contains two appendices. Appendix D summarizes comments received and staff responses on the first (February 1987) draft of NUREG-1150. Appendix E provides a similar summary of comments and responses, but for the second (June 1989) version of the report.

**NUREG-1199 R02:** STANDARD FORMAT AND CONTENT OF A LICENSE APPLICATION FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY. * Division of Low-Level Waste Management & Decommissioning (Post 870413). January 1991. 30pp. 9102200275. 56744:132.

The Standard Format and Content of a License Application for a Low-Level Radioactive Waste Disposal Facility, NUREG-1199, discusses the information to be provided in the Safety Analysis Report and establishes a uniform format for presenting the information required to meet the licensing requirements for land disposal of radioactive waste as required by 10 CFR 61. The use of the Standard Format will (1) help ensure that the Safety Analysis Report (SAR) contains the information required by 10 CFR 61, (2) aid the applicant in ensuring that the information is complete, (3) help persons reading the SAR to locate information, and (4) contribute to shortening the time required for the review process. The Standard Format and Content (NUREG-1199) ensures that the information required to perform the review is provided, and in a usable format while the Standard Review Plan, NUREG-1200, defines the technical review process.

**NUREG-1200 R02:** STANDARD REVIEW PLAN FOR THE REVIEW OF A LICENSE APPLICATION FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY. * Division of Low-Level Waste Management & Decommissioning (Post 870413). January 1991. 185pp. 9102200281. 56743:214.

The Standard Review Plan (SRP) is prepared for the guidance of staff reviewers in the Office of Nuclear Material Safety and Safeguards in performing safety reviews of applications to construct and operate a low-level waste disposal facility. The principal purpose of the SRP is to assure the quality and uniformity of staff reviews and to present a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. It is also a purpose of the SRP to make information about regulatory matters widely available and to improve communication and understanding of the staff's review process by interested members of the public and the nuclear industry. NUREG-1200 consists of 11 chapters containing approximately 60 individual SRP sections. Each section identifies who performs the review, the matters that are reviewed, the basis for review, how the review is performed, and the conclusions that are sought.

**NUREG-1214 R07:** HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. ALLENSPACH,F. Division of Licensee Performance & Quality Evaluation (Post 870411). February 1991. 117pp. 9103120085. 56948:152.

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 R08:** HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. ALLENSPACH,F. Division of Licensee Performance & Quality Evaluation (Post 870411). August 1991. 122pp. 9110090309. 59331:209.

See NUREG-1214,R07 abstract.

**NUREG-1232 V03 S02:** SAFETY EVALUATION REPORT ON TENNESSEE VALLEY AUTHORITY: BROWNS FERRY NUCLEAR PERFORMANCE PLAN.Browns Ferry Unit 2 Restart. ROSS,T.M. Division of Reactor Projects - I/II (Post 870411). January 1991. 123pp. 9102110178. 56652:274.

This safety evaluation report (SER) was prepared by the U.S. Nuclear Regulatory Commission (NRC) staff and represents the second and last supplement (SSER 2) to the staff's original SER published as Volume 3 of NUREG-1232 in April 1989. Supplement 1 of Volume 3 of NUREG-1232 (SSER 1) was published in October 1989. Like its predecessors, SSER 2 is composed of numerous safety evaluations by the staff regarding specific elements contained in the Browns Ferry Nuclear Performance Plan (BFNPP), Volume 3 (up to and including Revision 2), submitted by the Tennessee Valley Authority (TVA) for the Browns Ferry Nuclear Plant (BFN). The Browns Ferry Nuclear Plant consists of three boiling-water reactors (BWRs) at a site in Limestone County, Alabama. The BFNPP describes the corrective action plans and commitments made by TVA to resolve deficiencies with its nuclear programs before the startup of Unit 2. The staff has inspected and will continue to inspect TVA's implementation of these BFNPP corrective action plans that address staff concerns about TVA's nuclear programs. SSER 2 documents the NRC staff's safety evaluations and conclusions for those elements of the BFNPP that were not previously addressed by the staff or that remained open as a result of unresolved issues identified by the staff in previous SERs and inspections.

**NUREG-1266 V05:** NRC SAFETY RESEARCH IN SUPPORT OF REGULATION - FY 1990. * Office of Nuclear Regulatory Research (Post 860720). April 1991. 69pp. 9105160056. 57728:102.

This report, the sixth 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 1990. The goal of this office is to ensure that safety-related research provides the technical bases for rulemaking and for related decisions in support of NRC licensing and inspection activities. This research is necessary to make certain that the regulations that are imposed on licensees provide an adequate margin of safety so as to protect the health and safety of the public. This report describes both the direct contributions to scientific and technical knowledge with regard to nuclear safety and their regulatory applications.

**NUREG-1272 V05 N01:** OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA.1990 Annual Report - Power Reactors. * Office for Analysis & Evaluation of Operational Data, Director. July 1991. 245pp. 9108130351. 58773:088.

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 1990. The report is published in two separate parts. NUREG-1272, Vol. 5, 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. 5, No. 2, covers nonreactors and presents a review of the events and concerns during 1990 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 1980-1989.

**NUREG-1272 V05 N02:** OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA 1990 Annual Report - Nonreactors. * Office for Analysis & Evaluation of Operational Data, Director. July 1991. 96pp. 9109050303. 58988:134.

See NUREG-1272,V05,N01 abstract.

**NUREG-1275 V06:** OPERATING EXPERIENCE FEEDBACK REPORT - SOLENOID-OPERATED VALVE PROBLEMS.Commercial Power Reactors. ORNSTEIN,H.L. Office for Analysis & Evaluation of Operational Data, Director. February 1991. 116pp. 9103040368. C90-01. 56860:174.

This report highlights significant operating events involving observed or potential common-mode failures of solenoid-operated valves (SOVs) in U.S. plants. These events resulted in degradation or malfunction of multiple trains of safety systems as well as of multiple safety systems. On the basis of the evaluation of these events, the Office for Analysis and Evaluation of Operational Data (AEOD) of the U.S. Nuclear Regulatory Commission (NRC) concludes that the problems with solenoid-operated valves are an important issue that needs additional NRC and industry attention. This report also provides AEOD's recommendations for actions to reduce the occurrence of SOV common-mode failures.

**NUREG-1293 R01:** QUALITY ASSURANCE GUIDANCE FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY. PITTIGLIO,C.L.; HEDGES,D. Division of Low-Level Waste Management & Decommissioning (Post 870413). April 1991. 22pp. 9105030134. 57617:290.

This document provides guidance to an applicant on meeting the quality control (QC) requirements of 10 CFR 61.12(j) for a low-level radioactive waste (LLRW) disposal facility. The QC requirements, plus audits and managerial controls requirements, establish the need for developing a quality assurance (QA) program and the guidance provided herein. The criteria developed for this document are similar to the criteria developed for Appendix B to Title 10 of the Code of Federal Regulations (10 CFR) Part 50. Although Appendix B is not a regulatory requirement for an LLRW disposal facility, the criteria that were developed for 10 CFR Part 50 are basic to any QA program. This document establishes QA guidance for the design, construction, and operation of those structures, engineered or natural systems, and components whose function is required to meet the performance objectives of Subpart C of 10 CFR Part 61 and to limit exposure to or release of radioactivity.

**NUREG-1301:** OFFSITE DOSE CALCULATION MANUAL GUIDANCE: STANDARD RADIOLOGICAL EFFLUENT CONTROLS FOR PRESSURIZED WATER REACTORS.Generic Letter 89-01,Supplement No. 1. MEINKE,W.W.; ESSIG,T.H. Division of Radiation Protection & Emergency Preparedness (Post 870411). April 1991. 122pp. 9107100056. 58384:001.

This report contains guidance which may be voluntarily used by licensees who choose to implement the provision of Generic Letter 89-01, which allows Radiological Effluent Technical Specifications (RETS) to be removed from the main body of the Technical Specifications and placed in the Offsite Dose Calculation Manual (ODCM). Guidance is provided for standard effluent controls definitions, controls for effluent monitoring instrumentation, controls for effluent releases, controls for radiological environmental monitoring, and the basis for controls. Guidance on the formulation of RETS has been available in draft form (NUREG-0472 and -0473) for a number of years; the current effort simply recasts those RETS into standard radiological effluent controls for application to the ODCM. Also included for completeness are: (1) radiological environmental monitoring program guidance previously which had been available as a Branch Technical Position (Rev. 1, November 1979); (2) existing ODCM guidance; and (3) a reproduction of Generic Letter 89-01.

**NUREG-1302:** OFFSITE DOSE CALCULATION MANUAL GUIDANCE: STANDARD RADIOLOGICAL EFFLUENT CONTROLS FOR BOILING WATER REACTORS. Generic Letter 89-01,Supplement No. 1. MEINKE,W.W.; ESSIG,T.H. Division of Radiation Protection & Emergency Preparedness (Post 870411). April 1991. 127pp. 9106180015. 58131:030.

This report contains guidance which may be voluntarily used by licensees who choose to implement the provision of Generic Letter 89-01, which allows Radiological Effluent Technical Specifications (RETS) to be removed from the main body of the Technical Specifications and placed in the Offsite Dose Calculation Manual (ODCM). Guidance is provided for standard effluent controls definitions, controls for effluent monitoring instrumentation, controls for effluent releases, controls for radiological environmental monitoring, and the basis for controls. Guidance on the formulation of RETS has been available in draft form (NUREG-0472 and -0473) for a number of years; the current effort simply recasts those RETS into standard radiological effluent controls for application to the ODCM. Also included for completeness are: (1) radiological environmental monitoring program guidance previously which had been available as a Branch Technical Position (Rev. 1, November 1979); (2) existing ODCM guidance; and (3) a reproduction of Generic Letter 89-01.

**NUREG-1303 R01:** INCIDENT INVESTIGATION MANUAL. * Office for Analysis & Evaluation of Operational Data, Director. November 1991. 122pp. 9201060103. 60212:001.

The Incident Investigation Manual prescribes guidelines for the conduct of investigative activities of the U.S. Nuclear Regulatory Commission (NRC) Incident Investigation Teams (IITs). The purpose of this manual is to provide IITs guidance to ensure that NRC investigations of significant events are timely, structured, coordinated, and formally administered. The guidelines are intended to assist the investigation rather than limit the initiatives and good judgment of the IIT leader or members. The IIT leader and team members should use their experience and those techniques that provide the most confidence in assuring the IIT objectives are achieved. These guidelines address IIT activation, conduct of the investigation, conducting interviews, treatment of quarantined equipment, preparation of the team report and followup of staff actions.

**NUREG-1307 R02:** REPORT ON WASTE BURIAL CHARGES.Escalation Of Decommissioning Waste Disposal Costs At Low-Level Waste Burial Facilities. * Division of Regulatory Applications (Post 870413). July 1991. 42pp. 9108130285. 58765:108.

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 2 of this report corrects several errors in the calculations and disposal costs for the reference PWR and the reference BWR.

**NUREG-1321:** TESTING STANDARDS FOR PHYSICAL SECURITY SYSTEMS AT CATEGORY I FUEL CYCLE FACILITIES. DWYER,P.A. Division of Safeguards & Transportation (Post 870413). October 1991. 27pp. 9110280044. 59455:043.

This report is a compilation of physical security testing standards for use at fuel cycle facilities using or possessing formula quantities of strategic special nuclear material.

**NUREG-1322**: ACCEPTANCE CRITERIA FOR THE EVALUATION OF CATEGORY I FUEL CYCLE FACILITY PHYSICAL SECURITY PLANS. DWYER,P.A. Division of Safeguards & Transportation (Post 870413). October 1991. 32pp. 9110280054. 59455:007.

This report presents criteria developed from U.S. Nuclear Regulatory Commission regulations for the evaluation of physical security plans submitted by Category I fuel facility licensees. Category I refers to those licensees who use or possess a formula quantity of strategic special nuclear material.

**NUREG-1350 V03**: NUCLEAR REGULATORY COMMISSION INFORMATION DIGEST.1991 Edition. OLIVE,K.L. Division of Budget & Analysis (Post 890205). March 1991. 102pp. 9104260009. 57529:029.

The Nuclear Regulatory Commission Information Digest provides a summary of information about the U.S. Nuclear Regulatory Commission (NRC), NRC's regulatory responsibilities, and the areas NRC licenses. This digest is a compilation of 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 1990, with exceptions noted. For operating U.S. commercial nuclear power reactors, information on generating capacity and average capacity factor is obtained from monthly operating reports submitted to the NRC directly by the licensee. This information is reviewed for consistency only. No independent validation and/or verification is performed by the NRC. For detailed and complete information about tables and figures, refer to the source publications. This digest is published annually for the general use of the NRC staff and is available to the public.

**NUREG-1362**: REGULATORY ANALYSIS FOR FINAL RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL.Final Report. * Division of Safety Issue Resolution (Post 880717). December 1991. 200pp. 9201060099. 60197:184.

This regulatory analysis provides the supporting information for the final rule (10 CFR Part 54) that defines the Nuclear Regulatory Commission's requirements for renewing the operating licenses of commercial nuclear power plants. A set of four specific alternatives for the safety review of license renewal applications is defined and evaluated. These are: Alternative A--current licensing basis; Alternative B--extension of Alternative A to require assessment and managing of aging; Alternative C--extension of Alternative B to require assessment of design differences against selected new-plant standards using probabilistic risk assessment; and Alternative D--extension of Alternative B to require compliance with all new-plant standards. A quantitative comparison of the four alternatives in terms of impact-to-value ratio is presented, and Alternative B is the most cost-beneficial safety review alternative.

**NUREG-1363 V03**: ATOMIC SAFETY AND LICENSING BOARD PANEL ANNUAL REPORT.Fiscal Year 1990. COTTER,B.P. Atomic Safety & Licensing Board Panel. September 1991. 33pp. 9111070099. 59549:152.

In Fiscal Year 1990, the Atomic Safety and Licensing Board Panel (Panel) handled 40 proceedings involving the construction, operation, and maintenance of commercial nuclear power reactors or other activities requiring a license from the Nuclear Regulatory Commission. This report summarizes, highlights, and analyzes how the wide-ranging issues raised in these proceedings were addressed by the Judges and Licensing Boards of the Panel during the year.

**NUREG-1369**: PREAPPLICATION SAFETY EVALUATION REPORT FOR THE SODIUM ADVANCED FAST REACTOR (SAFR) LIQUID METAL REACTOR. KING,T.L. Office of Nuclear Regulatory Research (Post 860720). LANDRY,R.R.; THROM,E.D.; et al. Office of Nuclear Reactor Regulation, Director (Post 870411). December 1991. 267pp. 9201060347. 60201:261.

This safety evaluation report (SER) presents the final results of a preapplication design review for the Sodium Advanced Fast Reactor (SAFR) liquid metal reactor (Project 673). The SAFR conceptual design was submitted by the U.S. Department of Energy (DOE) in accordance with the U.S. Nuclear Regulatory Commission (NRC) "Statement of Policy for the Regulation of Advanced Nuclear Power Plants" (51 FR 24643) which provides for the early Commission review and interaction. The standard SAFR plant design consists of four identical reactor modules, referred to as "paks," each with a thermal output rating of 900 MWt, coupled with four steam turbine-generator sets. The total electrical output was to be 1400 MWe. This SER represents the NRC staff's preliminary technical evaluation of the safety features in the SAFR design. It must be recognized that final conclusions in all matters discussed in this SER require approval by the Commission. During the NRC staff review of the SAFR conceptual design, DOE terminated work on this design in September 1988. This SER documents the work done to that date and no additional work is planned for the SAFR.

**NUREG-1374**: TECHNICAL FINDINGS REL[illegible] TO GENERIC ISSUE 79.An Evaluation Of PWR Rea[illegible]essel Thermal Stress During Natural Convection Cooldo[illegible]GE,J.D. Division of Safety Issue Resolution (Post 88071[illegible] [illegible]ay 1991. 149pp. 9106180011. 58130:241.

This report summarizes work performed [illegible] the Nuclear Regulatory Commission staff to resolve Generic Issue 79, "Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown (NCC)." The report evaluates the effects of an NCC event on PWR reactor vessels (RVs), with particular emphasis on the closure flange region. A conservative independent confirmatory stress analysis of a B&W 177- fuel-assembly RV (B&W 177) was performed by the NRC contractor, and an independent fracture mechanics evaluation was performed by the staff. Based on these and a comparison of geometric similarity between the B&W 177 and other PWR RVs, the NRC staff developed findings that are applicable to all U.S. PWRs.

**NUREG-1375 V02**: SAFETY EVALUATION REVIEW OF THE PROTOTYPE LICENSE APPLICATION SAFETY ANALYSIS REPORT.Belowground Vault. * Division of Low-Level Waste Management & Decommissioning (Post 870413). September 1991. 52pp. 9110100255. 59336:344.

The U.S. Nuclear Regulatory Commission (NRC) staff and consultants reviewed a Prototype License Application Safety Analysis Report (PLASAR) submitted by the U.S. Department of Energy (DOE) for the belowground vault (BGV) alternative method of low-level radioactive waste disposal. In Volume 1 of NUREG-1375, the NRC staff provided the safety review results for an earth-mounded concrete bunker PLASAR. In the current report, the staff focused its review on the design, construction, and operational aspects of the BGV PLASAR. The staff developed review comments and questions using the Standard Review Plan (SRP), Rev. 1 (NUREG-1200) as the basis for evaluating the acceptability of the information provided in the BGV PLASAR. The detailed review comments provided in this report are intended to be useful guidance to facility developers and State regulators in addressing issues likely to be encountered in the review of a license application for a low-level- waste disposal facility.

**NUREG-1377 R02**: NRC RESEARCH PROGRAM ON PLANT AGING: LISTING AND SUMMARIES OF REPORTS ISSUED THROUGH JUNE 1991. KONDIC,N.N.; HILL,E.L. Division of Engineering (Post 870413). July 1991. 92pp. 9108130174. 58746:214.

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 June 1991 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-1382**: SAFETY EVALUATION REPORT RELATED TO THE FULL-TERM OPERATING LICENSE FOR OYSTER CREEK NUCLEAR GENERATING STATION.Docket No. 50-219.(General Public Utilities Nuclear Corp.,et al) * Division of Reactor Projects - I/II (Post 870411). January 1991. 161pp. 9101300226. 56534:121.

The Safety Evaluation Report for the full-term operating license application filed by GPU Nuclear Corporation and Jersey Central Power & Light Company for the Oyster Creek Nuclear Generating Station has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Ocean County, New Jersey. The staff concludes that the facility can continue to be operated without endangering the health and safety of the public.

**NUREG-1391**: CHEMICAL TOXICITY OF URANIUM HEXAFLUORIDE COMPARED TO ACUTE EFFECTS OF RADIATION.Final Report. MCGUIRE,S.A. Division of Regulatory Applications (Post 870413). February 1991. 17pp. 9103110209. 56942:190.

The chemical effects from acute exposures to uranium hexafluoride are compared to the nonstochastic effects from acute radiation doses of 25 rems to the whole body and 300 rems to the thyroid. The analysis concludes that an intake of about 10 mg of uranium in soluble form is roughly comparable, in terms of early effects, to an acute whole body dose of 25 rems because both are just below the threshold for significant nonstochastic effects. Similarly, an exposure to hydrogen fluoride at a concentration of 25 mg/m(3) for 30 minutes is roughly comparable because there would be no significant nonstochastic effects. For times other than 30 minutes, the concentration C of hydrogen fluoride considered to have the same effect can be calculated using a quadratic equation: C = 25 mg/m(3) (30 min/t)(0.5). The purpose of these analyses is to provide information for developing design and siting guidelines based on chemical toxicity for enrichment plants using uranium hexafluoride. These guidelines are to be similar, in terms of stochastic health effects, to criteria in NRC regulations for nuclear power plants, which are based on radiation doses.

**NUREG-1397**: AN ASSESSMENT OF DESIGN CONTROL PRACTICES AND DESIGN RECONSTITUTION PROGRAMS IN THE NUCLEAR POWER INDUSTRY. IMBRO,E.V. Division of Reactor Inspection & Safeguards (Post 870411). February 1991. 109pp. 9103120091. 56961:249.

This document summarizes the results of a survey of nuclear power plant design control practices and design reconstitution efforts conducted during 1989 at six utilities and with one nuclear steam supply vendor. Conclusions and observations resulting from the survey assessments are provided so that utilities and the NRC can consider actions to improve these programs.

**NUREG-1398**: ENVIRONMENTAL ASSESSMENT FOR FINAL RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL.Final Report. * Division of Safety Issue Resolution (Post 880717). December 1991. 64pp. 9201060281. 60201:052.

The possible environmental effects of promulgating nuclear power plant license renewal standards by the final rule, 10 CFR Part 54, rather than applying requirements in an ad hoc manner in individual licensing actions, are assessed. The rule requires the development of information and analyses to identify aging problems of systems, structures and components that will be of concern during the renewal term and will not be controlled by existing regulatory programs. Required actions may be replacement, refurbishment, inspection, testing or monitoring. Such actions will generally be within the range of similar actions taken for plants during the initial operating term. They would be primarily confined within the plants with potential for only minor disruption to the environment. It is unlikely that these actions would change the environmental effects already being experienced. The promulgation of 10 CFR Part 54 has clear advantages relative to regulatory stability and administrative efficiency. However, it will not result in environmental effects significantly different from those arising from relicensing under existing regulations. The NRC concludes that promulgation of 10 CFR Part 54 would not significantly affect the environment and, therefore, a full environmental impact statement is not required and a Finding of No Significant Impact can be made.

**NUREG-1400 DRFT FC**: AIR SAMPLING IN THE WORKPLACE.Draft Report For Comment. MCGUIRE,S.A. Division of Regulatory Applications (Post 870413). HICKEY,E.E.; STOETZEL,G.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1991. 97pp. 9111070110. 59549:186.

NUREG-1400 is a support document for Revision 1 of Regulatory Guide 8.25, "Air Sampling in the Workplace." The document addresses the aspects of designing, installing, and implementing an air sampling program at a facility licensed by the U.S. Nuclear Regulatory Commission to meet the requirements in the revision of 10 CFR 20. Determination of the need for air sampling is addressed using an evaluation process termed a hazard index calculation. Performance of a hazard index calculation will suggest the type and level of sampling needed to adequately assess the workplace air concentrations. Guidance is also provided on types of sampling available including a general air sampling, breathing zone air sampling, and early warning air sampling. In addition, location of samplers, statistical tests for determining sampling adequacy, derived air concentration adjustment for particle size analysis, calibration for volume of air sampled, quality control, and evaluation of program adequacy are discussed.

**NUREG-1401 DRFT FC**: REGULATORY ANALYSIS FOR GENERIC ISSUE 23: REACTOR COOLANT PUMP SEAL FAILURE.Draft Report For Comment. SHAUKAT,S.K.; JACKSON,J.E.; THATCHER,D.F. Division of Safety Issue Resolution (Post 880717). April 1991. 77pp. 9104250020. 57489:033.

This report presents the regulatory/backfit analysis for Generic Issue 23 (GI-23), "Reactor Coolant Pump Seal Failure." The regulatory analysis includes quality assurance provisions for reactor coolant pump seals, instrumentation and procedures for monitoring seal performance, and provisions for seal cooling during off-normal plant conditions involving loss of all seal cooling such as station blackout.

**NUREG-1407**: PROCEDURAL AND SUBMITTAL GUIDANCE FOR INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR SEVERE ACCIDENT VULNERABILITIES.Final Report. CHEN,J.T.; CHOKSHI,N.C.; KENNEALLY,R.M.; et al. Division of Safety Issue Resolution (Post 880717). June 1991. 81pp. 9107230240. 58494:303.

Based on a Policy Statement on Severe Accidents, the licensee of each nuclear power plant is requested to perform an individual plant examination. The plant examination systematically looks for vulnerabilities to severe accidents and cost-effective safety improvements that reduce or eliminate the important vulnerabilities. This document presents guidance for performing and reporting the results of the individual plant examination of external events. The guidance for reporting the results of the individual plant examination of internal events (IPE) is presented in NUREG-1335.

**NUREG-1412:** FOUNDATION FOR THE ADEQUACY OF THE LICENSING BASES.A Supplement To The Statement Of Considerations For The Rule On Nuclear Power Plant License Renewal (10 CFR Part 54). Final Report. * Office of Nuclear Reactor Regulation, Director (Post 870411). December 1991. 119pp. 9201060078. 60196:241.

The objective of this report is to describe the regulatory processes that assures that any plant-specific licensing bases will provide reasonable assurance that the operation of nuclear power plants will not be inimical to the public health and safety to the end of renewal period. It is on the adequacy of this process that the Commission has determined that a formal renewal licensing review against the full range of current safety requirements would not add significantly to safety and is not needed to assure that continued operation throughout the renewal term is not inimical to the public health and safety or common defense and security. This document illustrates in general terms how the regulatory process has evolved in major safety issue areas. It also provides examples illustrating why it is unnecessary to re-review an operating plant's basis, except for age-related degradation unique to license renewal, at the time of license renewal. The report is a supplement to the Statement of Considerations for the Nuclear Regulatory Commission's rule (10 CFR Part 54) that establishes the criteria and standards governing nuclear power plant license renewal.

**NUREG-1413:** SAFETY EVALUATION REPORT RELATED TO THE PRELIMINARY DESIGN OF THE STANDARD NUCLEAR STEAM SUPPLY REFERENCE SYSTEM,RESAR SP/90.Docket No. 50-601.(Westinghouse Electric Corporation, Inc.) * Division of Advanced Reactors & Special Projects (Post 901216). April 1991. 398pp. 9105220026. 57824:160.

This report provides the results of the NRC staff review of the Westinghouse Electric Corporation for a preliminary design approval of the SP/90 reactor contained in its reference safety analysis report. The standard safety analysis report describing the design of the facility was submitted from October 24, 1983 through March 9, 1987. Staff of the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, prepared this safety evaluation report of the RESAR SP/90. Based on its review, the staff concludes that there are open issues that, because of the stage of the design, have not been resolved at this stage of review. These issues are discussed in detail throughout this report, and a summary is provided in Section 1.6 of this report.

**NUREG-1415 V03 N02:** OFFICE OF THE INSPECTOR GENERAL.Semiannual Report.October 1990 - March 1991. * Office of the Inspector General (Post 890417). April 1991. 34pp. 9107030211. 58265:001.

Inspectors General are required, by the IG Act of 1978, as amended, to prepare semiannual reports which summarize 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 as required by the IG Act, and transmits the report to Congress.

**NUREG-1415 V04 N01:** OFFICE OF THE INSPECTOR GENERAL.Semiannual Report.April-September 1991. GLENN,W.L.; WATKINS,R.A.; HUBER,D.S. Office of the Inspector General (Post 890417). October 1991. 36pp. 9201090191. 60243:229.

Inspectors General are required, by the IG Act of 1978, as amended, to prepare semiannual reports which summarize 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 as required by the IG Act, and transmits the report to Congress.

**NUREG-1421:** REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC ISSUE 130: ESSENTIAL SERVICE WATER SYSTEM FAILURES AT MULTI-UNIT SITES. LEUNG,V.; BASDEKAS,D.; MAZETIS,G. Division of Safety Issue Resolution (Post 880717). June 1991. 34pp. 9107080230. 58307:099.

The essential service water system (ESWS) is required to provide cooling to nuclear power plants during normal operation and accident conditions. The ESWS typically supports component cooling water heat exchangers, containment spray heat exchangers, high-pressure injection pump oil coolers, emergency diesel generators, and auxiliary building ventilation coolers. Failure of the ESWS function could lead to severe consequences. This report presents the regulatory analysis for GI-130, "Essential Service Water System Failures at Multi-Unit Sites." The risk reduction estimates, cost/benefit analyses, and other insights gained during this effort have shown that implementation of the recommendations will significantly reduce risk and that these improvements are warranted in accordance with the backfit rule, 10CFR50.109(a)(3).

**NUREG-1423 V02:** A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON NUCLEAR WASTE.July 1990 - June 1991. * Advisory Committee on Nuclear Waste. August 1991. 98pp. 9108290265. 58929:237.

This compilation contains 20 reports issued by the Advisory Committee on Nuclear Waste during the third year of its operation. The reports were submitted to the Chairman, U.S. Nuclear Regulatory Commission, or to the Director, 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-1426 V01:** COMPILATION OF REPORTS FROM RESEARCH SUPPORTED BY THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING. 1965 - 1990. HISER,A.L. Division of Engineering (Post 870413). May 1991. 55pp. 9106200222. 57867:261.

Since 1965, the Materials Engineering Branch, Division of Engineering, of the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research, and its predecessors dating back to the Atomic Energy Commission (AEC), has sponsored research programs concerning the integrity of the primary system pressure boundary of light water reactors. The components of concern in these research programs have included the reactor pressure vessel (RPV), steam generators, and the piping. These research programs have covered a broad range of topics, including fracture mechanics analysis and experimental work for RPV and piping applications, inspection method development and qualification, and evaluation of irradiation effects to RPV steels. This report provides as complete a listing as practical of formal technical reports submitted to the NRC by the investigators working on these research programs. This listing includes topical, final and progress reports, and is segmented by topic area. In many cases a report will cover several topics (such as in the case of progress reports of multi-faceted programs), but is listed under only one topic. Therefore, in searching for reports on a specific topic, other related topic areas should be checked also.

**NUREG-1428:** ANALYSIS OF PUBLIC COMMENTS ON THE PROPOSED RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL. * Division of Safety Issue Resolution (Post 880717). December 1991. 370pp. 9201060074. 60195:231.

This report provides a summary and analysis of public comments on the proposed license renewal rule for nuclear power plants (10 CFR Part 54) published in the Federal Register on 17

July 1990. It also documents the NRC's resolution of the issues raised by the commenters. Comments from 121 organizations and 76 individuals were reviewed and analyzed to identify the issues, including those pertaining to the adequacy of the licensing basis, the performance of an integrated plant assessment, backfit considerations, and need for public hearings. The analysis included grouping of commenters' views according to the issues raised. The public comments analyzed in this report were taken into consideration in the development of the final rule and revisions to the supporting documents.

**NUREG-1429 DRFT FC**: ENVIRONMENTAL STANDARD REVIEW PLAN FOR THE REVIEW OF LICENSE RENEWAL APPLICATIONS FOR NUCLEAR POWER PLANTS.Draft Report For Comment. * Division of Advanced Reactors & Special Projects (Post 901216). August 1991. 97pp. 9109300073. 59227:215.

The Environmental Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants (ESRP-LR) is to be used by the NRC staff when performing environmental reviews of applications for the renewal of power reactor licenses. The use of the ESRP-LR provides a framework for the staff to determine whether or not environmental issues important to license renewal have been identified and the impacts evaluated and provides acceptance standards to help the reviewers comply with the National Environmental Policy Act.

**NUREG-1430 V1 DRF FC**: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Specifications.Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 421pp. 9102190210. 56755:013.

This draft report documents the results of the NRC staff review of new Standard Technical Specifications (STS) proposed by the Babcock and Wilcox Owners Group. The new STS were developed based on the criteria in the interim Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, dated February 6, 1987. The new STS will be used as bases for individual nuclear power plant owners to develop improved plant-specific technical specifications. The NRC staff is issuing this draft new STS for a 30 working-day comment period. Following the comment period, the NRC staff will analyze comments received, finalize the new STS, and issue them for plant-specific implementation. This report contains three volumes. Volume 1 contains the Specifications for all sections of the new STS. Volume 2 contains the Bases for Sections 2.0 - 3.3 of the new STS and Volume 3 contains the Bases for Sections 3.4 - 3.9 of the new STS.

**NUREG-1430 V2 DRF FC**: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Bases (Sections 2.0 - 3.3).Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 370pp. 9102190212. 56756:074.

See NUREG-1430,V01,DRF,FC abstract.

**NUREG-1430 V3 DRF FC**: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Bases (Sections 3.4 - 3.9).Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 471pp. 9102190214. 56757:084.

See NUREG-1430,V01,DRF,FC abstract.

**NUREG-1431 V1 DRF FC**: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Specifications.Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 484pp. 9102140299. 56696:057.

This draft report documents the results of the NRC staff review of new Standard Technical Specifications (STS) proposed by the Westinghouse Owners Group. The new STS were developed based on the criteria in the interim Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, dated February 6, 1987. The new STS will be used as bases for individual nuclear power plant owners to develop improved plant-specific technical specifications. The NRC staff is issuing this draft new STS for a 30 working-day comment period. Following the comment period, the NRC staff will analyze comments received, finalize the new STS, and issue them for plant-specific implementation. This report contains three volumes. Volume 1 contains the Specifications for all sections of the new STS. Volume 2 contains the Bases for Sections 2.0 - 3.3 of the new STS and Volume 3 contains the Bases for Sections 3.4 - 3.9 of the new STS.

**NUREG-1431 V2 DRF FC**: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS.Bases (Sections 2.0-3.3).Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 387pp. 9102140309. 56697:181.

See NUREG-1431,V01,DRF,FC abstract.

**NUREG-1431 V3 DRF FC**: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS.Bases (Sections 3.4-3.9).Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 644pp. 9102140312. 56694:101.

See NUREG-1431,V01,DRF,FC abstract.

**NUREG-1432 V1 DRF FC**: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS.Specifications.Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 518pp. 9102200257. 56753:020.

This draft report documents the results of the NRC staff review of new Standard Technical Specifications (STS) proposed by the Combustion Engineering Owners Group. The new STS were developed based on the criteria in the interim Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, dated February 6, 1987. The new STS will be used as bases for individual nuclear power plant owners to develop improved plant-specific technical specifications. The NRC staff is issuing this draft new STS for a 30 working-day comment period. Following the comment period, the NRC staff will analyze comments received, finalize the new STS and issue them for plant-specific implementation. This report contains three volumes. Volume 1 contains the Specifications for all sections of the new STS. Volume 2 contains the Bases for Sections 2.0 - 3.3 of the new STS and Volume 3 contains the Bases for Sections 3.4 - 3.9 of the new STS.

**NUREG-1432 V2 DRF FC**: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS.Bases (Sections 2.0 - 3.3).Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 560pp. 9102200262. 56751:180.

See NUREG-1432,V01,DRF,FC abstract.

**NUREG-1432 V3 DRF FC**: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS.Bases (Sections 3.4 - 3.9).Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 528pp. 9102200266. 56750:012.

See NRUEG-1432,V01,DRF,FC abstract.

**NUREG-1433 V1 DRF FC**: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC UNITS, BWR/4.Specifications.Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 485pp. 9102140324. 56704:290.

This draft report documents the results of the NRC staff review of new Standard Technical Specifications (STS) proposed by the BWR Owners Group for the BWR/4 design. The new STS were developed based on the criteria in the interim Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, dated February 6, 1987. The new STS will be used as bases for individual nuclear power plant owners to develop improved plant-specific technical specifications. The NRC staff is issuing this draft new STS for a 30 working-day comment period. Following the comment period, the NRC staff will analyze comments received, finalize the new STS, and issue them for plant-specific implementation. This report contains three volumes. Volume 1 contains the Specifications for all sections of the new STS. Volume 2 contains the Bases for Sections 2.0 - 3.3 of the new STS and Volume 3 contains the Bases for Sections 3.4 - 3.10 of the new STS.

**NUREG-1433 V2 DRF FC:** STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC UNITS, BWR/4.Bases (Sections 2.0 - 3.3).Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 419pp. 9102140322. 56703:231.

See NUREG-1433,V01,DRF,FC abstract.

**NUREG-1433 V3 DRF FC:** STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC UNITS, BWR/4.Bases (Sections 3.4 - 3.10).Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 475pp. 9102140295. 56702:116.

See NUREG-1433,V01,DRF,FC abstract.

**NUREG-1434 V1 DRF FC:** STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6.Specifications.Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 497pp. 9102200248. 56747:068.

This draft report documents the results of the NRC staff review of new Standard Technical Specifications (STS) proposed by the BWR Owners Group for the BWR/6 design. The new STS were developed based on the criteria in the interim Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, dated February 6, 1987. The new STS will be used as bases for individual nuclear power plant owners to develop improved plant-specific technical specifications. The NRC staff is issuing this draft new STS for a 30 working-day comment period. Following the comment period, the NRC staff will analyze comments received, finalize the new STS, and issue them for plant-specific implementation. This report contains three volumes. Volume 1 contains the Specifications for all sections of the new STS. Volume 2 contains the Bases for Sections 2.0 - 3.3 of the new STS and Volume 3 contains the Bases for Sections 3.4 - 3.10 of the new STS.

**NUREG-1434 V2 DRF FC:** STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6.Bases (Sections 2.0 - 3.3).Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 447pp. 9102200251. 56746:001.

See NUREG-1434,V01,DRF,FC abstract.

**NUREG-1434 V3 DRF FC:** STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6.Bases (Sections 3.4 - 3.10).Draft Report For Comment. * Division of Operational Events Assessment (Post 870411). January 1991. 496pp. 9102200254. 56748:225.

See NUREG-1434,V01,DRF,FC abstract.

**NUREG-1435 S01:** STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS.TMI Action Plan Requirements.Unresolved Safety Issues.Generic Safety Issues. * Program Management, Policy Development & Analysis Staff (Post 870411). December 1991. 107pp. 9201060172. 60207:075.

As part of ongoing U.S. Nuclear Regulatory Commission (NRC) efforts to ensure the quality and accountability of safety issue information, a program was established whereby an annual NUREG report would be published on the status of licensee implementation and NRC verification of safety issues in major NRC requirement areas. This information was compiled and reported in three NUREG volumes. Volume 1, published in March 1991, addressed the status of Three Mile Island (TMI) Action Plan Requirements. Volume 2, published in May 1991, addressed the status of unresolved safety issues (USIs). Volume 3, published in June 1991, addressed the implementation and verification status of generic safety issues (GSIs). This annual NUREG report combines these volumes into a single report and provides updated information as of September 30, 1991. The data contained in these NUREG reports are a product of the NRC's Safety Issues Management System database, which is maintained by the Project Management Staff in the Office of Nuclear Reactor Regulation and by NRC regional personnel. This report is to provide a comprehensive description of the implementation and verification status of TMI Action Plan Requirements, safety issues designated as USIs, and GSIs that have been resolved and involve implementation of an action or actions by licensees. This report makes the information available to other interested parties, including the public. An additional purpose of this NUREG report is to serve as a follow-on to NUREG-0933, "A Prioritization of Generic Safety Issues," which tracks safety issues up until requirements are approved for imposition at licensed plants or until the NRC issues a request for action by licensees.

**NUREG-1435 V01:** STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS.TMI Action Plan Requirements. * Program Management, Policy Development & Analysis Staff (Post 870411). March 1991. 672pp. 9104080296. 57295:083.

As part of ongoing U.S. Nuclear Regulatory Commission (NRC) efforts to ensure the quality and accountability of safety issue information, a program has been established whereby an annual NUREG series report will be published on the status of licensee implementation and NRC verification of safety issues in major NRC requirement areas. The data contained in this report are a product of the NRC's Safety Issues Management System database, which is maintained by the Project Management Staff in the Office of Nuclear Reactor Regulation and by personnel in the NRC regions. This report has been prepared in order to provide a comprehensive description of the implementation and verification status of all the TMI Action Plan requirements at licensed reactors, and to make this information available to other interested parties, including the public. A corollary purpose of this report is for it to serve as a follow-on to NUREG-0933, "A Prioritization of Generic Safety Issues," which tracks safety issues up until requirements are approved for imposition at licensed facilities.

**NUREG-1435 V02:** STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS.Unresolved Safety Issues. * Program Management, Policy Development & Analysis Staff (Post 870411). May 1991. 234pp. 9106120188. 58062:261.

As part of ongoing U.S. Nuclear Regulatory Commission (NRC) efforts to ensure the quality and accountability of safety issue information, a program has been established whereby an annual NUREG report will be published on the status of licensee implementation and NRC verification of safety issues in major NRC requirement areas. This report, the second volume of a three-volume series, addresses the status of unresolved safety issues at licensed plants. The data contained in this report are a product of the NRC's Safety Issues Management System database, which is maintained by the Project Management Staff in the Office of Nuclear Reactor Regulation and by personnel in the NRC regions. This report has been prepared in order to provide a comprehensive description of the implementation and verification status of all the TMI Action Plan requirements at licensed reactors, and to make this information available to other interested parties, including the public. A corollary purpose of this report is for it to serve as a follow-on to NUREG-0933, "A Prioritization of Generic Safety Issues," which tracks safety issues up until requirements are approved for imposition at licensed facilities.

**NUREG-1435 V03:** STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS.Generic Safety Issues. * Program Management, Policy Development & Analysis Staff (Post 870411). June 1991. 271pp. 9107080225. 58306:188.

As part of ongoing U.S. Nuclear Regulatory Commission (NRC) efforts to ensure the quality and accountability of safety issue information, a program has been established whereby an annual NUREG report will be published on the status of licensee implementation and NRC verification of safety issues in major NRC requirement areas. This report, the third volume of a three-volume series, addresses the status of generic safety issues at licensed plants. The data contained in this report are a product of the NRC's Safety Issues Management System database, which is maintained by the Project Management Staff in the Office of Nuclear Reactor Regulation and by personnel in the NRC regions. This report has been prepared in order to provide a comprehensive description of the implementation and verification status of all generic safety issues at licensed reactors, and to make this information available to other interested parties, including the public. A corollary purpose of this report is for it to serve as a follow-on to NUREG-0933, "A Prioritization of Generic Safety Issues," which tracks safety issues up until requirements are approved for imposition at licensed facilities.

**NUREG-1437 V1 DRF FC:** GENERIC ENVIRONMENTAL IMPACT STATEMENT FOR LICENSE RENEWAL OF NUCLEAR PLANTS.Main Report.Draft Report For Comment. * Division of Safety Issue Resolution (Post 880717). August 1991. 559pp. 9109300088. 59229:085.

This Generic Environmental Impact Statement (GEIS) examines the possible environmental impacts that could occur as a result of renewing licenses of individual nuclear power plants under the proposed 10 CFR Part 54. The GEIS, to the extent possible, establishes the bounds and significance of these potential impacts. The analyses in the GEIS encompass all operating light-water power reactors. For each type of environmental impact the GEIS attempts to establish generic findings covering as many plants as possible. This GEIS has three principal objectives: (1) to provide an understanding of the types and severity of environmental impacts that may occur as a result of license renewal of nuclear power plants under 10 CFR Part 54, (2) to identify and assess those impacts that are expected to be generic to license renewal, and (3) to support a rulemaking (10 CFR Part 51) to define the number and scope of issues that need to be addressed by the applicants in plant-by-plant license renewal proceedings. To accomplish these objectives, the GEIS makes maximum use of environmental and safety documentation from original licensing proceedings and information from state and federal regulatory agencies, the nuclear utility industry, the open literature, and professional contacts.

**NUREG-1437 V2 DRF FC:** GENERIC ENVIRONMENTAL IMPACT STATEMENT FOR LICENSE RENEWAL OF NUCLEAR PLANTS.Appendices.Draft Report For Comment. * Division of Safety Issue Resolution (Post 880717). August 1991. 643pp. 9109300090. 59230:284.

See NUREG-1437,V01,DRF,FC abstract.

**NUREG-1439:** STAFF TECHNICAL POSITION ON REGULATORY CONSIDERATIONS IN THE DESIGN AND CONSTRUCTION OF THE EXPLORATORY SHAFT FACILITY. GUPTA,D.; PESHEL,J.; BUNTING,J. Division of High-Level Waste Management (Post 870413). July 1991. 50pp. 9108130271. 58764:306.

The staff of the U.S. Nuclear Regulatory Commission has prepared this staff technical position for the purpose of compiling and further clarifying previous staff positions on regulatory considerations in the design and construction of the exploratory shaft facility (ESF). (The U.S. Department of Energy (DOE) now refers to the ESF as the "exploratory studies facility." DOE's change in terminology does not affect the positions taken in this guidance.) This document lists the key regulations in 10 CFR Part 60 that should be considered in the design and construction of the ESF and presents the staff position statements and corresponding discussions.

**NUREG-1440 DRFT FC:** REGULATORY ANALYSIS OF PROPOSED AMENDMENTS TO REGULATIONS CONCERNING THE ENVIRONMENTAL REVIEW FOR RENEWAL OF NUCLEAR POWER PLANT OPERATING LICENSES.Draft Report For Comment. * Division of Safety Issue Resolution (Post 880717). August 1991. 33pp. 9109300064. 59256:188.

This regulatory analysis provides the supporting information for a proposed rule that will amend the Nuclear Regulatory Commission's requirements for environmental review of applications for renewal of nuclear power plant operating licenses. After considering various options, the staff identified and analyzed two major alternatives. Alternative A is to not amend the regulations and to perform environmental reviews under the existing regulations. Alternative B is to assess, on a generic basis, the environmental impacts of renewing the operating license of individual nuclear power plants, and define the issues that will need to be further analyzed on a case-by-case basis. The findings of this assessment are to be codified in 10 CFR Part 51. The staff has selected Alternative B as the preferred alternative.

**NUREG-1441:** LESSONS LEARNED FROM THE POST-EMERGENCY TABLETOP EXERCISE IN BATON ROUGE,LOUISIANA,ON AUGUST 28 AND SEPTEMBER 18, 1990. WEINSTEIN,E. Incident Response Branch. BATES,G. Region 4 (Post 820201). PEYTON,L. Federal Emergency Management Agency. July 1991. 32pp. 9108190258. FEMA-REP-16. 58828:185.

On August 28 and September 18, 1990, Gulf States Utilities, the States of Louisiana and Mississippi, five local parishes, six Federal agencies, and the American Nuclear Insurers participated in a post-emergency TABLETOP exercise in Baton Rouge, Louisiana. The purpose of the exercise was to examine the post-emergency roles, responsibilities, and resources of utility, State, local, Federal and insurance organizations in response to a hypothetical accident at the River Bend Station in Louisiana resulting in a significant release of radiation to the environment. In pursuit of this goal, five major focus areas were addressed: (1) ingestion pathway response; (2) reentry, relocation and return; (3) decontamination of recovery; (4) indemnification of financial losses; and (5) deactivation of the emergency response. This report documents the lessons learned from that exercise.

**NUREG-1442:** POST-EMERGENCY RESPONSE RESOURCES GUIDE.Based On The Post-Emergency TABLETOP Exercise in Baton Rouge,Louisiana,On August 28 And September 18, 1990. WEINSTEIN,E. Incident Response Branch. BATES,G. Region 4 (Post 820201). * Federal Emergency Management Agency. July 1991. 38pp. 9108290223. FEMA-REP-17. 58910:215.

On August 28 and September 18, 1990, the States of Louisiana and Mississippi, Gulf States Utilities, five local parishes, six Federal agencies, and the American Nuclear Insurers participated in a post-emergency TABLETOP exercise in Baton Rouge, Louisiana. One of the products developed from that experience is this guide for understanding the responsibilities and obtaining resources for specific needs from the various participants, particularly those organizations within the Federal Government. This guide should assist State and local government organizations with identifying and obtaining those resources for the post-emergency response when theirs have been exhausted.

**NUREG-1443:** SAFETY EVALUATION REPORT RELATED TO THE FULL-TERM OPERATING LICENSE FOR SAN ONOFRE NUCLEAR GENERATING STATION,UNIT 1. Docket No. 50-206.(Southern California Edison Company And San Diego Gas And Electric Company) * Division of Reactor Projects - III,IV,V & Special Projects (870411-901215). July 1991. 43pp. 9108130279. 58746:306.

The safety evaluation report for the full-term operating license application filed by the Southern California Edison Company and the San Diego Gas and Electric Company has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in San Diego County, California. The staff has evaluated the issues related to the conversion of the provisional operating license to a full-term operating license and concludes that the facility can continue to be operated without endangering the health and safety of the public following the license conversion.

**NUREG-1445:** REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC SAFETY ISSUE-29. BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS. CHANG,T.Y. Division of Safety Issue Resolution (Post 880717). September 1991. 116pp. 9110100228. 59334:001.

Generic Safety Issue (GSI)-29 deals with staff concerns about public risk due to degradation or failure of safety-related bolting in nuclear power plants. The issue was initiated in November 1982. Value-impact studies of a mandatory program on safety-related bolting for operating plants were inconclusive; therefore, additional regulatory requirements for operating plants could not be justified in accordance with provisions of 10 CFR 50.109. In addition, based on (1) operating experience with bolting in both nuclear and conventional power plants, (2) the actions already taken through bulletins, generic letters, and information notices, and (3) the industry-proposed actions, the staff concluded that a sufficient technical basis exists for the resolution of GSI-29. The staff further concluded that leakage of bolted pressure joints is possible but catastrophic failure of a reactor coolant pressure boundary joint that will lead to significant accident sequences is highly unlikely. For future plants, it was concluded that a new Standard Review Plan section should be developed to codify existing bolting requirements and industry-developed initiatives.

**NUREG-1446:** STANDARDS FOR PROTECTION AGAINST RADIATION - 10 CFR PART 20. A Comparison Of The Existing And Revised Rules. COOL,D.A.; PETERSON,H.T. Division of Regulatory Applications (Post 870413). October 1991. 133pp. 9201060176. 60207:182.

On May 21, 1991, the Nuclear Regulatory Commission (NRC) issued a revision to its standards for protection against ionizing radiation, 10 CFR Part 20. Although the revised part (Sections 20.1001-20.2401) became effective on June 20, 1991, licensees may defer implementation of the revised rule until January 1, 1993. Licensees continue to be required to comply with the provisions of Sections 20.1-20.601 until the time they adopt the provisions of Sections 20.1001-20.2401. Therefore, between June 20, 1991, and January 1, 1993, both the provisions of Sections 20.1-20.601 and Sections 20.1001-20.2401 are in effect. This NUREG presents a comparative text of the provisions of the revised Part 20 (Sections 20.1001-20.2401) to the text of Sections 20.1-20.601 for use by the NRC staff and NRC licensees.

**NUREG-1450:** POTENTIAL CRITICALITY ACCIDENT AT THE GENERAL ELECTRIC NUCLEAR FUEL AND COMPONENT MANUFACTURING FACILITY, MAY 29, 1991. * Ofc of the Executive Director for Operations. August 1991. 230pp. 9108210183. 58857:167.

At the General Electric Nuclear Fuel and Component Manufacturing facility, located near Wilmington, North Carolina, on May 28 and 29, 1991, approximately 150 kilograms of uranium were inadvertently transferred from safe process tanks to an unsafe tank located at the waste treatment facility, thus creating the potential for a localized criticality safety problem. The excess uranium was ultimately safely recovered when the tank contents were centrifuged to remove the uranium- bearing material. Subsequently, the U.S. Nuclear Regulatory Commission dispatched an Incident Investigation Team to determine what happened, to identify probable causes, and to make appropriate findings and conclusions. This report describes the incident, the methodology used by the team in its investigation, and presents the team's findings and conclusions.

**NUREG-1455:** TRANSFORMER FAILURE AND COMMON-MODE LOSS OF INSTRUMENT POWER AT NINE MILE POINT UNIT 2 ON AUGUST 13, 1991. * Ofc of the Executive Director for Operations. October 1991. 237pp. 9111070103. 59551:291.

On August 13, 1991, at Nine Mile Point Unit 2 nuclear power plant, located near Scriba, New York, on Lake Ontario, the main transformer experienced an internal failure that resulted in degraded voltage which caused the simultaneous loss of five uninterruptible power supplies which in turn caused the loss of several nonsafety systems, including reactor control rod position indication, some reactor power and water indication, control room annunciators, the plant communications system, the plant process computer, and lighting at some locations. The reactor was subsequently brought to a safe shutdown. Following this event, the U.S. Nuclear Regulatory Commission dispatched an Incident Investigation Team to the site to determine what happened, to identify the probable causes, and to make appropriate findings and conclusions. This report describes the incident, the methodology used by the team in its investigation, and presents the teams findings and conclusions.

**NUREG/CP-0037:** PROCEEDINGS OF THE SEMINAR ON ASSESSMENT OF FRACTURE PREDICTION TECHNOLOGY: PIPING AND PRESSURE VESSELS. HISER,A.L.; MAYFIELD,M.E. Office of Nuclear Regulatory Research (Post 860720). February 1991. 334pp. 9103050499. 56880:187.

The 1990 Pressure Vessel and Piping Conference, sponsored by the American Society of Mechanical Engineers (ASME), was held in Nashville, Tennessee from June 18 to June 21, 1990. As part of that conference, representatives from the USNRC and AEA Technology in the United Kingdom jointly organized two panel sessions to discuss the current state of fracture prediction technologies for piping and pressure vessels. A total of nine presentations were given, contrasting analytical predictions with experimental results. This document provides summaries of each presentation and copies of the pertinent figures and other visual aids. This information has been compiled and published to permit reasonably prompt disseminations of the information presented. Based on the information presented during these two panel sessions, it appears that, while the current state of fracture prediction technology is reasonably well advanced, more work is needed to provide analysis methods capable of accurately predicting ductile crack extension.

**NUREG/CP-0114 V01:** PROCEEDINGS OF THE EIGHTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. April 1991. 678pp. 9105030130. 57622:001.

This three-volume report contains 100 papers out of the 128 that were presented at the Eighteenth Water Reactor Safety Information Meeting held at the Holiday Inn Crowne Plaza, Rockville, Maryland, during the week of October 22-24, 1990. 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 16 different papers presented by researchers from Denmark, Egypt, Germany, IAEA, Italy, Japan, Norway, Taiwan, UK and USSR. 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-0114 V02:** PROCEEDINGS OF THE EIGHTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. April 1991. 595pp. 9105150327. 57715:230.

See NUREG/CP-0114,V01 abstract.

**NUREG/CP-0114 V03:** PROCEEDINGS OF THE EIGHTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. April 1991. 582pp. 9105150310. 57710:300.

See NUREG/CP-0114,V01 abstract.

**NUREG/CP-0115:** PROCEEDINGS OF THE CSNI WORKSHOP ON PSA APPLICATIONS AND LIMITATIONS. MOLINA,T. Sandia National Laboratories. February 1991. 485pp. 9103040390. SAND90-2797. 56869:096.

This report contains the full papers submitted to the Committee on the Safety of Nuclear Installations Workshop on Probabilistic Safety Assessment (PSA) Applications and Limitations held in Santa Fe, New Mexico, USA, on September 4 through 6,

1990. The purpose of the Workshop was to provide an avenue for discussions in the following areas: (1) current PSA results, (2) current uses of PSA, (3) views on current limitations, (4) expert opinion, and (5) low probability numbers. The papers contained herein address these issues, along with several other related topics.

**NUREG/CP-0116 V01:** PROCEEDINGS OF THE 21ST DOE/NRC NUCLEAR AIR CLEANING CONFERENCE.Sessions 1 - 8.Held in San Diego, California, August 13-16, 1990. FIRST,M.W. Harvard School of Public Health, Boston, MA. February 1991. 464pp. 9103200077. CONF-900813. 57059:153.

This document contains the papers and the associated discussions of the 21st DOE/NRC Nuclear Air Cleaning Conference. Major topics are: (1) chemical processing systems, (2) reactor operations, (3) incineration and vitrification, (4) particulate filter developments, including filter testing and response to physical and temperature stress, (5) adsorption and testing of activated carbon and adsorber systems, (6) severe accident mitigation including modeling of emergency response systems, (7) nuclear waste management systems, (8) carbon-14 removal, (9) monitoring and measurement systems, (10) the development of standards and regulations and concerns with existing standards and regulations, and (11) nuclear air cleaning activities around the world.

**NUREG/CP-0116 V02:** PROCEEDINGS OF THE 21ST DOE/NRC NUCLEAR AIR CLEANING CONFERENCE SESSIONS 9 - 16.Held in San Diego, California, August 13-16, 1990. FIRST,M.W. Harvard School of Public Health, Boston, MA. February 1991. 484pp. 9103200094. CONF-900813. 57061:001.

See NUREG/CP-0116,V01 abstract.

**NUREG/CP-0118:** TRANSACTIONS OF THE NINETEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Office of Nuclear Regulatory Research (Post 860720). October 1991. 211pp. 9110280060. 59451:205.

This report contains summaries of papers on reactor safety research to be presented at the 19th Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel in Bethesda, Maryland, October 28-30, 1991. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, USNRC. Summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the Electric Power Research Institute, the nuclear industry, and from the governments and industry in Europe and Japan 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/CR-2000 V09N12:** LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of December 1990. * Oak Ridge National Laboratory. January 1991. 82pp. 9102140277. ORNL/NSIC-200. 56693:231.

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-0161, "Instructions for Preparation of Data Entry Sheets for Licensee Event Reports." For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73 - Licensee Event Report System) which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022, "Licensee Event Report System - Description of Systems and Guidelines for Reporting," provides supporting guidance and information on the revised LER rule. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, keyword, and component vendor indexes follow the summaries. Vendors are those identified by the utility when the LER form is initiated; the keywords for the component, system, and general keyword indexes are assigned by the computer using correlation tables from the Sequence Coding and Search System.

**NUREG/CR-2000 V10 N1:** LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of January 1991. * Oak Ridge National Laboratory. February 1991. 95pp. 9103200049. ORNL/NSIC-200. 57065:214.

See NUREG/CR-2000,V09,N12 abstract.

**NUREG/CR-2000 V10 N2:** LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of February 1991. * Oak Ridge National Laboratory. March 1991. 112pp. 9104220299. ORNL/NSIC-200. 57450:119.

See NUREG/CR-2000,V09,N12 abstract.

**NUREG/CR-2000 V10 N3:** LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of March 1991. * Oak Ridge National Laboratory. April 1991. 97pp. 9105170174. ORNL/NSIC-200. 57771:222.

See NUREG/CR-2000,V09,N12 abstract.

**NUREG/CR-2000 V10 N4:** LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of April 1991. * Oak Ridge National Laboratory. May 1991. 92pp. 9106120185. ORNL/NSIC-200. 58063:135.

See NUREG/CR-2000,V09,N12 abstract.

**NUREG/CR-2000 V10 N5:** LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of May 1991. * Oak Ridge National Laboratory. June 1991. 95pp. 9107220259. ORNL/NSIC-200. 58490:105.

See NUREG/CR-2000,V09,N12 abstract.

**NUREG/CR-2000 V10 N6:** LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of June 1991. * Oak Ridge National Laboratory. July 1991. 69pp. 9108130330. ORNL/NSIC-200. 58766:106.

See NUREG/CR-2000,V09,N12 abstract.

**NUREG/CR-2000 V10 N7:** LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of July 1991. * Oak Ridge National Laboratory. August 1991. 75pp. 9109050256. ORNL/NSIC-200. 58989:204.

See NUREG/CR-2000,V09,N12 abstract.

**NUREG/CR-2000 V10 N8:** LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of August 1991. * Oak Ridge National Laboratory. September 1991. 89pp. 9110100244. ORNL/NSIC-200. 59335:147.

See NUREG/CR-2000,V09,N12 abstract.

**NUREG/CR-2000 V10 N9:** LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of September 1991. * Oak Ridge National Laboratory. October 1991. 96pp. 9112310154. ORNL/NSIC-200. 60161:080.

See NUREG/CR-2000,V09,N12 abstract.

**NUREG/CR-2000 V10N10:** LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of October 1991. * Oak Ridge National Laboratory. November 1991. 141pp. 9112310155. ORNL/NSIC-200. 60160:299.

See NUREG/CR-2000,V09,N12 abstract.

**NUREG/CR-2000 V10N11:** LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of November 1991. * Oak Ridge National Laboratory. December 1991. 79pp. 9201140010. ORNL/NSIC-200. 60299:108.

See NUREG/CR-2000,V09,N12 abstract.

**NUREG/CR-2907 V09:** RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1988. TICHLER,J.; NORDEN,K.; CONGEMI,J. Brookhaven National Laboratory. July 1991. 331pp. 9106210010. BNL-NUREG-51581. 58743:001.

Releases of radioactive materials in airborne and liquid effluents from commercial light water reactors during 1988 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 1988 release data are summarized in tabular form. Data covering specific radionuclides are summarized.

**NUREG/CR-3145 V09:** GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION.Annual Report,October 1989 - September 1990. MEERT,J.; RUFF,L.; LAFORGE,R.; et al. Michigan, Univ. of, Ann Arbor, MI. January 1991. 42pp. 9103260132. 57152:343.

Earthquake activity in the Western Ohio-Indiana region has been monitored with a precision seismograph network consisting of nine stations located in west-central Ohio and four stations located in Indiana. Two local earthquakes have been recorded during this report period with magnitudes of 1.0m(L) and 2.2 m(L). Two near-regional events were recorded by the network (Fostoria, Ohio and Clinton, Illinois events) with magnitudes of 2.3 m(L) and 3.4 m(b)L(g). Three other regional events occurred with magnitudes ranging from 2.9 m(b) to 4.6 m(b). All the regional events occurred in, or near, regions with well-established histories of seismicity.

**NUREG/CR-3444 V08:** THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.Effects Of Composition On The Strength,Swelling, And Water-Immersion Properties of Cement-Solidified Ion-Exchange Resin Wastes. SOO,P.; MILIAN,L.W. Brookhaven National Laboratory. October 1991. 100pp. 9112310156. BNL-NUREG-51699. 60177:175.

A study has been completed on the degradation and failure mechanisms in simulated decontamination waste ion-exchange resins solidified in Portland cement. The mixed-bed resins were loaded with LOMI (low oxidation-state metal-ion) reagent and various cationic species to ascertain how they influenced the strength, swelling, set time, and water-immersion integrity of the resultant waste forms. It was found that the properties of the waste forms were very dependent on the composition, as expected, and that different mechanisms for degradation and failure were present for different compositional ranges.

**NUREG/CR-3469 V06:** OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA. KHAN,T.A.; VULIN,D.S.; LANE,S.G.; et al. Brookhaven National Laboratory. October 1991. 77pp. 9112310158. BNL-NUREG-51708. 60160:122.

One of the functions of the ALARA Center is to collect and disseminate information on dose reduction at nuclear power plants. This is the sixth report in the series of bibliographies of selected readings in radiation protection and ALARA that the Center publishes periodically. The abstracts in this bibliography were selected from proceedings of technical meetings, journals, research reports, searches of information data bases and reprints of published articles provided to us by the authors. The abstracts relate in one way or another to dose reduction at nuclear power plants, whether it is through good water chemistry, improvements in nuclear materials, better control of corrosion, robotics, and remote tooling or good operational health physics. The report contains 266 abstracts. Subject and author indices are provided. The subject index covers all previous volumes in this series. All information in the current volume is also available from the ALARA Center's on-line service, which is accessible by personal computer with the help of a modem. The preface of the report explains how the service may be accessed. The on-line service will be updated as new information is received.

**NUREG/CR-3916:** PRESSURIZED MELT EJECTION INTO WATER POOLS. TARBELL,W.W.; PILCH,M. Sandia National Laboratories. ROSS,J.W.; et al. Ktech Corp. March 1991. 110pp. 9103260215. SAND84-1531. 57161:230.

This report describes five experiments that were performed to study the influence of water pools on the behavior of pressure driven melts. Four of the tests used linear-scaled models of reactor cavities. The core simulant was a molten mixture of alumina and iron created by a metallothermitic reaction. In all experiments, the pressure-driven jet interacted energetically with the water pool. The apparatus containing the water pool was destroyed in all cases, showing clear evidence of a violent fuel-coolant interaction. Pressure records and high-speed framing camera data were attempted in each test. The recorded pressure waveforms appear to correlate with previous steam explosion experiments.

**NUREG/CR-3964 V02:** TECHNIQUES FOR DETERMINING PROBABILITIES OF EVENTS AND PROCESSES AFFECTING THE PERFORMANCE OF GEOLOGIC REPOSITORIES.Suggested Approaches. APOSTOLAKIS,G. California, Univ. of, Los Angeles, CA. BRAS,R. Rafael Bras Consulting Engineers. PRICE,L.; et al. Sandia National Laboratories. June 1991. 184pp. 9107010105. SAND86-0196. 58285:035.

The U.S. Environmental Protection Agency has established a standard for the performance of geologic repositories for the disposal of radioactive waste. This standard is probabilistic in nature, but the methods for determining probabilities of events and processes of interest in implementing such a standard are still being developed. Decision Theory, which involves Bayesian probability techniques, can serve as a framework for estimating the probability of occurrence of processes and events that are likely to disrupt a geologic repository. This report presents the mathematical basis for such a methodology and demonstrates an application of it in three areas: climate change, tectonic events, and human intrusion.

**NUREG/CR-4063:** AN INVESTIGATION OF CORE LIQUID LEVEL DEPRESSION IN SMALL BREAK LOSS-OF-COOLANT ACCIDENTS. SCHULTZ,R.R. EG&G Idaho, Inc. (subs. of EG&G, Inc.). MOTLEY,F.E.; STUMPF,H.; et al. Los Alamos National Laboratory. August 1991. 168pp. 9108290249. EGG-2636. 58912:040.

Core liquid level depression can result in partial core dryout and heatup early in a small break loss-of-coolant accident transient. Such behavior occurs when steam, trapped in the upper regions of the reactor primary system (between the loop seal and the core inventory), moves coolant out of the core region and uncovers the rod upper elevations. The net result is core liquid level depression. Core liquid level depression and subsequent core heatups are investigated using subscale data from the ROSA-IV Program's 1/48-scale Large Scale Test Facility (LSTF) and the 1/1705-scale Semiscale facility. Both facilities are Westinghouse-type, four-loop, pressurized water reactor simulators. The depression phenomena and factors which influence the minimum core level are described and illustrated using examples from the data. Analyses of the subject experiments, conducted using the TRAC-PF1/MOD1 (Version 12.7) thermal-hydraulic code, are also described and summarized. Finally, the response of a typical Westinghouse four-loop plant (RESAR-3S) was calculated to qualitatively study core liquid level depression in a full-scale system.

**NUREG/CR-4214 R1P2A1:** HEALTH EFFECTS MODELS FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE

ANALYSIS.Modifications Of Models Resulting From Recent Reports On Health Effects Of Ionizing Radiation.Low LET Radiation.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. August 1991. 88pp. 9110090265. LMF-132. 59329:308.

The Nuclear Regulatory Commission has sponsored several studies to identify and quantify the potential health effects of accidental releases of radionuclides from nuclear power plants. The most recent health effects models resulting from these efforts were published in two reports, NUREG/CR-4214, Rev. 1, Part I (1990) and Part II (1989). Several major health effects reports have been published recently that may impact the health effects models presented in these reports. This addendum to the Part II (1989) report, provides a review of the 1986 and 1988 reports by the United Nations Scientific Committee on the Effects of Atomic Radiation, the National Academy of Sciences/National Research Council BEIR V Committee report and Publication 60 of the International Commission on Radiological Protection as they relate to this report. The three main sections of this addendum discuss early occurring and continuing effects, late somatic effects, and genetic effects. The major changes to the NUREG/CR-4214 health effects models recommended in this addendum are for late somatic effects. These changes reflect recent changes in cancer risk factors that have come from longer followup and revised dosimetry in major studies like that on the Japanese A-bomb survivors. The results presented in this addendum should be used with the basic NUREG/CR-4214 reports listed above to obtain the most recent views on the potential health effects of radionuclides released accidentally from nuclear power plants.

**NUREG/CR-4219 V07 N1:** HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For October 1989 - March 1990. PENNELL,W.E. Oak Ridge National Laboratory. March 1991. 103pp. 9104220334. ORNL/TM-9593. 57447:285.

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. In the current reporting period, reorganization of the original HSST program into separate programs with emphasis on fracture mechanics technology (HSST) and materials irradiation effects (HSSI) has been completed. The revised HSST program is organized in 10 Tasks. These are (1) Program Management, (2) Fracture Methodology and Analysis, (3) Material Characterization Tasks, (4) Special Technical Assistance, (5) Crack Arrest Technology, (6) Cleavage Crack Initiation, (7) Cladding Evaluations, (8) Pressurized-Thermal-Shock Technology, (9) Analysis Methods Validation, (10) Fracture Evaluation Tests. The program tasks have been structured to place emphasis on the resolution fracture issues with near-term licensing significance.

**NUREG/CR-4219 V07 N2:** HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For April-September 1990. PENNELL,W.E. Oak Ridge National Laboratory. September 1991. 139pp. 9110090303. ORNL/TM-9593. 59327:001.

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. Reorganization of the original HSST Program into separate programs with emphasis on fracture-mechanics technology (HSST) and materials-irradiation effects (HSSI) was previously completed. The revised HSST Program is organized in 10 tasks: (1) program management, (2) fracture methodology and analysis, (3) material characterization and properties, (4) special technical assistance, (5) crack-arrest technology, (6) cleavage-crack initiation, (7) cladding evaluations, (8) pressurized-thermal-shock technology, (9) analysis methods validation, and (10) fracture evaluation tests. 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 related research programs both in the United States and abroad.

**NUREG/CR-4235:** SELECTION OF SILICEOUS AGGREGATE FOR CONCRETE. CLIFTON,J.R.; KNAB,L.I. National Institute of Standards & Technology (formerly National Bureau of Standa. January 1991. 22pp. 9102060171. NISTIR 4327. 56651:156.

Alkali-aggregate expansive reactions are one of the potentially serious degradation problems that could affect the structural stability of underground concrete structures for disposing of low-level radioactive waste (LLW). It appears that all aggregates react to some degree with alkalies in cement. In the majority of cases the reactions are beneficial (e.g., increasing the bond between aggregate and hydrated cement paste) or innocuous. In some cases, however, the reactions result in the formation of expansive products which can cause serious cracking of the concrete. This report deals with the selection of siliceous aggregates to avoid deleterious alkali-aggregate expansions. Current practices used to prevent expansive alkali-silica reactions and the standard test methods used to identify reactive aggregates are first discussed. Then the results of a study on using a new alkali-silica reactivity test to select siliceous aggregates for use in the concrete of LLW disposal structures are presented. It is recommended that siliceous aggregates, selected for constructing underground vaults for disposal of LLW, have an expansion of less than 0.10 percent using the new test.

**NUREG/CR-4269:** MODELS OF TRANSPORT PROCESSES IN CONCRETE. POMMERSHEIM,J.; CLIFTON,J.R. National Institute of Standards & Technology (formerly National Bureau of Standa. January 1991. 104pp. 9101300151. NISTIR 4405. 56535:024.

An approach being considered by the U.S. Nuclear Regulatory Commission for disposal of low-level radioactive waste is to place the waste forms in concrete vaults buried underground. The vaults would need a service life of 500 years. Approaches for predicting the service life of concrete of such vaults include the use of mathematical models. Mathematical models are presented in this report for the major degradation processes anticipated for the concrete vaults, which are corrosion of steel reinforcement, sulfate attack, acid attack, and leaching. The models mathematically represent rate controlling processes including diffusion, convection, and reaction and sorption of chemical species. These models can form the basis for predicting the life of concrete under in-service conditions.

**NUREG/CR-4295:** BOND STRENGTH OF CEMENTITIOUS BOREHOLE PLUGS IN WELDED TUFF. AKGUN,H.; DAEMEN,J.J.K. Arizona, Univ. of, Tucson, AZ. February 1991. 315pp. 9103200031. 57063:039.

This study includes a systematic investigation of the bond strength of cementitious borehole plugs in welded tuff. Analytical and numerical analysis of borehole plug-rock stress transfer mechanics is performed. The interface strength and deformation are studied as a function of Young's modulus ratio of plug and rock, plug length, and rock cylinder outside-to-inside radius ratio. The tensile stresses in and near an axially loaded plug are analyzed. The frictional interface strength of an axially loaded borehole plug, the effect of axial stress and lateral external stress, and thermal effects are also analyzed. Implications for plug design are discussed. The main conclusion is a strong recommendation to design friction plugs in shafts, drifts, tunnels, or boreholes with a minimum length to diameter ratio of four. Such a geometrical design will reduce tensile stresses in the plug and in the host rock to a level which should minimize the risk of long-term deterioration caused by excessive tensile stresses. Push-out tests have been used to determine the bond strength by applying an axial load to cement plugs emplaced in boreholes in welded tuff cylinders. A total of 130 push-out tests have been performed as a function of borehole size, plug length, temperature, and degree of saturation of the host tuff. The use of four different borehole radii enables evaluation of size effects. A well-defined exponential strength decrease with increasing plug diameter results. While these extrapolated strengths can be used for the design of large diameter plugs, e.g., in shafts or drifts, it would be desirable to confirm the extrapolations by tests on larger plugs.

**NUREG/CR-4302 V02:** AGING AND SERVICE WEAR OF CHECK VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.Aging Assessments And Monitoring Method Evaluations. HAYNES,H.D. Oak Ridge National Laboratory. April 1991. 73pp. 9104220317. ORNL-6193. 57448:278.

Check valves are used extensively in nuclear power plant safety systems and balance-of-plant systems. The failures of these valves have resulted in significant maintenance efforts and, on occasion, have resulted in water hammer, overpressurization of low-pressure systems, and damage to flow system components. These failures have largely been attributed to severe degradation of integral parts (e.g., hinge pins, hinge arms, discs, and disc nut pins) resulting from instability (flutter) of check valve discs under normal plant operating conditions. Present surveillance requirements for nuclear power plant check valves have been inadequate for timely detection and trending of such degradation because neither the flutter nor the resulting wear can be detected prior to failure. Consequently, the U.S. Nuclear Regulatory Commission has had a continuing strong interest in resolving check valve problems. In support of the Nuclear Plant Aging Research Program, Oak Ridge National Laboratory has carried out an evaluation of several developmental and/or commercially available check valve diagnostic monitoring methods, in particular, those based on measurements of acoustic emission, ultrasonics, and magnetic flux. In each case, the evaluations have been focused on the capability of each method to provide diagnostic information useful in determining check valve aging and service wear effects (degradation), check valve failures, and undesirable operating modes. A description of each monitoring method is provided in this report, including examples of test data acquired under controlled laboratory conditions. In some cases, field test data acquired in situ are also presented. The methods are compared, and suggested areas in need of further development are identified.

**NUREG/CR-4427:** AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE BYRON AND BRAIDWOOD NUCLEAR POWER PLANTS. MOFFITT,N.E.; GORE,B.F.; VO,T.V. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1991. 35pp. 9108130320. PNL-7492. 58766:069.

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. Byron and Braidwood were selected for the fourth study in this program. The product of this effort is a prioritized listing of AFW failures which have occurred at the plants 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 Byron/Braidwood plants.

**NUREG/CR-4444:** RADIATION SAFETY ISSUES RELATED TO RADIOLABELED ANTIBODIES. BARBER,D.E.; BAUM,J.W.; MEINHOLD,C.B. Brookhaven National Laboratory. March 1991. 208pp. 9104040290. BNL-NUREG-52275. 57261:140.

Techniques related to the use of radiolabeled antibodies in humans are reviewed and evaluated in this report. It is intended as an informational resource for the U.S. Nuclear Regulatory Commission (NRC) and NRC licensees. Descriptions of techniques and health and safety issues are provided. Principal methods for labeling antibodies are summarized to help identify related radiation safety problems in the preparation of dosages for administration to patients. The descriptions are derived from an extensive literature review and consultations with experts in the field. A glossary of terms and acronyms is also included. An assessment was made of the extent of the involvement of organizations (other than the NRC) with safety issues related to radiolabeled antibodies in order to identify regulatory issues which require attention. Federal regulations and guides were also reviewed for their relevance. A few (but significant) differences between the use of common radiopharmaceuticals and radiolabeled antibodies were observed. The clearance rate of whole, radiolabeled immunoglobulin is somewhat slower than common radiopharmaceuticals, and new methods of administration are being used. New nuclides are being used or considered (e.g., Re-186 and At-211) for labeling antibodies. Some of these nuclides present new dosimetry, instrument calibration, and patient management problems. Subjects related to radiation safety that require additional research are identified.

**NUREG/CR-4469 V11:** NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semi-Annual Report, April-September 1989. DOCTOR,S.R.; GOOD,M.S.; GREEN,E.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1991. 72pp. 9110090288. PNL-5711. 59331:141.

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 April 1989 through September 1989.

**NUREG/CR-4513:** ESTIMATION OF FRACTURE TOUGHNESS OF CAST STAINLESS STEELS DURING THERMAL AGING IN LWR SYSTEMS. CHOPRA,O.K. Argonne National Laboratory. June 1991. 74pp. 9107010083. ANL-90/42. 58249:001.

A procedure and correlations are presented for predicting the change in fracture toughness of cast stainless steel components due to thermal aging during service in light water reactors (LWRs) at 280-330 C (535-625 F). The fracture toughness J-R curve and Charpy-impact energy of aged cast stainless steels are estimated from known material information. Fracture toughness of a specific cast stainless steel is estimated from the extent and kinetics of thermal embrittlement. The extent of thermal embrittlement is characterized by the room-temperature "normalized" Charpy-impact energy. A correlation for the extent of embrittlement at "saturation," i.e., the minimum impact energy that would be achieved for the material after long-term aging, is given in terms of a material parameter, $\Phi$, which is determined from the chemical composition. The fracture toughness J-R curve for the material is then obtained from correlations between room-temperature Charpy- impact energy and fracture toughness parameters. Fracture toughness as a function of time and temperature of reactor service is estimated from the kinetics of thermal embrittlement, which is determined from chemical composition. A common "lower-bound" J-R curve for cast stainless steels with unknown chemical composition is also defined for a given material specification, ferrite content, and temperature.

**NUREG/CR-4551 V2R1P2**: EVALUATION OF SEVERE ACCIDENT RISKS: QUANTIFICATION OF MAJOR INPUT PARAMETERS.Experts' Determination Of Containment Loads And Molten Core Containment Interaction Issues. HARPER,F.T.; PAYNE,A.C.; BREEDING,R.J.; et al. Sandia National Laboratories. April 1991. 469pp. 9105150319. SAND86-1309. 57714:121.

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, the Severe Accident Risk Reduction Program (SARRP) has completed a revised calculation of the risk to the general public from severe accidents at five nuclear power plants: Surry, Sequoyah, Zion, Peach Bottom, and Grand Gulf. 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 initiated by events, both internal to the power station and external to the power station was assessed. Much of the important input to the logic models was generated by expert panels. This document presents the distributions and the rationale supporting the distributions for the questions posed to the Containment Loads and Molten Core Interaction Expert Panels.

**NUREG/CR-4599 V01 N1**: SHORT CRACKS IN PIPING AND PIPING WELDS.Semiannual Report, March-September 1990. WILKOWSKI,G.M.; AHMAD,J.; BRUST,F.; et al. Battelle Memorial Institute. May 1991. 129pp. 9105300205. BMI-2173. 57867:354.

This is the first semiannual report of the U.S. Nuclear Regulatory Commission's Short Cracks in Piping and Piping Welds research program. The program began in March 1990 and will extend for 4 years. The intent 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. Only quasi-static loading rates are evaluated since the NRC's International Piping Integrity Research Group Program is evaluating the effects of seismic loading rates on cracked piping systems. Additional efforts involve investigating phenomena discovered during the course of conducting the Degraded Piping program. These include the evaluation of the occurrence of unstable crack jumps in ferritic steels at LWR temperatures, and the occurrence of anisotropic fracture properties causing, helical crack growth. Both of these phenomena may affect the safety margins implicit in leak before break (LBB) analyses. Other investigations deal with the fracture behavior of bi-metallic welds, and improvements in crack opening area analyses used in LBB. Since much of the work in this program was just beginning during this first reporting period and progress is limited, a complete statement of work for the whole program is provided in this report.

**NUREG/CR-4659 V04**: SEISMIC FRAGILITY OF NUCLEAR POWER PLANT COMPONENTS (PHASE II).A Fragility Handbook On Eighteen Components. BANDYOPADHYAY; HOFMAYER,C.H.; KASSIR,M.K.; et al. Brookhaven National Laboratory. June 1991. 36pp. 9108130313. BNL-NUREG-52007. 58766:031.

Fragility estimates of seven equipment classes were published in earlier reports. This report presents fragility analysis results for eleven additional equipment categories. The fragility levels are expressed in probabilistic terms. For users' convenience, the concluding report includes a summary of fragility results of all eighteen equipment classes. A set of conversion factors based on judgment is recommended for use of the information for early vintage equipment. The knowledge gained in conducting the Component Fragility Program and similar other programs is expected to provide a new direction for seismic verification and qualification of equipment.

**NUREG/CR-4666**: CLOSEOUT OF IE BULLETIN 84-02: FAILURES OF GENERAL ELECTRIC TYPE HFA RELAYS IN USE IN CLASS 1E SAFETY SYSTEMS. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. January 1991. 54pp. 9104300323. PARAMETER IE163. 57557:129.

Documentation is provided in this report to close IE Bulletin 84-02 regarding the failure of General Electric Type HFA relays in Class 1E safety systems. The relay failures were due to aging of coil wire insulation and nylon or Lexan spools under certain environmental conditions. The bulletin was issued to nuclear power reactor licensees and holders of construction permits to provide assurance that the manufacturer's recommendations for corrective actions would be implemented. The bulletin required four specific actions, plus a review of the general concerns of the bulletin even though some facilities had different relays from those of bulletin concern. Evaluation of utility responses, NRC/Region inspection reports, and regional telephone calls has resulted in bulletin closeout of 116 (98%) of the 118 facilities to which the bulletin was issued for action. Facilities which were shut down or had construction halted indefinitely or permanently when the report was issued are not included in this review. A follow-up item is proposed in Appendix C for the two facilities with open status. Background information is supplied in the introduction and Appendix A.

**NUREG/CR-4667 V09**: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April-September 1989. KASSNER,T.F.; PARK,J.Y.; RUTHER,W.E.; et al. Argonne National Laboratory. March 1991. 30pp. 9104220214. ANL-90/48. 57450:263.

This report summarizes work performed by Argonne National Laboratory on environmentally assisted cracking in light water reactors during the six months from April 1989 to September 1989. Topics that were investigated include (1) stress corrosion cracking(SCC) of A533-Gr B steel in simulated boiling-water-reactor environments, (2) SCC of Types 347 and CF-3 cast duplex stainless steel (SS), and (3) effects of heat-to-heat variation on SCC of Type 304 SS. Crack-growth-rate (CGR) tests were performed on conventional (unplated) and nickel-or gold-plated A533-Gr B specimens to provide insight into whether the surface layer on the low-alloy steel, either oxide corrosion products or a noble metal,influences the overall SCC process. CGR tests were also conducted on specimens of Type 347 SS with different heat-treatments, and a specimen of CF-3 cast SS with a ferrite content of 15.6%. CGR data on these specimens were compared with reference fatigue crack growth curves in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix A. The influence of approximately 1.0 ppm of CuCl indeoxygentated water on the SCC susceptibility of Types 316NG and 347 SS and A533-Gr B and A106-Gr B ferritic steels was determined in constant-extension-rate tensile (CERT) tests at 200 C. The CERT results indicated that the alternative SSs were considerably more resistant to SCC than is sensitized Type 304 SS. The low-alloy ferritic steels exhibited only ductile fracture in this environment.

**NUREG/CR-4667 V10:** ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1989 - March 1990. RUTHER,W.E.; SHACK,W.J.; CHUNG,H.M.; et al. Argonne National Laboratory. March 1991. 30pp. 9104220294. ANL-91/5. 57450:231.

This report summarizes work performed by Argonne National Laboratory on environmentally assisted cracking in light water reactors during the six months from October 1989 to March 1990. Low-cycle fatigue tests were performed on Type 316NG SS to better understand the effects of cyclic strain range, frequency, and temperature on fatigue life in air and in simulated BWR water, and to assess the degree of conservatism in the ASME Code Section III fatigue design curves. Fracture mechanics crack-growth-rate tests were carried out on a composite specimen of A533-GrB/Inconel-182/Inconel-600, plated with nickel, to establish whether a transgranular crack will initiate in the ferritic steel from an intergranular crack in the Inconel-182 weld metal at low stress intensity associated with crack growth in the Inconel-182 weld metal. Irradiated stainless steels from absorber-rod tubes, control-rod cladding, and flux thimbles of several BWRS and PWRS were obtained to investigate the nature and extent of radiation-induced segregation in the steels and correlate it with susceptibility to intergranular failure in the materials. Specimens have been prepared for Auger electron spectroscopy analyses of segregation of alloying elements on intergranular fracture surfaces.

**NUREG/CR-4667 V11:** ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April-September 1990. CHUNG,H.M.; KASSNER,T.F.; SHACK,W.J.; et al. Argonne National Laboratory. May 1991. 37pp. 9105300212. ANL-91/9. 57867:316.

This report summarizes work performed by Argonne National Laboratory on environmentally assisted cracking in light water reactors during the 6 months from April 1990 to September 1990. Crack growth rate (CGR) tests were performed on a composite A533 Gr-B/Inconel-182 specimen in which a stress corrosion crack in the Inconel-182 weld metal penetrated and grew into the A533-Gr B steel. CGR tests were also conducted on conventional (nonplated) and Ni- or Au-plated A533-Gr B specimens. CGR data on the A533-Gr B specimens were compared with the fatigue crack reference curves in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix A. High- and commercial-purity, HP and CP, respectively, specimens of Type 304 SS from BWR absorber rod tubes, irradiated during service to fluence levels of 6x10(20) 2x10(21) n cm(-2) (E>1 MeV) in two reactors were examined by Auger electron spectroscopy to characterize irradiation-induced grain boundary segregation and depletion of alloying and impurity elements, which have been associated with irradiation-assisted stress corrosion cracking of the steel. Intergranular fracture surfaces in high fluence CP material were characterized by relatively high levels of Si, P, and Ni segregation. Segregation of the impurity elements and intergranular failure in the HP material were negligible for a similar fluence level. However, grain boundary depletion of Cr was more significant in HP material than in CP material, which indicates that irradiation-induced segregation of impurity elements and depletion of alloying elements are interdependent.

**NUREG/CR-4667 V12:** ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1990 - March 1991. SHACK,W.J.; HICKS,P.D.; RUTHER,W.E.; et al. Argonne National Laboratory. August 1991. 67pp. 9110090270. ANL-91/24. 59330:036.

This report summarizes work performed by Argonne National Laboratory on environmentally assisted cracking in light water reactors during the six months from October 1990 to March 1991. Fatigue life of A533-Gr B pressure vessel steel was studied in high-purity (HP) deoxygenated water, in simulated PWR water, and in air. Fatigue data are compared with the design curve in Section III Appendix A of the ASME Boiler and Pressure Vessel Code. Equations in Section XI of the ASME Boiler and Pressure Vessel Code that relate crack growth rates (CGRs) of ferritic steels to loading parameters have been modified to incorporate CGR data that we recently acquired at high load ratios. The effect of water flow rate on the SCC behavior of Type 316NG stainless steel (SS) was investigated in fracture-mechanics CGR tests in HP oxygenated water at 289 degrees C. Corrosion fatigue curves for austenitic SS in Section XI of the ASME Boiler and Pressure Vessel Code have been modified to be more consistent with SCC data in simulated LWR environments at high load ratios. High- and commercial-purity (CP) specimens of Type 304 SS from BWR absorber-rod tubes, irradiated during service in two reactors to fluence levels of 1.4-2 x 10(21) n cm(-2) (E>1 MeV), were examined by Auger electron spectroscopy to characterize irradiation-induced grain boundary segregation and depletion of alloying and impurity elements, which have been associated with irradiation-assisted SCC of the steel. Slow-strain-rate tensile tests have been conducted in air and in simulated BWR water on specimens obtained from the irradiated CP Type 304 SS absorber-rod tubes.

**NUREG/CR-4670:** RADIONUCLIDE DISTRIBUTIONS AND MIGRATION MECHANISMS AT SHALLOW LAND BURIAL SITES.Final Report Of PNL Research Investigations On The Distribution, Migration, And Containment Of Radionuclides At Maxey Flats, Kentucky. KIRBY,L.J.; TOSTE,A.P.; THOMAS,C.W.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1991. 95pp. 9103120103. PNL-7582. 56948:269.

During the past several years, Pacific Northwest Laboratory (PNL) has conducted research at the Maxey Flats Disposal Site (MFDS) for the U.S. Nuclear Regulatory Commission (NRC). This work has identified the spectrum of radionuclides present in the waste trenches, determined the processes that were occurring relative to degradation of radioactive material within the burial trenches, determined the chemical and physical characteristics of the trench leachates and the chemical forms of the leached radionuclides, determined the mobility of these radionuclides, investigated the subsurface and surface transport processes, determined the biological uptake by the native vegetation, developed strategies for environmental monitoring, and investigated other factors that influence the long-term fate of the radionuclide inventory at the disposal site. This report is a final summary of the research conducted by PNL and presents the results and discussions relative to the above investigative areas.

**NUREG/CR-4674 V13:** PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1990 A STATUS REPORT.Main Report And Appendix A. MINARICK,J.W. Science Applications International Corp. (formerly Science Applications, Inc.). CLETCHER,J.W. Professional Analysis, Inc. COPINGER,D.A.; et al. Oak Ridge National Laboratory. August 1991. 180pp. 9109300062. ORNL/NOAC-232. 59227:001.

Twenty-eight operational events with conditional probabilities of core damage of 1.0 x 10(-6) or higher occurring at commercial light-water reactors during 1990 are considered to be precursors to potential severe core damage. These are described along with associated significance estimates, categorization, and subsequent analyses. This study is a continuation of earlier work, which evaluated the 1969-1981 and 1984-1989 events. The report discusses (1) the general rationale for this study, (2) the selection and documentation of events as precursors, (3) the estimation and use of conditional probabilities of subsequent severe core damage to rank precursor events, and (4) the plant models used in the analysis process.

**NUREG/CR-4674 V14:** PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1990 A STATUS REPORT.Appendixes B And C. MINARICK,J.W. Science Applications International Corp. (formerly Science Applications, Inc.). CLETCHER,J.W. Professional Analysis, Inc. COPINGER,D.A.; et al. Oak Ridge National Laboratory. August 1991. 493pp. 9109300085. ORNL/NOAC-232. 59227:312.

See NUREG/CR-4674,V13 abstract.

**NUREG/CR-4690 V01 R1:** GENERIC COMMUNICATIONS INDEX.Listings Of Communications 1971 - 1989. HAGEMEYER,D.; TOWLE,H. Science Applications International Corp. (formerly Science Applications, Inc.). May 1991. 446pp. 9108190276. SAIC-90/1393. 58826:062.

As part of its program to feed back information on operating experience to industry, the U.S. Nuclear Regulatory Commission (NRC) issues generic communications called bulletins (about 5/yr), circulars (now discontinued), generic letters (about 20/yr), and information notices (about 100/yr). The report presents an updated Generic Communications Index (GCI; previously published in NUREG/CR-4690, Vol. 1, December 1987) of all such communications from 1971, when such documentation started, to 1989. The GCI consists of records, one for each communication, containing fields for identification number, title, NRC technical contact, general system or topic, specific component or topic, cause or defect, potential effect, remarks, and vendors involved. To facilitate information retrieval, the report also contains topical listings of generic communications numbers.

**NUREG/CR-4735 V07:** EVALUATION AND COMPILATION OF DOE WASTE PACKAGE TEST DATA. Biannual Report: February-July 1989. FRAKER,A.C.; ESCALANTE,E. National Institute of Standards & Technology (formerly National Bureau of Standa. INTERRANTE,C. Division of High-Level Waste Management (Post 870413). December 1991. 136pp. 9201080106. 60232:001.

This report summarizes evaluations by the National Institute of Standards and Technology of Department of Energy activities on waste packages designed for containment of radioactive high-level nuclear waste for the six-month period, February through July, 1989. 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. Outlines for planned interpretative reports on the topics of aqueous corrosion of copper, mechanisms of stress corrosion cracking and internal failure modes of Zircaloy cladding are included. For the publications reviewed during this reporting period, short discussions are given to supplement the completed reviews and evaluations. Included in this report is an overall review of a 1984 report on glass leaching mechanisms, as well as reviews for each of the seven chapters of this report.

**NUREG/CR-4744 V04 N1:** LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,October 1988 - March 1989. CHOPRA,O.K.; CHUNG,H.M. Argonne National Laboratory. May 1991. 42pp. 9105300192. ANL-90/44. 57868:245.

This progress report summarizes work performed by Argonne National Laboratory on long-term embrittlement of cast duplex stainless steels in LWR systems during the 6 months from October 1988 to March 1989. Charpy-impact data are presented for several heats of cast stainless steel aged at temperatures between 320 and 450 C for times up to 30,000 h. Thermal aging decreases impact energy and shifts transition curves to higher temperatures. A saturation effect is observed for room-temperature impact energy and upper-shelf energy. Charpy data are analyzed to obtain the activation energy of the kinetics of embrittlement. The results suggest that the activation energy of embrittlement is not constant in the temperature range of 290-400 C, but increases as temperature decreases. A correlation is presented for estimating the extent of embrittlement of cast stainless steels from known material parameters. The degradation in mechanical properties can be reversed by annealing the embrittled material for 1 h at 550 C and then water quenching.

**NUREG/CR-4744 V04 N2:** LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,April-September 1989. CHOPRA,O.K.; SATHER,A.; BUSH,L.Y. Argonne National Laboratory. June 1991. 341pp. 9107010128. ANL-90/49. 58251:095.

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 6 months from April to September 1989. Tensile and fracture toughness data are presented for several heats of cast stainless steel that were aged up to 30,000 h at temperatures of 290-450 C. The results indicate that thermal aging increases the tensile stress and decreases the fracture toughness of the materials. In general, CF-3 steels are the least sensitive to thermal aging embrittlement and CF-8M steels are the most sensitive. The increase in flow stress of fully-aged cast stainless steels is ~10% for CF-3 steels and ~20% for CF-8 and CF-8M steels. The fracture toughness J(IC) and average tearing modulus for heats that are sensitive to thermal aging (e.g., CF-8M steels) are as low as ~90 kJ/m(2) and ~60, respectively. Correlations are presented for estimating the increase in flow stress of the steels from data for the kinetics of thermal embrittlement.

**NUREG/CR-4744 V05 N1:** LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,October 1989 - March 1990. CHOPRA,O.K.; BUSH,L.Y. Argonne National Laboratory. July 1991. 46pp. 9108130298. ANL-91/7. 58765:260.

This progress report summarizes work performed by ANL on long-term thermal embrittlement of cast duplex stainless steels in LWR systems during the six months from October 1989 to March 1990. The results from Charpy-impact tests and microhardness measurements of the ferrite phase for several heats of cast stainless steel aged up to 30,000 h at 290-400 degrees C are analyzed to establish the kinetics of thermal embrittlement. Correlations are presented for predicting the extent and kinetics of thermal embrittlement of cast stainless steels from material information that can be determined from the certified material test record. The extent of embrittlement is characterized by the room-temperature "normalized" Charpy-impact energy. Based on the information available, two methods are presented for estimating the extent of embrittlement at "saturation," i.e., the minimum impact energy that would be achieved for the material after long-term aging. The first method utilizes only the chemical composition of the steel. The second method is used when metallographic information on the ferrite morphology, i.e., ferrite content and mean ferrite spacing of the steel, is also available. The change in Charpy-impact energy as a function of time and temperature of reactor service is then estimated from the extent of embrittlement at saturation and from the correlations describing the kinetics of embrittlement, which is expressed in terms of the chemical composition and aging behavior of the steel at 400 degrees C.

**NUREG/CR-4744 V05 N2:** LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,April-September 1990. CHOPRA,O.K. Argonne National Laboratory. July 1991. 108pp. 9108130289. ANL-91/10. 58765:152.

This progress report summarizes work performed by Argonne National Laboratory on long-term embrittlement of cast duplex stainless steels in LWR systems during the six months from April-September 1990. A procedure and correlations are presented for predicting fracture toughness J-R curves and impact strength of aged cast stainless steels from known material information. Fracture toughness of a specific cast stainless steel is estimated from the extent and kinetics of embrittlement. The extent of embrittlement is characterized by the room-temperature Charpy-impact energy. A correlation for the extent of embrittlement at saturation is given in terms of a material parameter, Φ, which is determined from the ferrite morphology and/or chemical composition. Charpy-impact energy as a function of time and temperature of reactor service is estimated from the kinetics of embrittlement, which are determined from chemical composition. The fracture toughness J-R curve for the material is then obtained from correlations between room-temperature Charpy-impact energy and fracture toughness parameters. A "lower-bound" J-R curve for cast stainless steels with unknown chemical composition is also defined for a given material specification and temperature. Mechanical-property degradation suffered by cast stainless steel components from the decommissioned Shippingport reactor has been characterized. The results are used to validate the correlations and benchmark the laboratory studies. Charpy-impact, tensile, and fracture toughness data for materials from the hot-leg shutoff valve and cold-leg check valves and pump volute are presented. The results indicate a modest degree of embrittlement.

**NUREG/CR-4757:** LINE-LOSS DETERMINATION FOR AIR SAMPLER SYSTEMS. GLISSMEYER,J.A.; SEHMEL,G.A. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1991. 105pp. 9102280238. PNL-7597. 56836:278.

Iodine deposition can potentially bias the results of radioiodine air sampling systems. To develop guidance and acceptance criteria for determinations of line-loss correction factors, the data on laboratory sampler simulations, field tests on samplers and experimentally measured iodine deposition rates were reviewed. Sampling system design features and operating conditions at several power reactors are discussed. Measurements of iodine deposition rates on various air sampler construction materials were reviewed, and predicted air sampler performance based on the data was presented. Three examples of field tests of air sampler performance for radioiodine were examined. A model of iodine deposition and resuspension was extensively reviewed, and suggestions were made for incorporating variable resuspension rates. Three principal methods for determining radioiodine line-loss factors were defined and compared: in-place field tests, laboratory mock-up with modelled extrapolations to various release rate modes, and modelling based on laboratory data on similar materials. Guidelines for applying these methods were given. Research was recommended to determine whether the three methods were comparable so the less-expensive method could be substituted for the preferred field tests.

**NUREG/CR-4816 R01:** PR-EDB: POWER REACTOR EMBRITTLEMENT DATA BASE,VERSION 1. Program Description. STALLMANN,F.W.; KAM,F.B.K.; TAYLOR,B.J. Oak Ridge National Laboratory. July 1991. 37pp. 9108190281. ORNL/TM-10328. 58827:148.

Data concerning radiation embrittlement of pressure vessel steels in commercial power reactors have been collected from available surveillance reports. The purpose of this NRC-sponsored program is to provide the technical bases for voluntary consensus standards, regulatory guides, standard review plans, and codes. The data can also be used for the exploration and verification of embrittlement prediction models. The data files are given in dBASE III PLUS format and can be accessed with any personal computer using the DOS operating system. Menu-driven software is provided for easy access to the data including curve fitting and plotting facilities. This software has drastically reduced the time and effort for data processing and evaluation compared to previous data bases. The current version of the Power Reactor Embrittlement Data Base (PR-EDB) lists the test results of 117 base materials (plates and forgings), 85 welds, and 86 heat-affected-zone materials that were irradiated in 241 capsules of 82 reactors. Many capsules also contained correlation materials (standard reference materials, SRMs) from the ASTM plate and two HSST plates (01 and 02). Material from the Humboldt Bay reactor was used as an SRM for some General Electric reactors. The Electric Power Research Institute (EPRI), reactor vendors, and utilities have provided back-up quality assurance checks of the PR-EDB.

**NUREG/CR-4867:** RELAY TEST PROGRAM.Series 1 Vibration Tests. BANDYOPADHYAY; HOFMAYER,C.H.; SHTEYNGART,S. Brookhaven National Laboratory. January 1991. 164pp. 9102250185. BNL-NUREG-52277. 56789:051.

Brookhaven National Laboratory has conducted a test program on relays to determine the influence of parameters related to design, electrical conditions and vibratory motion on their respective seismic capacity levels. Single frequency excitation was used for most of the test runs; multifrequency random motion was also used for some number of test runs. The test data have been evaluated and the results are presented in this report.

**NUREG/CR-4893:** TECHNICAL FINDINGS REPORT FOR GENERIC ISSUE 135.Steam Generator And Steam Line Overfill Issues. * SCIENTECH, Inc. May 1991. 112pp. 9106040379. SCIE-42-89. 57903:262.

A detailed review of the tasks and available literature pertaining to Generic Issue 135 (GI 135), Steam Generator and Steam Line Overfill Issues, has been conducted and is documented in this technical findings report. The purpose of the review was to evaluate the current status of the issues and to determine whether sufficient information exists for resolution, or whether additional work is required. Based on the review, it is concluded that all issues are either resolved or are being pursether additional work is required. Based on the review, it is concluded that all issues are either resolved or are being pursued as part of other activities. In addition, a data search and evaluation were conducted on the frequency and effects of steam generator overfill events. Potential mitigating actions were considered. It was concluded that the public health and safety risks associated with these events are relatively minor and do not justify additional mitigating actions or regulations.

**NUREG/CR-4911:** INCENTIVE REGULATION OF NUCLEAR POWER PLANTS BY STATE REGULATORS. MARTIN,R.L.; BAKER,K.; OLSON,J. Battelle Human Affairs Research Centers. February 1991. 95pp. 9103040385. PNL-7596. 56869:001.

The Nuclear Regulatory Commission (NRC) monitors incentive programs established by state regulators in order to obtain current information and to consider the potential safety effects of the incentive programs as applied to nuclear units. The current report is an update of NUREG/CR-5509, "Incentive Regulation of Nuclear Power Plants by State Public Utility Commissions," published in December 1989. The information in this report was obtained from interviews conducted with each state regulator and each utility with a minimum entitlement of 10%. The agreements, orders, and settlements from which each incentive program was implemented were reviewed as required. The interview and supporting documentation form the basis for the individual state reports describing the structure and financial impact of each incentive program. The programs currently in effect represent the adoption of an existing nuclear performance incentive program proposal and one new program. In addition, since 1989 a number of nuclear units have been included in one existing program; while one program was discontinued and another one concluded.

**NUREG/CR-4918 V05:** CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Progress Report On Field Experiments At A Humid Region Site,Beltsville,Maryland. SCHULZ,R.K. California, Univ. of, Berkeley, CA. RIDKY,R.W. Maryland, Univ. of, College Park, MD. O'DONNELL,E. Waste Management Branch (Post 910830). October 1991. 23pp. 9111110278. 59575:330.

The project objective is to assess means for controlling water infiltration through waste disposal unit covers in humid regions. Experimental work is being performed in large scale lysimeters (70'x45'10') at Beltsville, MD and results of the assessment are applicable to disposal of LLW, uranium mill tailings, hazardous waste, and sanitary landfills. Three concepts are under investigation: (1) resistive layer barrier, (2) conductive layer barrier, and bioengineering water management. The resistive layer barrier consists of compacted earth (clay). The conductive layer barrier is a special case of the capillary barrier and it requires a flow layer (e.g., fine sandy loam) over a capillary break. As long as unsaturated conditions are maintained water is conducted by the flow layer to below the waste. This barrier is most efficient at low flow rates and is thus best placed below a resistive layer barrier. Such a combination of the resistive layer over the conductive layer barrier promises to be highly effective provided there is no appreciable subsidence. Bioengineering water management is a surface cover that is designed to accommodate subsidence. It consists of impermeable panels which enhance run-off and limit infiltration. Vegetation is planted in narrow openings between panels to transpire water from below the panels. This system has successfully dewatered two lysimeters thus demonstrating that this procedure could be used for remedial action ("drying out") existing water-logged disposal sites at low cost.

**NUREG/CR-5128:** EVALUATION AND REFINEMENT OF LEAK-RATE ESTIMATION MODELS. PAUL,D.D.; AHMAD,J.; SCOTT,P.M.; et al. Battelle Memorial Institute. April 1991. 98pp. 9104290259. BMI-2164. 57530:173.

Leak-rate estimation models are important elements in developing a leak-before-break methodology in piping integrity and safety analyses. Existing thermal hydraulic and crack-opening-area models used in current leak-rate estimations have been incorporated into a single computer code for leak-rate estimation. The code is called SQUIRT, which stands for Seepage Quantification of Upsets In Reactor Tubes. The SQUIRT program has been validated by comparing its thermal hydraulic predictions with the limited experimental data that have been published on two-phase flow through slits and cracks, and by comparing its crack-opening-area predictions with data from the Degraded Piping Program. In addition, leak-rate experiments were conducted to obtain validation data for a circumferential fatigue crack in a carbon steel pipe girth weld.

**NUREG/CR-5139:** DOSE-REDUCTION TECHNIQUES FOR HIGH-DOSE WORKER GROUPS IN NUCLEAR POWER PLANTS. KHAN,T.A.; BAUM,J.W.; DIONNE,B.J. Brookhaven National Laboratory. March 1991. 82pp. 9103260083. BNL-NUREG-52278. 57152:011.

This report summarizes the main findings of a study of the extent of radiation dose received by special work groups in the nuclear power industry. Work groups which chronically get large doses were investigated, using information provided by the industry. The tasks that give high doses to these work groups were examined and techniques described that were found to be particularly successful in reducing dose. Quantitative information on the extent of radiation doses to various work groups shows that significant numbers of workers in several critical groups receive doses greater than 1 and even 2 rem per year, particularly contract personnel and workers at BWR-type plants. The number of radiation workers whose lifetime dose is greater than their age is much less. Although the techniques presented would go some way in reducing dose, it is likely that a sizeable reduction to the high- dose work groups may require development of new dose-reduction techniques as well as major changes in procedures.

**NUREG/CR-5167:** COST/BENEFIT ANALYSIS FOR GENERIC ISSUE 23: REACTOR COOLANT PUMP SEAL FAILURE. NEVE,R.G.; HEISELMANN,H.W. SCIENTECH, Inc. April 1991. 134pp. 9104250014. SCIE-NRC-001-90. 57490:089.

The cost/benefit analysis for Generic Issue (GI-23), "Reactor Coolant Pump Seal Failure," is presented. The cost/benefit analysis comprises three items: (1) treat the reactor coolant pump (RCP) seal assembly as an item performing a safety-related function similar to other of the reactor coolant pressure boundary, applying quality assurance requirements consistent with B of 10 CFR 50 and applicable General Design Criteria of Appendix A, (2) provide RCP manufacturer-recommended instrumentation and instructions for monitoring RCP seal performance and detecting incipient RCP seal failures, and (3) provide RCP seal cooling during off-normal conditions involving loss of all seal cooling such as station blackout. Cost/benefit analysis results are favorable for all items based on the established guideline of $1000/person rem. This report along Technical Findings Document (NUREG/CR-4948) are intended to provide background information and input to the regulatory analysis report for GI-23.

**NUREG/CR-5282:** ESTIMATION OF CONTAINMENT PRESSURE LOADING DUE TO DIRECT CONTAINMENT HEATING FOR THE ZION PLANT. TUTU,N.K. Brookhaven National Laboratory. PARK,C.K. Korea Atomic Energy Research Institute. GRIMSHAW,C.A.; et al. Margrove Consulting, Ltd. March 1991. 57pp. 9103260234. BNL-NUREG-52181. 57159:239.

This report presents the results of a series of calculations at Brookhaven National Laboratory (BNL) to provide estimates of the DCH containment pressure loading in the Zion plant subject to a wide range of initial conditions and phenomenological assumptions. The containment loading calculations were performed using a version of the CONTAIN code with update modifications, which parametrically characterize DCH phenomena (CONTAIN-DCH, Version 1.10). The range of calculation parameters was selected to represent many of the current uncertainties in DCH initial conditions and uncertainties in modeling DCH phenomena. The parameters varied in the sensitivity study included: primary system pressure at vessel failure, core melt inventory, melt and steam flow rates through the reactor cavity, melt droplet size, melt trapping rate, extent of hydrogen combustion, quenching of trapped debris, and co- dispersal of water from reactor cavity. The choice of CONTAIN calculation input parameters is discussed and results are presented for both a seven-cell and a single-cell nodalization of the Zion containment building.

**NUREG/CR-5285:** CLOSEOUT OF IE BULLETIN 79-13: CRACKING IN FEEDWATER SYSTEM PIPING. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. March 1991. 51pp. 9103260106. PARAMETER IE176. 57152:148.

This report documents closeout of IE Bulletin 79-13 regarding cracks in the feedwater system piping of certain PWRs. Closeout is based on implementation and verification of six required actions by licensees and three required actions by designated applicants for operating licenses (DAOLs). Evaluations of licensee responses, NRC/Regional inspection reports, and NRC memoranda in accordance with specific criteria indicates that the bulletin is closed for all of the 54 PWRs required to respond, including 13 DAOLs. It is concluded that (1) actions required by the bulletin have been taken by the affected facilities, and (2) the concerns expressed in the bulletin were validated in that cracks were found and corrected at 18 of the 54 facilities. Background information is provided in the Introduction and Appendix A.

**NUREG/CR-5288:** CLOSEOUT OF IE BULLETIN 80-06:ENGINEERED SAFETY FEATURE (ESF) RESET CONTROLS. FOLEY,W.J.; DEAN,R.S.; KLOEHN,B.A.; et al. PARAMETER, Inc. February 1991. 33pp. 9102280242. PARAMETER IE164. 56836:246.

Documentation is provided in this report for the closeout of IE Bulletin 80-06 regarding the change of safety-related equipment from the emergency or safety mode upon reset of the ESF signal. Closeout is based on the implementation and verification of four required actions. Evaluation of utility responses and NRC/Region inspection reports, in accordance with two criteria, indicates that the bulletin is closed for 61 (95%) of the 64 operating nuclear power facilities to which it was issued for action. Follow-up items are proposed for the three facilities with open status, Browns Ferry 1, 3, and Millstone 1. Facilities which were shut down indefinitely or permanently at the time of issuance of this report are not included in this review. Background information is presented in the Introduction and Appendix A. The conclusion is made that the bulletin concerns have been resolved pending closeout of Browns Ferry 1, 3, and Millstone 1.

**NUREG/CR-5300 V01:** INTEGRATED RELIABILITY AND RISK ANALYSIS SYSTEM (IRRAS) VERSION 2.5.Reference Manual. RUSSELL,K.D.; MCKAY,M.K.; SATTISON,M.B.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). March 1991. 438pp. 9104220309. EGG-2613. 57448:348.

The Integrated Reliability and Risk Analysis System (IRRAS) is a state-of-the-art, microcomputer-based probabilistic risk assessment (PRA) model development and analysis tool to address key nuclear plant safety issues. IRRAS is an integrated software tool that gives the user the ability to create and analyze fault trees and accident sequences using a microcomputer. This program provides functions that range from graphical fault tree construction to cut set generation and quantification. Version 1.0 of the IRRAS program was released in February of 1987. Since that time, many user comments and enhancements have been incorporated into the program providing a much more powerful and user-friendly system. This version has been designated IRRAS 2.5 and is the subject of this Reference Manual. Version 2.5 of IRRAS provides the same capabilities as Version 1.0 and adds a relational data base facility for managing the data, improved functionality, and improved algorithm performance.

**NUREG/CR-5304:** RADIONUCLIDE BEHAVIOR IN THE ENVIRONMENT. TVETEN,U. Institutt for Energiteknikk (Institute for Energy Technology). * Sandia National Laboratories. September 1991. 85pp. 9201060126. SAND90-7116. 60182:221.

Available data on radionuclide behavior are reviewed for quality and consistency in the measurement of (1) initial ground concentration in Norway of radionuclides from Chernobyl, and (2) subsequent concentrations of these radionuclides in various environmental media as a function of time. The data were then used to verify and indicate improvements in consequence models of radionuclide behavior in the MACCS code. The models were of environmental processes such as migration into soil, weathering, resuspension, food chain contamination, and loss or reconcentration by runoff. In most areas of the MACCS code that were examined, the models and the data were in agreement. A few models were found to be faulty or inadequate.

**NUREG/CR-5309:** CLOSEOUT OF IE BULLETIN 83-07: APPARENTLY FRAUDULENT PRODUCTS SOLD BY RAY MILLER,INC. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. March 1991. 43pp. 9103260089. PARAMETER IE200. 57152:093.

Documentation is provided in this report for the closeout of IE Bulletin 83-07 regarding apparently fraudulent products sold by Ray Miller, Inc., to nuclear power and fuel facilities. The bulletin and two supplements were issued to all holders of nuclear power reactor or fuel facility operating licenses or construction permits. Four actions were required of all affected facilities to provide assurance that fraudulent items are not used in safety-related applications, unless qualified by tests. Review of utility responses and NRC/Region inspection reports shows that the bulletin is closed for all of the 118 power facilities and for the two fuel facilities to which it was issued for action. Facilities which were shut down or had construction halted indefinitely or permanently at the time of issuance of this report are not included. It is concluded that all bulletin concerns have been resolved. Background information is supplied in the Introduction and Appendix A.

**NUREG/CR-5312:** A THERMODYNAMIC MODEL OF FUEL DISRUPTION IN ST-1. GRIMLEY,A.J. Sandia National Laboratories. February 1991. 22pp. 9103120079. SAND88-3324. 56942:244.

A thermodynamic model that qualitatively accounts for the observed fuel disruption in the ST-1 test is presented. The model is based on Winslow's equation for the oxygen pressure over hypostoichiometric fuel and the reducing nature of the test atmosphere. The stoichiometry of the fuel is calculated as a function of temperature. This calculation predicts partial liquefaction of the irradiated fuel in the test.

**NUREG/CR-5331:** MELCOR ANALYSES FOR ACCIDENT PROGRESSION ISSUES. DINGMAN,S.E.; SHAFFER,C.J.; PAYNE,A.C.; et al. Sandia National Laboratories. January 1991. 185pp. 9102060162. SAND89-0072. 56592:053.

Results of calculations performed with MELCOR and HECTR in support of the NUREG-1150 study are presented in this report. The analyses examined a wide range of issues. The analyses included integral calculations covering an entire accident sequence, as well as calculations that addressed specific issues that could affect several accident sequences. The results of the analyses for Grand Gulf, Peach Bottom, LaSalle, and Sequoyah are described, and the major conclusions are summarized.

**NUREG/CR-5343:** RADIONUCLIDE CHARACTERIZATION OF REACTOR DECOMMISSIONING WASTE AND SPENT FUEL ASSEMBLY HARDWARE.Progress Report. ROBERTSON,D.E.; THOMAS,C.W.; WYNHOFF,N.C.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. January 1991. 105pp. 9102050011. PNL-6806. 56575:314.

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. Radiological characterization studies at Shippingport have shown that neutron activation products, dominated by (60)Co, make up the residual radionuclide inventory. Fission products and transuranic radionuclides are essentially absent. Waste classification assessments have shown that all decommissioning materials (except reactor pressure vessel internals) could be disposed of as Class A waste. Measurements and assessments of spent fuel assembly hardware have shown that (63)Ni, (59)Ni, and (94)Nb sometimes greatly exceed the 10FR61 Class C limit for some components, and thus would require disposal in a high level waste repository. These measurements are providing the basis for an assessment of the disposal options for these types of highly radioactive materials. Comparisons of predicted (calculated) activation product concentrations with the empirical data are providing an assessment of the accuracy of calculational methods.

**NUREG/CR-5345:** FISSION PRODUCT RELEASE AND FUEL BEHAVIOR OF IRRADIATED LIGHT WATER REACTOR FUEL UNDER SEVERE ACCIDENT CONDITIONS. The ACRR ST-1 Experiment. ALLEN,M.D.; STOCKMAN,H.W.; REIL,K.O.; et al. Sandia National Laboratories. November 1991. 308pp. 9112310162. SAND89-0308. 60159:174.

The Annular Core Research Reactor (ACRR) Source Term (ST) Experiment program was designed to obtain time-resolved data on the release of fission products from irradiated fuels under well-controlled light water reactor severe accident conditions. The ST-1 Experiment was the first of two experiments designed to investigate fission product release. ST-1 was conducted in a highly reducing environment at a system pressure of approximately 0.19 MPa, and at maximum fuel temperatures of about 2490 K. The data will be used for the development and validation of mechanistic fission product release computer codes such as VICTORIA.

**NUREG/CR-5352 R01:** VAM2D - VARIABLY SATURATED ANALYSIS MODEL IN TWO DIMENSIONS. Version 5.2 With Hysteresis And Chain Decay Transport. Documentation And User's Guide. HUYAKORN,P.S.; KOOL,J.B.; WU,Y.S. HydroGeoLogic, Inc. October 1991. 297pp. 9112310172. 60158:237.

This report documents a two-dimensional finite element model, VAM2D, developed to simulate water flow and solute transport in variably saturated porous media. Both flow and transport simulation can be handled concurrently or sequentially. The formulation of the governing equations and the numerical procedures used in the code are presented. The flow equation is approximated using the Galerkin finite element method. Nonlinear soil moisture characteristics and atmosphere boundary conditions (e.g., infiltration, evaporation and seepage face), are treated using Picard and Newton-Raphson iterations. Hysteresis effects and anisotropy in the unsaturated hydraulic conductivity can be taken into account if needed. The contaminant transport simulation can account for advection, hydrodynamic dispersion, linear equilibrium sorption, and first-order degradation. Transport of a single component or a multi-component decay chain can be handled. The transport equation is approximated using an upstream weighted residual method. Several test problems are presented to verify the code and demonstrate its utility. These problems range from simple one-dimensional to complex two-dimensional and axisymmetric problems.

**NUREG/CR-5377:** REVIEW OF THE CHRONIC EXPOSURE PATHWAY MODELS IN MACCS AND SEVERAL OTHER WELL-KNOWN PROBABILISTIC RISK ASSESSMENT MODELS. TVETEN,U. Institutt for Energiteknikk. June 1990. 113pp. 9101300222. 56534:008.

The purpose of this report is to document the results of the work performed by the author in connection with the following task, performed for U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Division of Systems Research: MACCS Chronic Exposure Pathway Models - Review the chronic exposure pathway models implemented in the MELCOR Accident Consequence Code System (MACCS) and compare those models to the chronic exposure pathway models implemented in similar codes developed in countries that are members of OECD. MACCS has been compared to the following internationally well-known the codes: ARANO (Finland), CRAC/CRAC2 (USA), NECTAR (United Kingdom), NUCRAC (USA), UFOMOD (Federal Republic of Germany). A direct comparison has in many respects proved to be difficult to perform, because of the many basic differences between the approaches chosen. But the report contains comprehensive descriptions of the various approaches and default values of most of the important parameters. It also contains numerous remarks/comments at points where the approach chosen (by MACCS or any of the other codes) may have weaknesses or faults; or where the descriptions/manuals are incomplete, difficult to understand, or not consistent with information given in other documentation.

**NUREG/CR-5382:** SCREENING OF GENERIC SAFETY ISSUES FOR LICENSE RENEWAL CONSIDERATIONS. FARAMARZI,A.; HUGHES,A.A.; SETH,S.S. MITRE Corp. December 1991. 82pp. 9112310175. MTR-90W00467. 60153:156.

The U.S. Nuclear Regulatory Commission (NRC) is developing regulations for renewing the operating licenses of nuclear power plants to ensure that they operate safely beyond the present license terms of 40 years. One consideration relates to past resolutions of generic safety issues (GSIs) that did not result in backfit requirements on the licensees. The consideration of an additional operating term of 20 years which the proposed license renewal rule allows, could have retrospective implication for the basis of those GSI resolutions. As part of its technical support to the NRC for the development of license renewal regulations, MITRE has performed an independent review of the GSIs to identify those that could be potentially affected by license renewal considerations. This report describes the screening process and the results of that work.

**NUREG/CR-5395 V01:** MULTILOOP INTEGRAL SYSTEM TEST (MIST):FINAL REPORT.Summary. GLOUDEMANS,J.R. Babcock & Wilcox Co. April 1991. 184pp. 9105220046. EPRI/NP-6480. 57826:165.

The Multiloop Integral System Test (MIST) is part of a multiphase program started in 1983 to address small-break loss-of-coolant accidents (SBLOCAs) specific to Babcock and Wilcox designed plants. MIST is sponsored by the U.S. Nuclear Regulatory Commission, the Babcock & Wilcox Owners Group, the Electric Power Research Institute, and Babcock and Wilcox. The unique features of the Babcock and Wilcox design, specifically the hot leg U-bends and steam generators, prevented the use of existing integral system data or existing integral facilities to address the thermal-hydraulic SBLOCA questions. MIST was specifically designed and constructed for this program, and an existing facility--the Once Through Integral System (OTIS)--was also used. Data from MIST and OTIS are used to benchmark the adequacy of system codes, such as RELAP5 and TRAC, for predicting abnormal plant transients. The MIST program is reported in 11 volumes. Volumes 2 through 8 pertain to groups of Phase 3 tests by type; Volume 9 presents inter-group comparisons; Volume 10 provides comparisons between the RELAP5/MOD2 calculations and MIST observations, and Volume 11 (with addendum) presents the later Phase 4 tests. This is Volume 1 of the MIST final report, a summary of the entire MIST program. Major topics include, Test Advisory Group (TAG) issues, facility scaling and design, test matrix, observations, comparison of RELAP5 calculations to MIST observations, and MIST versus the TAG issues. MIST generated consistent integral-system data covering a wide range of transient interactions. MIST provided insight into integral system behavior and assisted the code effort. The MIST observations addressed each of the TAG issues.

**NUREG/CR-5423:** THE PROBABILITY OF LINER FAILURE IN A MARK-I CONTAINMENT. THEOFANOUS,T.G.; AMARASOORIYA,W.; YAN,H.; et al. California, Univ. of, Santa Barbara, CA. August 1991. 561pp. 9110090256. 59327:140.

An integrated analysis of Mark-I liner attack in a postulated core melt accident is presented. The approach consists of the mechanistic treatment of the sequence of a physical phenomena that lead to liner contact by corium debris, and their coupling through a probabilistic framework that allows representation of uncertainties. We emphasize a physically consistent treatment in each sequence, but allow for qualitatively different scenarios to represent the range of behavior due to model uncertainties. The results are presented in a format that allows their direct use in PRAs, and, in particular, expert opinion is incorporated by a new methodological approach (first applied in our study of alpha-mode failure--NUREG/CR-5030) that involves expert review of, and comment on, a fully documented study all under one cover.

**NUREG/CR-5432 V01:** RECOMMENDATIONS TO THE NRC FOR SOIL COVER SYSTEMS OVER URANIUM MILL TAILINGS AND LOW-LEVEL RADIOACTIVE WASTES. Identification And Ranking Of Soils For Disposal Facility Covers. BENNETT,R.D. Army, Dept. of, Army Engineer Waterways Experiment Station. February 1991. 57pp. 9103200019. 57066:174.

The U.S. Army Engineer Waterways Experiment Station (WES) provides recommendations to the U.S. Nuclear Regulatory Commission (NRC) for the selection, placement, compaction, testing, and acceptance of soils proposed to be placed in cover systems over uranium mill tailings and low-level radioactive wastes. The recommendations from WES are contained in three volumes of NUREG/CR-5432. Volume 1 identifies the various soil types and engineering properties that are needed to fulfill important soil cover functions. The identified soils are then ranked according to their capability to perform the low-permeability and filter and drainage functions. Volume 2 provides recommendations for conducting pertinent laboratory and field tests to ensure acceptable soil cover performance. Volume 3 covers recommendations from WES on proper field construction methods including guidance on quality control testing and inspections. Recommendations are given for sealing penetrations (e.g., observation wells) that are required to penetrate covers for environmental monitoring of disposal facility performance.

**NUREG/CR-5432 V02:** RECOMMENDATIONS TO THE NRC FOR SOIL COVER SYSTEMS OVER URANIUM MILL TAILINGS AND LOW-LEVEL RADIOACTIVE WASTES. Laboratory And Field Tests For Soil Covers. BENNETT,R.D.; HORZ,R.C. Army, Dept. of, Army Engineer Waterways Experiment Station. February 1991. 76pp. 9103200023. 57065:346.

See NUREG/CR-5432,V01 abstract.

**NUREG/CR-5432 V03:** RECOMMENDATIONS TO THE NRC FOR SOIL COVER SYSTEMS OVER URANIUM MILL TAILINGS AND LOW-LEVEL RADIOACTIVE WASTES. Construction Methods And Guidance For Sealing Penetrations In Soil Covers. BENNETT,R.D.; KIMBRELL,A.F. Army, Dept. of, Army Engineer Waterways Experiment Station. February 1991. 81pp. 9103200024. 57066:067.

See NUREG/CR-5432,V01 abstract.

**NUREG/CR-5440:** CRITICAL ASSESSMENT OF SEISMIC AND GEOMECHANICS LITERATURE RELATED TO A HIGH-LEVEL NUCLEAR WASTE UNDERGROUND REPOSITORY. KANA,D.D.; VANZANT,B.W.; et al. Center for Nuclear Waste Regulatory Analyses. BRADY,B.H.G. Itasca Consulting Group, Inc. June 1991. 176pp. 9106250161. CNWRA89-001. 58225:001.

A comprehensive literature assessment has been conducted to determine the nature and scope of technical information available to characterize the seismic performance of an underground repository and associated facilities. Significant deficiencies were identified in current practices for prediction of seismic response of underground excavations in jointed rock. Conventional analytical methods, based on a continuum representation of the host rock mass may have limited applicability in a fractured media. Field observations and laboratory experiments indicate that, in jointed rock, the behavior of the joints controls the overall performance of underground excavations. Further, under repetitive seismic loading, shear displacement develops progressively at block boundaries. Field observations correlating seismicity and groundwater conditions have provided significant information on hydrological response to seismic events. However, lack of a comprehensive model of geohydrological response to seismicity has limited the transportability of conclusions from field observations.

**NUREG/CR-5456:** ANALYSIS OF FLOW STRATIFICATION IN THE SURGE LINE OF THE COMANCHE PEAK REACTOR. SUN,J.G.; SHEN,Y.H.; SHA,W.T. Argonne National Laboratory. April 1991. 58pp. 9105160064. ANL-91/6. 57728:043.

A number of nuclear power plants have reported failure of reactor components due to flow stratification. Therefore, a fundamental understanding of, and a capability to predict, flow stratification in a reactor system is critically important to reactor performance and safety. The work presented here is the first step in this direction and will contribute to the resolution of the issue of flow stratification. An analysis is performed using the COMMIX-1C computer program for the surge line of the Comanche Peak reactor. A comparison is made between the calculated results from the COMMIX code and the plant-measured data, and the agreement is good.

**NUREG/CR-5464:** ANION RETENTION IN SOIL: POSSIBLE APPLICATION TO REDUCE MITGRATION OF BURIED TECHNETIUM AND IODINE.A Review. GU,B.; SCHULZ,R.K. California, Univ. of, Berkeley, CA. October 1991. 42pp. 9111110288. 59576:272.

This report summarizes a literature review of our present knowledge of the anion exchange properties of a number of soils and minerals, which may potentially be used as anion exchangers to retard migration of such anions as iodide (I-), iodate (IO(3)-) and pertechnetate (TcO(4)-) away from the disposal site and thus prevent contamination of ground water. The amorphous clays allophane and imogolite, derived from volcanic parent material, are found to be among the most important soil components capable of developing appreciable amounts of positive charge for anion exchange even at about neutral pH. The magnitude of the surface charge of these amphoteric materials depends on the ratio of SiO(2)/Al(2)O(3), soil pH and concentration of electrolyte. Decreases in the SiO(2)/Al(2)O(3) ratio and soil pH result in an increase in soil AEC. Allophane and imogolite rich soils have an AEC ranging from 1 to 18 meq/100g at pH about 6. Highly weathered soils dominated by Fe and Al oxides and kaolinite may develop a significant amount of AEC as soil pH falls. On a wide range of those soils, AEC ranges from 0 to 2 meq/100g at about pH 6. The retention of radionuclides, iodine (I) and technetium (Tc), by soils is associated with both soil organic matter, and Fe and Al oxides, whereas sorption on layer silicate minerals is negligible. Fe and Al oxides become more important in the retention of anionic I-, IO(3)- and TcO(4)- as pH falls, since more positive charge is developed on the oxide surfaces. Although few studies, if any, have been conducted on I and Tc sorption by soil allophane and imogolite, it is estimated that a surface plough soil (2 million pounds soil per acre) with 5 meq/100g AEC, as is commonly found in andisols, shall retain approximately 5900 kg I and 4500 kg Tc, respectively, by the anion exchange mechanism. It is conceivable that an anion exchanger such as an andisol could be used to modify the near field environment of a radioactive waste disposal facility. This whole disposal system would then offer similar migration resistance to anions as is normally afforded to cations by usual and normal soils. Future studies on this subject are recommended.

**NUREG/CR-5467:** RISK-BASED INSPECTION GUIDE FOR CRYSTAL RIVER UNIT 3 NUCLEAR POWER PLANT. SMITH,R.W.; DUKELOW,J.S.; VO,T.V.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. June 1991. ??pp. 9107080262. PNL-7108. 58309:318.

The Level 1 probabilistic risk assessment (PRA) for Crystal River Unit 3 (CR-3) has been analyzed to identify plant systems and components important to minimizing public risk, as measured by system contributions to plant core damage frequency, and to identify the primary failure modes of these components. The report presents a series of tables, organized by system and prioritized by risk importance, which identify components associated with 98% of the inspectable risk due to plant operation. The systems addressed, in descending order of risk importance are: Low Pressure Injection, AC Power, Service Water, Demineralized Water, High Pressure Injection, DC Power, Emergency Feedwater, Reactor Coolant Pressure Control, and Power Conversion. This ranking is based on the Fussell-Vesely measure of risk importance, i.e., the fraction of the total core damage frequency which involves failures of the system of interest.

**NUREG/CR-5481:** DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-4. OSBORNE,M.F.; LORENZ,R.A.; COLLINS,J.L.; et al. Oak Ridge National Laboratory. January 1991. 72pp. 9101300072. ORNL/TM-11400. 56535:305.

Test VI-4, the fourth in a series of high-temperature fission product release tests in the vertical test apparatus, was conducted in a flowing hydrogen-helium atmosphere. The test specimen was a 15.2-cm-long section of a fuel rod from the BR3 reactor in Belgium, which had been irradiated to a burnup of 47 MWd/kg. Using an induction furnace, it was heated under simulated LWR accident conditions to a test temperature of 2400 K for 20 minutes. Radioactivity measurements showed that the fuel collapsed after the Zircaloy cladding melted during heatup. The total release of fission products from the fuel was 85% for Kr-85, less than 1% for Ru-106, 3.9% for Sb-125, 96% for both Cs-134 and Cs-137, and 13% for Eu-154. Most of the Eu was retained in the furnace, but most of the other elements were released to the collection system. Small fractions of other fission products (Sr, Te, and Ba), as well as fuel (U and Pu) were released also. The total mass release from the 100 g fuel specimen was 0.40 g, with 40% of this as vapor and 60% as aerosol.

**NUREG/CR-5495:** CONCEPTUALIZATION OF A HYPOTHETICAL HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE IN UNSATURATED,FRACTURED TUFF. PARSONS,A.M.; OLAGUE,N.E.; GALLEGOS,D.P. Sandia National Laboratories. January 1991. 177pp. 9102110177. SAND89-2965. 56651:350.

Under the sponsorship of the U.S. Nuclear Regulatory Commission (NRC), Sandia National Laboratories (SNL) is developing a performance assessment methodology for the analysis of long-term disposal and isolation of high-level nuclear waste (HLW) in alternative geologic media. As part of this exercise, SNL created a conceptualization of ground-water flow and radionuclide transport in the far field (i.e., outside the disturbed zone) of a hypothetical HLW repository site located in unsaturated, fractured tuff formations. This study provides a foundation for the development of conceptual, mathematical, and numerical models to be used in this performance assessment methodology. This conceptualization is site specific in terms of geometry, the regional ground-water flow system, stratigraphy, and structure in that these are based on information from Yucca Mountain located on the Nevada Test Site. However, in terms of processes in unsaturated and saturated, fractured, porous media, the model is generic. This report also provides a review and evaluation of previously proposed conceptual models of unsaturated and saturated flow and solute transport. This report provides a qualitative description of a hypothetical HLW repository site in fractured tuff. However, evaluation of the current knowledge of flow and transport at Yucca Mountain does not yield a single conceptual model. Instead, multiple conceptual models are possible given the existing information.

**NUREG/CR-5518:** QUALITY ASSURANCE PROCEDURES FOR THE CONTAIN SEVERE REACTOR ACCIDENT COMPUTER CODE. RUSSELL,N.A.; WASHINGTON,K.E.; BERGERON,K.D.; et al. Sandia National Laboratories. January 1991. 147pp. 9102050015. SAND90-0011. 56576:059.

The CONTAIN quality assurance program follows a strict set of procedures designed to ensure the integrity of the code, to avoid errors in the code, and to prolong the life of the code. The code itself is maintained under a code-configuration control system that provides an historical record of changes. All changes are incorporated using an update processor that allows separate identification of improvements made to each successive code version. Code modifications and improvements are formally reviewed and checked. An exhaustive, multilevel test program validates the theory and implementation of all code changes through assessment calculations that compare the code-predicted results to standard handbooks or idealized test cases. A document trail and archive establish the problems solved by the software, the verification and validation of the software, software changes and subsequent reverification and revalidation, and the tracking of software problems and actions taken to resolve those problems. This document describes in detail the CONTAIN quality assurance procedures.

**NUREG/CR-5520:** PROCEDURES GUIDE FOR EXTRACTING AND LOADING PROBABILISTIC RISK ASSESSMENT DATA INTO MAR-D USING IRRAS 2.5. FOWLER,R.D.; JUDD,D.L.; PHAM,M.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). November 1991. 110pp. 9112310179. EGG-2630. 60158:046.

The Models and Results Data Base (MAR-D) can be used to organize information from probabilistic risk assessments. Data may be entered into MAR-D electronically or manually. This report concentrates on manual data-entry methods and documents the use of the Integrated Reliability and Risk Analysis System (IRRAS), Version 2.5, and ASCII text editors to load level 1 (internal event) PRA models into MAR-D. Step-by-step instructions for using IRRAS 2.5 are provided, which will help the user transfer data from a printed (hardcopy) source to MAR-D.

**NUREG/CR-5522:** A COMPARISON OF PARAMETER ESTIMATION AND SENSITIVITY ANALYSIS TECHNIQUES AND THEIR IMPACT ON THE UNCERTAINTY IN GROUND WATER FLOW MODEL PREDICTIONS. ZIMMERMAN,D.A. Gram, Inc. HANSON,R.T. Interior, Dept. of, Geological Survey. DAVIS,P.A. Sandia National Laboratories. May 1991. 191pp. 9107220255. SAND90-0128. 58468:175.

The Nuclear Regulatory Commission (NRC) and the Environmental Protection Agency (EPA) are the regulating agencies for high-level radioactive waste (HLW) repositories. The regulations promulgated by these agencies stipulate numerical standards for repository performance and explicitly mandate the treatment of uncertainty in the analyses performed in support of a repository license application. This work documents a comparison of sensitivity and uncertainty analysis techniques which are likely to be used in support of repository performance assessments to determine compliance with the regulations. A variety of parameter estimation and sensitivity analysis techniques were applied to a model of the Avra Valley aquifer, Arizona. Differences in the parameter estimates and estimation errors were compared and the effect that these differences have on the uncertainty in the flow model predictions was examined. Also, the effect that different conceptual models can have on the output uncertainty was examined. Uncertainty propagation was performed via Monte Carlo simulation and the importance of screening the transmissivity realizations for realism was evaluated. Two approaches to sensitivity analyses were used, statistical and deterministic; these were applied to evaluate the sensitivity of the groundwater travel time to changes in transmissivity. The effect of different boundary conditions on the calculated sensitivity derivatives was also evaluated. Parameter estimates and estimation errors were obtained via geostatistical and inverse techniques. The patterns in these fields were generally similar, discrepancies were explained. It is shown that the best kriged estimate for use in a flow model is not necessarily the one with the lowest kriging errors. Also, paradoxically, the output uncertainty is greater when the input parameters are correlated versus when they are uncorrelated. The "throughput" of the kriging techniques suggests that the mean estimates derived from these techniques are frequently "off the mark" or inconsistent with the conceptual model. With no screening of the input parameter estimates for realism, non-conservative travel time estimates were obtained. The differential analysis sensitivity technique is shown to be dependent on the choice of design point providing only a local measure of the sensitivity. The statistical approach to sensitivity identifies parameters which are both sensitive and uncertain, i.e., it shows when the uncertainty in a model parameter is important. Sensitivity estimates are also shown to be dependent on the choice of boundary conditions used.

**NUREG/CR-5525:** HYDROGEN-AIR-DILUENT DETONATION STUDY FOR NUCLEAR REACTOR SAFETY ANALYSES. STAMPS,D.W.; BENEDICK,W.B.; TIESZEN,S.R. Sandia National Laboratories. January 1991. 97pp. 9101300088. SAND89-2398. 56534:282.

The detonability of hydrogen-air-diluent mixtures was investigated experimentally in the 0.43 m diameter, 13.1 m long Heated Detonation Tube (HDT) for the effects of variations in hydrogen and diluent concentration, initial pressure, and initial temperature. The data were correlated using a ZND chemical kinetics model. The detonation limits in the HDT were obtained experimentally for lean and rich hydrogen-air mixtures and stoichiometric hydrogen-air-steam mixtures. The addition of a diluent, such as steam or carbon dioxide, increases the detonation cell width for all mixtures. In general, an increase in the initial pressure or temperature produces a decrease in the cell width. In the HDT, the detonable range of hydrogen in a hydrogen-air mixture initially at 1 atm pressure is between 11.6% and 74.9% for mixtures at 20 degrees C, and 9.4% and 76.9% for mixtures at 100 degrees C. The detonation limit is between 38.8% and

40.5% steam for a stoichiometric hydrogen-air-steam mixture initially at 100 degrees C and 1 atm. The detonation limit is between 29.6% and 31.9% steam for a stoichiometric hydrogen-air-steam mixture with a final predetonation mixture temperature and pressure of approximately 100 degrees C and 2.6 atm.

**NUREG/CR-5526:** ANALYSIS OF RISK REDUCTION MEASURES APPLIED TO SHARED ESSENTIAL SERVICE WATER SYSTEMS AT MULTI-UNIT SITES. KOHUT,P.; MUSICKI,Z.; FITZPATRICK,R. Brookhaven National Laboratory. June 1991. 171pp. 9107010104. BNL-NUREG-52225. 58250:245.

This report summarizes a study performed by Brookhaven National Laboratory for the U.S. Nuclear Regulatory Commission in support of the resolution of NRC Generic Issue 130. GI-130 is concerned with the potential core damage vulnerability resulting from failure of the emergency service water (ESW) system in selected multiplant units. These multiplant units are all twin pressurized water reactor designs that have only two ESW pumps per unit (one per train) backed up by a unit-to-unit crosstie capability. This generic issue applies to seven U.S. sites (14 plants). The study established and analyzed the core damage vulnerability and identified potential improvements for the ESW system. It obtained generic estimates of the risk reduction potential and cost effectiveness of each potential improvement. The analysis also investigated the cost/benefit aspects of selected combinations of potential improvements.

**NUREG/CR-5529:** AN ASSESSMENT OF BWR MARK III CONTAINMENT CHALLENGES, FAILURE MODES, AND POTENTIAL IMPROVEMENTS IN PERFORMANCE. SCHROEDER,J.A.; PAFFORD,D.J.; KELLY,D.L.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). January 1991. 307pp. 9102060148. EGG-2594. 56592:262.

This report describes risk-significant challenges posed to Mark III containment systems by severe accidents as identified for Grand Gulf. Design similarities and differences between the Mark III plants that are important to containment performance are summarized. The accident sequences responsible for the challenges and the postulated containment failure modes associated with each challenge are identified and described. Improvements are discussed that have the potential either to prevent or delay containment failure, or to mitigate the offsite consequences of a fission product release. For each of these potential improvements, a qualitative analysis is provided. A limited quantitative risk analysis is provided for selected potential improvements.

**NUREG/CR-5531:** MELCOR 1.8.0: A [illegible]ITER CODE FOR NUCLEAR REACTOR SEVERE AC[illegible] SOURCE TERM AND RISK ASSESSMENT ANAL[illegible] SUMMERS,R.M.; COLE,R.K.; BOUCHERON,E.A.; et al. Sa[illegible] National Laboratories. January 1991. 177pp. 9101300080. SAND90-0364. 56535:128.

MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants. MELCOR is being developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission as a second-generation plant risk assessment tool and the successor to the Source Term Code Package. The entire spectrum of severe accident phenomena, including reactor coolant system and containment thermal-hydraulic response, core heatup, degradation and relocation, and fission product release and transport, is treated in MELCOR in a unified framework for both boiling water reactors and pressurized water reactors. MELCOR has been especially designed to facilitate sensitivity and uncertainty analyses. Its current uses include estimation of severe accident source terms and their sensitivities and uncertainties in a variety of applications. This report is a summary of MELCOR 1.8.0, the code version released in March 1989. Condensed information is presented on its developmental history, structure, modeling features and capabilities, verification and validation, and quality assurance. Detailed documentation on these aspects of MELCOR, including users' guides, reference manuals, programmers' guides, and assessment and application reports, is available in draft form and is distributed to MELCOR users.

**NUREG/CR-5536:** DCM3D: A DUAL-CONTINUUM, THREE-DIMENSIONAL, GROUND-WATER FLOW CODE FOR UNSATURATED, FRACTURED, POROUS MEDIA. UPDEGRAFF,C.D. Gram, Inc. LEE,C.E. Applied Physics, Inc. GALLEGOS,D.P. Sandia National Laboratories. February 1991. 151pp. 9103120075. SAND90-7015. 56948:001.

This report constitutes the user's manual for DCM3D. DCM3D is a computer code for solving three-dimensional, ground-water flow problems in variably saturated, fractured porous media. The code is based on a dual-continuum model with porous media comprising one continuum and fractures comprising the other. The continua are connected by a transfer term that depends on the unsaturated permeability of the porous medium. An integrated finite-difference scheme is used to discretize the governing equations in space. The time-dependent term is allowed to remain continuous. The resulting set of ordinary differential equations (ODE's) is solved with a general ODE solver, LSODES. The code is capable of handling transient, spatially dependent source terms and boundary conditions. The boundary conditions can be either prescribed head or prescribed flux.

**NUREG/CR-5537:** APPROACHES FOR THE VALIDATION OF MODELS USED FOR PERFORMANCE ASSESSMENT OF HIGH-LEVEL NUCLEAR WASTE REPOSITORIES. DAVIS,P.A.; OLAGUE,N.E. Sandia National Laboratories. GOODRICH,M.T. Gram, Inc. March 1991. 34pp. 9104220336. SAND90-0575. 57447:246.

The purpose of this report is to provide general approaches and concepts that can be applied in validation of models used in performance assessment of high-level waste (HLW) repositories. The approaches are based on a validation strategy that Sandia National Laboratories (SNL) has implemented as participants in the International Transport Validation Study (INTRAVAL). This strategy focuses on the demonstration that performance assessment models are adequate representations of the real systems they are intended to represent, given the pertinent regulatory requirements rather than proving absolute correctness from the purely scientific point of view. Positions that are taken consist of the following: (1) due to the relevant time and space scales, models that are used to assess the performance of a HLW repository can never be validated; therefore, (2) validation is a process that consists of building confidence in these models and not providing "validated" models; in this context, (3) model validation includes comparisons to "reality," however, adequacy for the given purpose (assessing compliance with regulations) is the overall goal; (4) comparisons to "reality" consist of comparing model predictions against laboratory and field experiments, natural analogues, and site-specific information; (5) when comparing experimental data to model predictions, a model can be either "invalid" or "not invalid," based on the null hypothesis concept, however, confidence in the model arises in finding a model to be "not invalid" over a wide range of conditions; (6) an attempt should be made to consider in the validation process all plausible conceptual models; and (7) when comparing experimental data to model predictions, a logical systematic approach should be followed (i.e., model input tested separately from model structure). This report discusses (1) the definition of validation in the context of performance assessment for HLW repositories, (2) the need for validation, (3) an approach to validation, and (4) an approach to comparing model predictions with experimental data proposed by the authors.

**NUREG/CR-5538 V01:** INFLUENCE OF ORGANIZATIONAL FACTORS ON PERFORMANCE RELIABILITY.Overview And Detailed Methodological Development. HABER,S.B.; O'BRIEN,J.N.; METLAY,D.S.; et al. Brookhaven National Laboratory. December 1991. 138pp. 9201080112. BNL-NUREG-[illegible]2301. 60232:137.

This is the first volume of a two-volume report. Volume II will be published at a later date. This report presents the results of a research project conducted by Brookhaven National Laboratory for the United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. The purpose of the project was to develop a general methodology to be used in the assessment of the organizational factors which affect performance reliability (safety) in a nuclear power plant. The research described in this report includes the development of the Nuclear Organization and Management Analysis Concept (NOMAC). This concept characterizes the organizational factors that impact safety performance in a nuclear power plant and identifies some methods for systematically measuring and analyzing the influence of these factors on safety performance. This report is divided into two parts: Part I presents an overview of the development of the methodology, while Part II provides more details and a technical analysis of the methodological development. Specifically, the results of two demonstration studies, the feasibility of the methodology, and a specific application for which the methodology was developed are presented.

**NUREG/CR-5539:** A SELF-TEACHING CURRICULUM FOR THE NRC/SNL LOW-LEVEL WASTE PERFORMANCE ASSESSMENT METHODOLOGY. CHU,M.S.Y.; KOZAK,M.W.; CAMPBELL,J.E.; et al. Sandia National Laboratories. January 1991. 195pp. 9102190208. SAND90-0585. 56754:178.

A performance assessment methodology has been developed for use by the U.S. Nuclear Regulatory Commission in evaluating license applications for low-level waste disposal facilities. This report provides detailed guidance on input and output procedures for the computer codes recommended for use in the methodology. Seven sample problems are provided for various aspects of a performance assessment analysis of a simple hypothetical conceptual model. When combined, these sample problems demonstrate how the methodology is used to produce a dose history for the site under normal conditions, and to demonstrate an analysis of an intruder scenario.

**NUREG/CR-5543:** A SYSTEMATIC PROCESS FOR DEVELOPING AND ASSESSING ACCIDENT MANAGEMENT PLANS. HANSON,D.J.; BLACKMAN,H.S.; MEYER,O.R.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). April 1991. 94pp. 9104290253. EGG-2595. 57530:271.

This document describes a four-step approach for developing criteria recommended for use in assessing the adequacy of accident management plans. Two steps of the approach have been completed and provide a prototype process that could be used to develop an accident management plan. Based on this process, a preliminary set of assessment criteria are derived. These preliminary criteria will be refined and improved when the remaining steps of the approach are completed, that is, after the prototype process is validated through application.

**NUREG/CR-5546:** AN INVESTIGATION OF THE EFFECTS OF THERMAL AGING ON THE FIRE DAMAGEABILITY OF ELECTRIC CABLES. NOWLEN,S.P. Sandia National Laboratories. May 1991. 96pp. 9106120194. SAND90-0696. 58062:037.

This report documents the findings of an experimental investigation of the effects of thermal aging on the fire damageability of electric cables. Two popular types of nuclear qualified cables were evaluated. For each cable type, both unaged (i.e., new off the reel) and thermally aged samples were exposed to steady-state elevated temperature environments until conductor-to-conductor electrical shorting was observed. Plots of the time to electrical failure versus the exposure temperature were developed thermal damage thresholds were determined. For one cable type, the thermally aged cables were less vulnerable to thermal damage than were the unaged samples as demonstrated by an increase in the thermal damage threshold for the aged samples, and an extended survival time at exposure temperatures above the damage threshold for aged samples compared to unaged samples. For the second cable, the threshold of thermal damage was lowered somewhat by the aging process, an indication of an increased vulnerability to thermal damage due to aging. However, for the higher temperature exposures, no statistical difference between the damage times for aged and unaged cable samples was noted. For both cable types, the changes in the thermal damage threshold served were not considered significant in terms of fire risk.

**NUREG/CR-5550:** PASSIVE NONDESTRUCTIVE ASSAY OF NUCLEAR MATERIALS. REILLY,D.; ENSSLIN,N.; SMITH,H.; et al. Los Alamos National Laboratory. March 1991. 709pp. 9106210188. LA-UR-90-732. 58858:037.

This book is a general reference on the theory and application of passive nondestructive assay (NDA) techniques, or PANDA. It is part of a four-volume set on nuclear material measurement and accountability sponsored by the US Nuclear Regulatory Commission (NRC). Although we discuss a few active NDA techniques, they have been treated in detail in another book in the NRC series authored by T. Gozani. The book's intended audience ranges from NDA neophytes to experienced practitioners. While the major motivation to write this book was provided by the NRC, there has long been a desire at Los Alamos to prepare a text of this kind. Many of the techniques and instruments described herein were developed at Los Alamos, and we welcome the opportunity to describe the techniques more completely than is possible in reports or papers.

**NUREG/CR-5551:** TWO NEW NDT TECHNIQUES FOR INSPECTION OF CONTAINMENT WELDS BENEATH COATINGS.Final Report,October 1989 - March 1990. FITZPATRICK,G.; THOME,D.K. Physical Research, Inc. June 1991. 26pp. 9107220301. 58489:054.

Two new nondestructive testing methods were evaluated for inspection of containment welds beneath coatings, including magneto-optic imaging and Hall effect measurements. Traditional inspection methods, including magnetic particle inspection, are unsatisfactory in the nuclear containment environment because paint or other coatings must be removed to provide reliable results. This creates radioactive waste, potential airborne contamination, and prolonged radiation exposure to inspection personnel. The new methods offer great improvement because of increased sensitivity and rapid scanning capability. Results obtained during Phase I demonstrated that magneto-optic imaging methods offered good detection of cracking in welded carbon steel samples, even through paint. Direct, real-time images were obtained with this technique in a video format ideal for complete documentation of the full inspection. A new method for rapidly inducing the required magnetic fields for inspection was also demonstrated and offers the potential for eliminating bulky, high current power supplies or magnetic yokes. Results obtained with the Hall effect were not as promising as they were on aluminum, due to electrical interference problems and variable biasing caused by residual magnetic fields in the parts. The technique may still be useful for inspecting tight spaces not accessible with magneto-optic imaging devices, but will require significant development.

**NUREG/CR-5555:** AGING ASSESSMENT OF THE WESTINGHOUSE PWR CONTROL ROD DRIVE SYSTEM. GUNTHER,W.; SULLIVAN,K. Brookhaven National Laboratory. March 1991. 194pp. 9104080294. BNL-NUREG-52232. 57294:246.

A study of the effects of aging on the Westinghouse Control Rod Drive (CRD) System was performed as part of the Nuclear Plant Aging Research (NPAR) Program. The objectives of the NPAR Program are to provide a technical basis for identifying and evaluating the degradation caused by age in nuclear power plant systems, structures, and components. The information from this and other NPAR studies will be used to assess the impact of aging on plant safety and to develop effective mitigating actions. The operating experience data were evaluated to identify predominant failure modes, causes, and effects. For this study, the CRD system boundary includes the power and logic cabinets associated with the manual control of rod movement.

and the control rod mechanism itself. The aging-related degradation of the interconnecting cables and connectors and the rod position indicating system also were considered. The evaluation of the data, when coupled with an assessment of the materials of construction and the operating environment, leads to the conclusion that the Westinghouse CRD system is subject to degradation from aging, which could affect its intended safety function as a plant ages. The number of CRD system failures which have resulted in a reactor trip (challenge to the safety system) warrants a higher level of regulatory and industry attention.

**NUREG/CR-5558:** GENERIC ISSUE 87: FLEXIBLE WEDGE GATE VALVE TEST PROGRAM. Phase II Results And Analysis. STEELE,R.; DEWALL,K.G.; WATKINS,J.C. EG&G Idaho, Inc. (subs. of EG&G, Inc.). January 1991. 83pp. 9102110169. EGG-2600. 56653:222.

Qualification and flow isolation tests were conducted to analyze the ability of selected boiling water reactor process valves to perform their containment isolation functions at high energy pipe break conditions and other more normal flow conditions. Numerous parameters were measured to assess valve and motor-operator performance at various valve loadings and to assess industry practices for predicting valve and motor operator requirements. The valves tested were representative of those used in reactor water cleanup systems in boiling water reactors and those used in boiling water reactor high-pressure coolant injection (HPCI) steam lines. These tests will provide further information for the U.S. Nuclear Regulatory Commission Generic Issue-87, "Failure of the HPCI Steam Line Without Isolation," and Generic Letter 89-10, "Safety-Related Motor Operated Valve Testing and Surveillance."

**NUREG/CR-5561:** ANALYSIS OF BELLOWS EXPANSION JOINTS IN THE SEQUOYAH CONTAINMENT. GREIMANN,L.; WASSEF,W.; FANOUS,F.; et al. Iowa State Univ., Ames, IA. December 1991. 98pp. 9201090203. SAND90-7020. 60244:187.

Bellows expansion joints are an integral part of the containment building pressure boundary in some nuclear power plants. They are used at piping penetrations to minimize the loadings on the containment shell due to differential movement between the shell and piping. The purpose of this study was to investigate bellows behavior in the unlikely event of a severe accident inside the containment building. The study began with a survey of available information on bellows design, analysis, and past test programs. This information was then used to assess the ultimate behavior of the bellows in the Sequoyah containment. It was determined that the bellows at penetration X-47 in the Sequoyah containment would experience the worst loading conditions during a severe accident. Finite element calculations of bellows X-47 were conducted to examine the deformation and resulting strains caused by the combination of axial compression, lateral offset, bending, and internal pressure that would be applied to the bellows during a severe accident. Because of convergence problems, the analyses could not be continued to a point of obvious bellows failure. However, large inelastic bending strains, up to 8%, were calculated. A test program to determine the ultimate bellows behavior and develop data for validation of analytical methods is recommended.

**NUREG/CR-5565:** THE RESPONSE OF BWR MARK II CONTAINMENTS TO STATION BLACKOUT SEVERE ACCIDENT SEQUENCES. GREENE,S.R.; HODGE,S.A.; HYMAN,C.R.; et al. Oak Ridge National Laboratory. May 1991. 313pp. 9106210012. ORNL/TM-11548. 58167:216.

This report describes the results of a series of calculations conducted to investigate the response of BWR Mark II containments to short-term and long-term station blackout severe accident sequences. The BWR-LTAS, BWRSAR, and MELCOR codes were employed to conduct quantitative accident sequence progression and containment response analyses for several station blackout scenarios. The accident mitigation effectiveness of automatic depressurization system actuation, drywell flooding via containment spray operation, and debris quenching in Mark II suppression pools is assessed.

**NUREG/CR-5571:** THE RESPONSE OF BWR MARK III CONTAINMENTS TO SHORT-TERM STATION BLACKOUT SEVERE ACCIDENT SEQUENCES. GREENE,S.R.; HODGE,S.A.; HYMAN,C.R.; et al. Oak Ridge National Laboratory. June 1991. 332pp. 9107230268. ORNL/TM-11549. 58495:026.

This report describes the results of a series of calculations conducted to investigate the response of BWR Mark III containments to short-term station blackout severe accident sequences. The BWR-LTAS, BWRSAR, and MELCOR codes were employed to conduct quantitative accident sequence progression and containment response analyses for several station blackout scenarios. The accident mitigation effectiveness of containment venting and backup emergency power for containment hydrogen ignitors and drywell vacuum breakers is assessed.

**NUREG/CR-5577:** EXTENSION AND EXTRAPOLATION OF J-R CURVES AND THEIR APPLICATION TO THE LOW UPPER SHELF TOUGHNESS ISSUE. JOYCE,J.A. U.S. Naval Academy, Annapolis, MD. HACKETT,E.M. David W. Taylor Naval Research & Development Center. March 1991. 102pp. 9104020091. 57222:193.

This document develops methods of measuring experimentally the limits of valid fracture mechanics data that can be obtained from small fracture mechanics specimens. The proposed technique generally shows that present ASTM limits are overly conservative and the new technique would allow almost a three-fold increase in the amount of crack extension allowed in the testing of a surveillance specimen. Analytic relationships are then developed to allow use of the new experimentally measured limit to J controlled crack growth for design or failure analysis applications to correlate best with the omega criterion which defines limits on both the maximum J level and the maximum crack extension allowable for a particular specimen size and material toughness combination. The final section looks at the problem of extrapolation of J-R curve data when needed for a structure fracture analysis. Several forms of extrapolation relationships are compared from the point of view of accurate and conservative extrapolation, particularly from the standpoint of tearing instability analysis of a growing ductile crack on the material upper shelf.

**NUREG/CR-5581:** UNSATURATED FLOW AND TRANSPORT THROUGH FRACTURED ROCK RELATED TO HIGH-LEVEL WASTE REPOSITORIES.Final Report - Phase III. EVANS,D.D.; RASMUSSEN,T.C. Arizona, Univ. of, Tucson, AZ. January 1991. 79pp. 9102050020. 56576:206.

Research results are summarized for a U.S. Nuclear Regulatory Commission contract with the University of Arizona focusing on field and laboratory methods for characterizing unsaturated fluid flow and solute transport related to high-level radioactive waste repositories. Characterization activities are presented for the Apache Leap Tuff field site. The field site is located in unsaturated, fractured tuff in central Arizona. Hydraulic, pneumatic, and thermal characteristics of the tuff are summarized, along with methodologies employed to monitor and sample hydrologic and geochemical processes at the field site. Thermohydrologic experiments are reported which provide laboratory and field data related to the effects of non-isothermal conditions and flow and transport in unsaturated, fractured rock.

**NUREG/CR-5585:** THE HIGH LEVEL VIBRATION TEST PROGRAM.Final Report. PARK,Y.J.; CURRERI,J.R.; HOFMAYER,C.H. Brookhaven National Laboratory. May 1991. 488pp. 9107030236. BNL-NUREG-52240. 58295:349.

As part of a cooperative study between the United States and Japan, the U.S. Nuclear Regulatory Commission (USNRC) and the Ministry of International Trade and Industry (MITI) of Japan agreed to perform a test program that would subject to a large scale piping model to significant plastic strains under excitation conditions greater than the design condition for nuclear power plants. The objective was to compare the results of the tests with state-of- the-art analyses. Comparisons were done at different excitation levels from elastic to elastic-plastic to levels where cracking was induced in the test model. The vibration tests and post-test examination were carried out in Japan by the Nuclear Power Engineering Test Center (NUPEC). Input motion development and pre- and post-test analysis were carried out in the United States at the Brookhaven National Laboratory (BNL) and the Electric Power Research Institute (EPRI). This report describes the results of the cooperative studies performed both in Japan and the United States.

**NUREG/CR-5592:** ANALYTICAL STUDIES OF TRANSVERSE STRAIN EFFECTS ON FRACTURE TOUGHNESS FOR CIRCUMFERENTIALLY ORIENTED CRACKS. SHUM,D.K.M.; MERKLE,J.G.; KEENEY-WALKER; et al. Oak Ridge National Laboratory. April 1991. 151pp. 9105160051. ORNL/TM-11581. 57729:023.

The objective of this report is to describe the development of analysis methods for estimating the decrease in crack initiation toughness, from a reference plane strain value, due to positive straining along the crack front of a circumferential flaw in a reactor pressure vessel. The analysis methods are based on two different approaches that are currently being developed to analyze and to explain the effects of transverse strain and stress states on fracture toughness. The first approach is a micro-mechanical approach that provides a relation between fracture toughness and more fundamental material properties that can be determined experimentally. The second approach focuses on the development of correlation parameters that relate fracture toughness with nominal stress and strain states. In the first phase of this work, the scope of the investigation is limited to crack front constraint conditions that can be described in terms of conventional one-parameter (K or J) in-plane near-tip fields and the transverse strain. Validation checks of the analysis methods against existing fracture data for conditions of contained crack tip yieldings are promising but incomplete. Recommendations for subsequent phases of the work considered necessary to provide more precise estimates on the effects of positive out-of-plane straining on the crack initiation toughness of circumferentially oriented flaws are included.

**NUREG/CR-5595:** FORECAST: REGULATORY EFFECTS COST ANALYSIS SOFTWARE MANUAL. Version 3.0. LOPEZ,B.; SCIACCA,F.W. Science & Engineering Associates, Inc. November 1991. 123pp. 9112310182. SEA89-461-11-A1. 60156:289.

Over the past several years the NRC has developed a generic cost methodology for the quantification of cost/economic impacts associated with a wide range of new or revised regulatory requirements. This methodology has been developed to aid the NRC in preparing Regulatory Impact Analyses (RIAs). These generic costing methods can be useful in quantifying impacts both to industry and to the NRC. The FORECAST program was developed to facilitate the use of the generic costing methodology. This PC program integrates the major cost considerations that may be required because of a regulatory change. FORECAST automates much of the calculations typically needed in an RIA and thus reduces the time and labor required to perform these analyses. More importantly, its integrated and consistent treatment of the different cost elements should help assure comprehensiveness, uniformity, and accuracy in the preparation of needed cost estimates.

**NUREG/CR-5598:** IMMERSION STUDIES ON CANDIDATE CONTAINER ALLOYS FOR THE TUFF REPOSITORY. BEAVERS,J.A.; DURR,C.L. Cortest Columbus, Inc. May 1991. 122pp. 9105300197. 57868:123.

Cortest Columbus Technologies is investigating the long-term performance of container materials used for high-level radioactive waste packages. This information is being developed for the Nuclear Regulatory Commission to aid in their assessment of the Department of Energy's application to construct a geologic repository for disposal of high-level radioactive waste. This report summarizes the results of exposure studies performed on two copper-base and two Fe-Cr-Ni alloys in simulated Tuff Repository conditions. Testing was performed at 90 C in three environments; simulated J-13 well water, and two environments that simulated the chemical effects resulting from boiling and irradiation of the groundwater. Creviced specimens and U-bends were exposed to liquid, to vapor above the condensed phase, and to alternate immersion. A rod specimen was used to monitor corrosion at the vapor-liquid interface. The specimens were evaluated by electrochemical, gravimetric, and metallographic techniques following approximately 2000 hours of exposure. Results of the exposure tests indicated that all four alloys exhibited acceptable general corrosion rates in simulated J-13 well water. These rates decreased with time. Incipient pitting was observed under deposits on Alloy 825 and pitting was observed on both Alloy CDA 102 and Alloy CDA 715 in the simulated J-13 well water. No stress corrosion cracking (SCC) was observed in U-bend specimens of any of the alloys in simulated J-13 well water. General corrosion rates of the copper-base alloys in an active-corrosion environment were moderate, and no SCC was observed. However, severe pitting and crevice corrosion occurred in this environment. Both Alloy 304L and Alloy 825 exhibited low general corrosion rates with no evidence of localized corrosion (pitting, crevice corrosion or SCC) in a pitting environment in the absence of hydrogen peroxide (H(2)O(2)), a species added to simulate the effects of radiolysis. Alloy 825 continued to exhibit good corrosion performance after H(2)O(2) was added to the pitting environment; whereas, Alloy 304L exhibited both pitting and SCC as a result of the H(2)O(2) addition.

**NUREG/CR-5601:** EFFECTS OF PH ON THE RELEASE OF RADIONUCLIDES AND CHELATING AGENTS FROM CEMENT-SOLIDIFIED DECONTAMINATION ION-EXCHANGE RESINS COLLECTED FROM OPERATING NUCLEAR POWER STATIONS. MCISAAC,C.V.; AKERS,D.W.; MCCONNELL,J.W. EG&G Idaho, Inc. (subs. of EG&G, Inc.). June 1991. 316pp. 9107080245. EGG-2605. 58307:229.

Data are presented on the physical stability and leachability of radionuclides and chelating agents from cement-solidified decontamination ion-exchange resin wastes collected from two operating commercial light water reactors. Small-scale waste-form specimens collected during solidifications performed at the Brunswick Steam Electric Plant Unit 1 and at the James A. FitzPatrick Nuclear Power Station were leach-tested and subjected to compressive strength testing in accordance with the Nuclear Regulatory Commission's "Technical Position on Waste Form" (Revision 1). Samples of untreated resin waste collected from each solidification vessel before the solidification process were analyzed for concentrations of radionuclides, selected transition metals, and chelating agents to determine the quantities of these chemicals in the waste-form specimens. The chelating agents included oxalic, citric, and picolinic acids. In order to determine the effect of leachant chemical composition and pH on the stability and leachability of the waste forms, waste-form specimens were leached in various leachants. Results of this study indicate that differences in pH do not affect releases from cement-solidified decontamination ion- exchange resin waste forms, but that differences in leachant chemistry and the presence of chelating agents may affect the releases of radionuclides and chelating agents. Also, this study indicates that the cumulative releases of radionuclides and chelating agents are similar for waste-form specimens that decomposed and those that retained their

**NUREG/CR-5606:** A REVIEW OF THE SOUTH TEXAS PROJECT PROBABILISTIC SAFETY ANALYSIS FOR ACCIDENT FREQUENCY ESTIMATES AND CONTAINMENT BINNING. WHEELER,T.A.; LAMBRIGHT,J.A.; et al. Sandia National Laboratories. DARBY,J.L. Science & Engineering Associates, Inc. August 1991. 345pp. 9109050259. SAND90-1970. 58989:281.

The objective of this review is to evaluate the South Texas Project (STP) Probabilistic Safety Analysis (PSA) for the USNRC. The PSA was reviewed for thoroughness of analysis, accuracy in plant modeling, legitimacy of assumptions, and overall quality of the work. The review is limited to the internal event analysis and the fire accident analysis. This review is not a pass/fail evaluation of the adequacy of the PSA. The adequacy of the analysis depends on the intended uses and must be addressed on a case-by-case basis by the licensee and the NRC. This review identifies strengths, weaknesses, and areas where additional clarification would assist the NRC in evaluating the PSA for specific regulatory purposes. The licensee, Houston Lighting and Power (HLP), reviewed this report prior to its final release to the NRC. The responses provided by HLP are provided in detail in appendices to this report, and they are summarized in the main body of the report. All issues raised in the review were adequately addressed by HLP in their responses.

**NUREG/CR-5611:** ISSUES AND APPROACHES FOR USING EQUIPMENT RELIABILITY ALERT LEVELS. LOFGREN,E.V.; GREGORY,S.H. Science Applications International Corp. (formerly Science Applications, Inc.). * Brookhaven National Laboratory. June 1991. 137pp. 9107080250. BNL-NUREG-52251. 58308:218.

This report describes work accomplished to identify issues and approaches to establish alert levels for component reliability. Reliability alert levels are established on standby component counts of success and failure, where equipment demands are monitored and counted to ascertain if assumptions about acceptable reliability are likely to be correct. A Monte Carlo simulation was used to determine the detection responses and false alarm rates for several alert level systems. The detection responses were obtained in response to a specified reliability degradation. Two of the alert systems were demonstrated with actual failure data on the Emergency Diesel Generator (EDG) for five plants. Burden and risk measures of effectiveness were developed to compare different alert level schemes having different detection responses and false alarm rates.

**NUREG/CR-5612:** DEGRADATION MODELING WITH APPLICATION TO AGING AND MAINTENANCE EFFECTIVENESS EVALUATIONS. SAMANTA,P.K.; VESELY,W.E.; HSU,F.; et al. Brookhaven National Laboratory. March 1991. 73pp. 9104220330. BNL-NUREG-52252. 57448:028.

This report describes a degradation modeling approach to analyze data on component degradation and failure to understand the processes in aging of components. As used here, degradation modeling is the analysis of information on component degradation in order to develop models of the process and its implications. This particular modeling focuses on the analysis of the times of component degradations, to model how the rate of degradation changes with the age of the component. The methodology presented also discusses the effectiveness of maintenance as applicable to aging evaluations. The specific applications whic presented also discusses the effectiveness of maintenance as applicable to aging evaluations. The specific applications which are performed show quantitative models of component degradation rates and component failure rates from plant-specific data. The statistical techniques which were developed and applied allow aging trends to be effectively identified in the degradation data, and in the failure data. Initial estimates of the effectiveness of maintenance in limiting degradations from becoming failures also were developed. These results are important first steps in degradation modeling and show that degradation can be modeled to identify aging trends.

**NUREG/CR-5614:** PERFORMANCE OF INTACT AND PARTIALLY DEGRADED CONCRETE BARRIERS IN LIMITING FLUID FLOW. WALTON,J.C.; SEITZ,R.R. EG&G Idaho, Inc. (subs. of EG&G, Inc.). July 1991. 56pp. 9107220292. EGG-2614. 58489:093.

Concrete barriers will play a critical role in the long-term isolation of low-level radioactive wastes. Over time the barriers will degrade, and in many cases, the fundamental processes controlling performance of the barriers will be different for intact and degraded conditions. This document examines factors controlling fluid flow through intact and degraded concrete disposal facilities. Simplified models are presented for predicting build up of fluid above a vault; fluid flow through and around intact vaults, through flaws in coatings/liners applied to a vault, and through cracks in a concrete vault; and the influence of different backfill materials around the outside of the vault. Example calculations are presented to illustrate the parameters and processes that influence fluid flow.

**NUREG/CR-5618:** USER'S MANUAL FOR THE NEFTRAN II COMPUTER CODE. OLAGUE,N.E.; LONGSINE,D.E.; CAMPBELL,J.E.; et al. Sandia National Laboratories. February 1991. 259pp. 9103050503. SAND90-2089. 56876:095.

This document describes the NEFTRAN II (NEtwork Flow and TRANsport in Time-Dependent Velocity Fields) computer code and is intended to provide the reader with sufficient information to use the code. NEFTRAN II was developed as part of a performance assessment methodology for storage of high-level nuclear waste in unsaturated, welded tuff. NEFTRAN II is a successor to the NEFTRAN and NWFT/DVM computer codes and contains several new capabilities. These capabilities include: (1) the ability to input pore velocities directly to the transport model and bypass the network fluid flow model, (2) the ability to transport radionuclides in time-dependent velocity fields, (3) the ability to account for the effect of time-dependent saturation changes on the retardation factor, and (4) the ability to account for time-dependent flow rates through the source regime. In addition to these changes, the input to NEFTRAN II has been modified to be more convenient for the user. This document is divided into four main sections consisting of (1) a description of all the models contained in the code, (2) a description of the program and subprograms in the code, (3) a data input guide, and (4) verification and sample problems. Although NEFTRAN II is the fourth generation code, this document is a complete description of the code and reference to past user's manuals should not be necessary.

**NUREG/CR-5619:** THE IMPACT OF THERMAL AGING ON THE FLAMMABILITY OF ELECTRIC CABLES. NOWLEN,S.P. Sandia National Laboratories. March 1991. 35pp. 9103200039. SAND90-2121. 57065:309.

An investigation of the impact of thermal aging on the flammability of two common types of nuclear grade electrical cables has been performed. Four large-scale flammability tests were performed with each of the two cable types tested in both an unaged (i.e., new off the reel) and a thermally aged (artificially aged) condition. In all cases, the fire was observed to consume virtually all of the combustible cable jacket and insulation material present. However, for both cable types tested, the thermal aging process caused a decrease in the cable flammability as demonstrated by decreases in the rate of fire growth, peak fire intensity, total heat released, and near fire temperatures. This result is consistent with past cable aging studies because it has been observed that the thermal aging process will drive off certain of the more volatile constituents of a polymeric material. Presumably, when these aged materials are subjected to a fire, the evolution of volatile combustible gases is reduced as compared to the unaged materials, and hence, flammability is reduced. The results of these tests indicate that, at least for the two cable types tested, the evaluation of cable flammability using unaged cable samples will remain a conservative indicator of cable flammability in a thermally aged condition.

**NUREG/CR-5620:** THATCH: A COMPUTER CODE FOR MODELLING THERMAL NETWORKS OF HIGH-TEMPERATURE GAS-COOLED NUCLEAR REACTORS. KROEGER,P.G.; KENNETT,R.J.; COLMAN,J.; et al. Brookhaven National Laboratory. October 1991. 354pp. 9112310185. BNL-NUREG-52297. 60157:052.

This report documents the THATCH code, which models thermal and flow networks of solids and coolant channels in two-dimensional geometries. The main application of THATCH is for reactor thermo-hydraulic transients in High-Temperature Gas-Cooled Reactors (HTGRs). The code simulates core heatup transients, heat transfer to general sinks or to specific air or water-cooled reactor cavity cooling systems. Graphite oxidation during air or water ingress can be modelled, including the chemical energy release. A point kinetics model is available for reactivity excursions. For most slow HTGR transients a user-selected nodalization of the core in r-z geometry is used. A separate model of heat transfer in the symmetry element of each fuel element is available for rapid transients. The report describes the mathematical models and the method of solution. It describes the code structure and its various procedures. Details of the input data and file usage, is given for the code and for the preprocessing and postprocessing options. The THATCH model of the current 350 MW(th) reactor is described. Input data for four sample cases are given, with output available in fiche form. Installation requirements, code limitations, and some common error indications are listed.

**NUREG/CR-5623:** BWR MARK II EX-VESSEL CORIUM INTERACTION ANALYSES. GREENE,S.R.; LEVIN,A.E.; HYMAN,C.R.; et al. Oak Ridge National Laboratory. November 1991. 222pp. 9112310189. ORNL/TM-11644. 60156:067.

This report describes the results of a series of studies conducted to investigate the behavior of core debris within a BWR Mark II containment. These studies focused on the interaction of core debris with concrete and steel structures (downcomers and in-pedestal floor drains) within the drywell, the transport of debris through these drains and downcomers into the wetwell, and on debris-water reactions within the wetwell. Estimates of the conditions under which debris would penetrate the in-pedestal drain lines, the time-dependent behavior of the debris within the drain lines, and the amount of debris which might enter the suppression pool via these drain lines are provided. An assessment of the conditions under which the upper lip of the downcomers would be expected to fail (i.e. melt) due to exposure to hot core debris is presented. Finally, the unique characteristics of debris water interactions in Mark II containments are discussed, the existing knowledge base regarding core-concrete debris-water interactions is summarized, and an evaluation of the applicability of the MELCOR 1.8.0 code's debris-water interaction model to BWR Mark II's is presented.

**NUREG/CR-5628:** PENNSYLVANIA SEISMIC MONITORING NETWORK AND RELATED TECTONIC STUDIES.Final Report. ALEXANDER,S.S. Pennsylvania State Univ., University Park, PA. June 1991. 37pp. 9107010103. 58251:056.

The magnitude 4.2 earthquake that occurred near Lancaster, Pennsylvania on April 23, 1984 was among the largest in the historic record for that area. The mainshock occurred as an oblique thrust on a steeply dipping NS-striking fault at a focal depth of 4.7 km. The associated principal stress determinations showed a maximum compressive stress oriented approximately N70E in agreement with a large body of crustal stress data elsewhere in eastern North America. Relocation of earlier events near Lancaster revealed an elongated and nearly NS trending zone of seismicity. The activity seems to be associated with cross-strike features that intersect the ENE trending lithologic units of the Triassic Basin in the Lancaster area. Other activity during the monitoring interval of this report was confined to eastern Pennsylvania. In general the earthquakes that occurred are located in areas of past historic seismicity. Block-tectonic structures resulting from pre- Ordovician tectonic displacement appear to influence the distribution of contemporary seismicity in Pennsylvania and surrounding areas.

**NUREG/CR-5630:** PWR DRY CONTAINMENT PARAMETRIC STUDIES. GIDO,R.G.; WILLIAMS,D.C.; GREGORY,J.J. Sandia National Laboratories. April 1991. 212pp. 9105160046. SAND90-2339. 57728:171.

Surry was used as a representative dry containment plant for the evaluation of possible ways that containment performance could be improved. Sensitivity studies using the NUREG-1150 models and methodologies were used to estimate the reduction of risk potentials resulting from bypass scrubbing and DCH partial depressurization. These studies showed that the greatest reduction of risk occurs when bypass releases are mitigated by scrubbing or prevented. High-pressure DCH are also important. The CONTAIN code was used to estimate reduction in peak containment pressure resulting from mitigation actions including venting, partial depressurization, inerting and igniters. Specifically, the reductions were ~2 bar with early depressurization and ~3 bar with igniters. Limited studies of the benefits of venting and inerting were made, but additional investigations are needed to complete this area of investigation. A brief discussion regarding concepts to mitigate the consequences of bypass is presented. CONTAIN-code calculations were performed to investigate the possible overpressurization of the containment for the station blackout scenario.

**NUREG/CR-5634:** IDENTIFICATION AND ASSESSMENT OF CONTAINMENT AND RELEASE MANAGEMENT STRATEGIES FOR A BWR MARK I CONTAINMENT. LIN,C.C.; LEHNER,J.R. Brookhaven National Laboratory. September 1991. 212pp. 9110090260. BNL-NUREG-52259. 59329:072.

This report identifies and assesses accident management strategies which could be important for preventing containment failure and/or mitigating the release of fission products during a severe accident in a BWR plant with a Mark I type of containment. Based on information available from probabilistic risk assessments and other existing severe accident research, and using simplified containment and release event trees, the report identifies the challenges a Mark I containment could face during the course of a severe accident, the mechanisms behind these challenges, and the strategies that could be used to mitigate the challenges. A safety objective tree is developed which provides the connection between the safety objectives, the safety functions, the challenges, and the strategies. The strategies were assessed by applying them to certain severe accident sequence categories which have one or more of the following characteristics: have high probability of core damage or high consequences, lead to a number of challenges, and involve the failure of multiple systems.

**NUREG/CR-5639:** UNCERTAINTY EVALUATION METHODS FOR WASTE PACKAGE PERFORMANCE ASSESSMENT. WU,Y.-T. Center for Nuclear Waste Regulatory Analyses. JOURNEL,A.G. Stanford Univ., Stanford, CA. ABRAMSON,L.R.; et al. Probabilistic Risk Analysis Branch (880717-910829). January 1991. 160pp. 9102280257. 56837:132.

This report identifies and investigates methodologies to deal with uncertainties in assessing high-level nuclear waste package performance. Four uncertainty evaluation methods (probability-distribution approach, bounding approach, expert judgment, and sensitivity analysis) are suggested as the elements of a methodology that, without either diminishing or enhancing the input uncertainties, can evaluate performance uncertainty. Such a methodology can also help identify critical inputs as a guide to reducing uncertainty so as to provide reasonable assurance that the risk objectives are met. This report examines the current qualitative waste containment regulation and shows how, in conjunction with the identified uncertainty evaluation methodology, a framework for a quantitative probability-based rule can be developed, which takes account of the uncertainties. Current NRC regulation requires that the waste packages provide "substantially complete containment" (SCC) during the containment period. The term "SCC" is ambiguous and subject to interpretation. This report, together with an accompanying report which describes the technical considerations that must be addressed to satisfy high-level waste containment requirements, provides a basis for a third report to develop recommendations for regulatory uncertainty reduction in the "containment" requirement of 10 CFR Part 60.

**NUREG/CR-5641:** STUDY OF OPERATIONAL RISK-BASED CONFIGURATION CONTROL. SAMANTA,P.K.; KIM,I.S. Brookhaven National Laboratory. VESELY,W.E. Science Applications International Corp. (formerly Science Applications, Inc.). August 1991. 154pp. 9108290243. BNL-NUREG-52261. 58912:208.

This report studies aspects of a risk-based configuration control system to detect and control plant configurations from a risk perspective. Configuration control, as the term is used here, is the management of component configurations to achieve specific objectives. One important objective is to control risk and safety. Another is to operate efficiently and to make effective use of available resources. PSA-based evaluations are performed to study configuration contributions to core-melt frequency and core-melt probability for two plants. Some equipment configurations can cause large core-melt frequency and there are a number of such configurations that are not currently controlled by technical specifications. However, the expected frequency of occurrence of the impacting configurations is small and the core-melt probability contributions are also generally small. The insights from this evaluation are used to develop the framework for an effective risk-based configuration control system. The focal points of such a system and the requirements for tools development for implementing the system are defined. The requirements of risk models needed for the system, and the uses of plant-specific data are also discussed.

**NUREG/CR-5645:** ACOUSTIC EMISSION/FLAW RELATIONSHIPS FOR INSERVICE MONITORING OF LWRS. HUTTON,P.H.; KURTZ,R.J.; FRIESEL,M.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1991. 1,125pp. 9112300090. PNL-7479. 60126:121.

The program concerning Acoustic Emission/Flaw Relationships for Inservice Monitoring of LWRs was initiated in FY76 with the objective of validating the application of acoustic emission (AE) to monitor nuclear reactor pressure-containing components during operation to detect cracking. The program has been supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research with supplemental support from the Tennessee Valley Authority. Research and development has been performed by Pacific Northwest Laboratory, operated for the Department of Energy by Battelle Memorial Institute. The program has shown the feasibility of continuous, online AE monitoring to detect crack growth and produced validated methods for applying the technology. Included are relationships for estimating flaw severity from AE data and field applications at Watts Bar Unit 1 Reactor, Limerick Unit 1 Reactor, and the High Flux Isotope Reactor. This report discusses the program scope and organization, the three program phases and the results obtained, standard and code activities, and instrumentation and software developed under this program.

**NUREG/CR-5647:** FISSION PRODUCT PLATEOUT AND LIFTOFF IN THE MHTGR PRIMARY SYSTEM: A REVIEW. WICHNER,R.P. Oak Ridge National Laboratory. April 1991. 127pp. 9105030145. ORNL/TM-11685. 57617:163.

A review is presented of methods for predicting radioactivity release resulting from depressurization of an MHTGR primary system. The various types of deposition mechanisms effective for iodine, cesium, strontium, and silver are discussed in terms of their chemical characteristics and the nature of the materials in the primary system. Emphasis is given to iodine behavior, including the quantity available for release, the types of "plateout" locations, and the effect of dust on distribution and release. The behavior of fission products cesium, strontium, and silver in such accidents is presented qualitatively. A major part of the review deals with expected dust levels, types, and transport. Available information on the level and nature of dust in the HTGR primary system is reviewed. A summary is presented of dust deposition and liftoff mechanisms. This study concludes that iodine releases from dry depressurization events are likely to be extremely low, due to low degrees of chemical desorption, liftoff, and a low involvement of iodine with dust. Mechanisms controlling the distribution and liftoff of fission product material in the primary system, depend strongly on the chemical nature of the individual elements. Therefore, both plateout and liftoff models should reflect those unique chemical and physical properties.

**NUREG/CR-5648:** TRANSPORT CALCULATIONS OF NEUTRON TRANSMISSION THROUGH STEEL USING ENDF/B-V,REVISED ENDF/B-V,AND ENDF/B-VI IRON EVALUATIONS. WILLIAMS,M.L.; ABOUGHANTOUS,C.; ASGARI,M.; et al. Louisiana State Univ., Baton Rouge, LA. April 1991. 56pp. 9104290244. ORNL/TM-11686. 57531:005.

The ENDF/B-VI evaluated nuclear data file has been recently released by the U.S. National Nuclear Data Center during 1990. Among the most eagerly awaited new cross-section evaluations in this data collection are those for the natural iron isotopes, due to their importance in nuclear systems analysis and because the previous ENDF/B data (version V, which was released in 1979) are known to underestimate the transmission of fast neutrons through steel structures such as reactor pressure vessels and radiation shielding. In this paper, a comparison is made of results obtained from neutron transport calculations performed with these two ENDF/B versions (V and VI) of iron data as well as an intermediate, revised version V evaluation that was proposed in 1986. Several different response parameters that are sensitive to high energy neutrons are examined, for a variety of geometrical configurations and source spectra. It is found that the two newer iron evaluations substantially increase the transmission of high energy neutrons through steel components with an incident fission spectrum source. Preliminary estimates indicate that the version VI iron evaluation will considerably improve the agreement between calculations and experimental dosimeter measurements used in light water reactor pressure vessel fluence analysis. The calculated leakage spectrum of D-T fusion neutrons from an iron sphere is also improved for energies above 4 MeV, but large discrepancies with the measured spectrum are still observed at lower energies.

**NUREG/CR-5651:** AN INVESTIGATION OF CRACK-TIP STRESS FIELD CRITERIA FOR PREDICTING CLEAVAGE-CRACK INITIATION. KEENEY-WALKER; BASS,B.R.; LANDES,J.D. Oak Ridge National Laboratory. September 1991. 52pp. 9110080367. ORNL/TM-11692. 59326:001.

Cleavage crack initiation in large-scale wide-plate (WP) specimens could not be accurately predicted from small, compact (CT) specimens utilizing a linear elastic fracture mechanics, K(Ic), methodology. In the wide-plate tests conducted by the Heavy-Section Steel Technology Program at Oak Ridge National Laboratory, crack initiation has consistently occurred at stress intensity (K(I)) values ranging from two to four times those predicted by the CT specimens. The work centers around nonlinear two- and three-dimensional finite-element analyses of the crack-tip stress fields in these geometries. Analyses were conducted on CT and WP specimens for which cleavage initiation fracture had been measured in laboratory tests. The local crack-tip fields generated for these specimens were then used in the evaluation of fracture correlation parameters to augment the K(I) parameter for predicting cleavage initiation. Parameters of hydrostatic constraint and of maximum principal stress, measured volumetrically, are included in these evaluations. The results suggest that the cleavage initiation process can be correlated with the local crack-tip fields via a maximum principal stress criterion based on achieving a critical area within a critical stress contour. This criterion has been successfully applied to correlate cleavage initiation in 2T-CT and WP specimen geometries.

**NUREG/CR-5654:** CONTAINMENT VENTING ANALYSIS FOR THE SHOREHAM NUCLEAR POWER STATION. GALYEAN,W.J.; KELLY,D.L. EG&G Idaho, Inc. (subs. of EG&G, Inc.). March 1991. 208pp. 9104080287. EGG-2632. 57321:172.

An evaluation of the Sh... m Mark II containment was performed to identify the eff... of containment venting on core melt frequency, containment failure mode, and offsite consequences. The analysis was based on the Long Island Lighting Company's updated 1988 probabilities risk assessment of the Shoreham plant with the proposed Supplemental Containment System (SCS). The SCS is a filtered containment vent system based on the Swedish Filtra system installed at the Barseback Nuclear Power Station in southern Sweden. The following three different containment venting strategies were examined for their effects on plant risk: (1) venting using the proposed Filtra system, (2) venting using the existing equipment at Shoreham, and (3) no venting. In addition, the consequences of containment venting were examined in conjunction with two sets of assumptions about the effects of a harsh reactor building environment, produced by containment failure or venting through the existing containment and reactor building heating, ventilating, and air conditioning systems, on the equipment located there. Specifically, the analyses studied the consequences when a harsh reactor building environment is assumed to have either no adverse effect on equipment or to fail all equipment.

**NUREG/CR-5655:** SUBMERGENCE AND HIGH TEMPERATURE STEAM TESTING OF CLASS 1E ELECTRICAL CABLES. JACOBUS,M.J. Sandia National Laboratories. FUEHRER,G.F. Science & Engineering Associates, Inc. May 1991. 98pp. 9106180008. SAND90-2629. 58130:143.

This report describes the results of high temperature steam testing and submergence testing of 12 different cable products that are representative of typical cables used inside containments of U.S. light water reactors. Both tests were performed after the cables were exposed to simultaneous thermal and radiation aging, followed by exposure to loss-of-coolant accident simulations. The results of the high temperature steam test indicate the approximate thermal failure thresholds for each cable type. The results of the submergence test indicate that a number of cable types can withstand submergence at elevated temperature, even after exposure to a loss-of-coolant accident simulation.

**NUREG/CR-5656:** EXTRAN: A COMPUTER CODE FOR ESTIMATING CONCENTRATIONS OF TOXIC SUBSTANCES AT CONTROL ROOM AIR INTAKES. RAMSDELL,J.V. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1991. 164pp. 9103260116. PNL-7510. 57154:059.

This report presents the NRC staff with a tool for assessing the potential effects of accidental releases of radioactive materials and toxic substances on habitability of nuclear facility control rooms. The tool is a computer code that estimates concentrations at nuclear facility control room air intakes given information about the release and the environmental conditions. The name of the computer code is EXTRAN. EXTRAN combines procedures for estimating the amount of airborne material, a Gaussian puff dispersion model, and the most recent algorithms for estimating diffusion coefficients in building wakes. It is a modular computer code, written in FORTRAN-77, that runs on personal computers. It uses a math coprocessor, if present, but does not require one. Code output may be directed to a printer or disk files.

**NUREG/CR-5658:** FPFP 2: A CODE FOR FOLLOWING AIRBO...E FISSION PRODUCTS IN GENERIC NUCLEAR PLANT FLOW PATHS. OWCZARSKI,P.C.; BURK,K.W.; RAMS...ELL,J.V.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1991. 82pp. 9104020084. PNL-7513. 57...257.

This report describes the technical bases and use of the computer code FPFP—2 (Fission Product Flow Paths). FPFP—2 was developed to estimate the concentrations and flow rates of airborne fission products along a generic flow path following a transient or puff source of fission products at the beginning of the flow path. This report serves as a user's guide for FPFP—2. A complete code description, code operating instructions, code listing, and an example of the use of FPFP—2 support the use of the code.

**NUREG/CR-5660:** STATIC AND SIMULATED SEISMIC TESTING OF THE TRG-7 THROUGH -16 SHEAR WALL STRUCTURES. FARRAR,C.R.; DOVE,R.C. Los Alamos National Laboratory. BAKER,W.E. New Mexico, Univ. of, Albuquerque, NM. September 1991. 112pp. 9110080384. LA-11992-MS. 59308:323.

Results from the static, simulated seismic base excitation, and experimental modal analysis tests performed on the TRG-7 through -16 structures are reported. These results were used to establish the scalability of static and dynamic response measured on small structural models to the dynamic response of conventional concrete structures. In addition, these tests provided information concerning cumulative damage effects that occur in concrete structures when they are subjected to different dynamic load sequences. In contrast to previous results obtained in the early part of this program, TRG-7 through -16 responded to simulated seismic excitations with theoretical stiffness values until peak nominal base shear stress levels of 150 psi were reached. A summary of all experimental data obtained during this program is provided.

**NUREG/CR-5662:** HYDROGEN COMBUSTION,CONTROL,AND VALUE-IMPACT ANALYSIS FOR PWR DRY CONTAINMENTS. YANG,J.W.; MUSICKI,Z.; NIMNUAL,S. Brookhaven National Laboratory. June 1991. 204pp. 9107230271. BNL-NUREG-52271. 58496:001.

Hydrogen issues applicable to PWRs with dry containment designs are reviewed based on existing information from the NRC's severe accident research program. Additional calculations were performed using the CONTAIN code for a multi-compartment model of the Zion plant. The review includes in-vessel and ex-vessel hydrogen generation, time and modes of hydrogen release, hydrogen mixing and transport in the containment, hydrogen combustion mechanisms, hydrogen control methods and the equipment survivability. A cost-benefit analysis of the hydrogen ignition system was performed for the Zion and Surry plants. Potential for hydrogen detonation in these plants was evaluated.

**NUREG/CR-5663:** RELAP5 THERMAL-HYDRAULIC ANALYSIS OF THE WNP1 PRESSURIZED WATER REACTOR. MARTIN,R.P. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1991. 76pp. 9106040385. EGG-2633. 57904:014.

Thermal-hydraulic analyses of five hypothetical accident scenarios were performed with the RELAP5/MOD3 computer code for the Babcock and Wilcox Company Washington Nuclear Project Unit 1 (WNPl) pressurized water reactor. This work was sponsored by the U.S. Nuclear Regulatory Commission (NRC) and is being performed in conjunction with future analysis work at the NRC Technical Training Center in Chattanooga, Tennessee. The accident scenarios were chosen to assess and benchmark the thermal-hydraulic capabilities of the Technical Training Center WNP1 simulator to model abnormal transient conditions.

**NUREG/CR-5665:** A SYSTEMATIC APPROACH TO REPETITIVE FAILURES. ODLAND,D.J. Sonalysts, Inc. February 1991. 33pp. 9103110198. SI-14940000-1. 56942:209.

This report presents a model of a systematic approach to address and correct repetitive failures. In this context, repetitive failures are the recurring inability of a system, subsystem, structure, or component to perform its intended function. The report presents a systematic method for identifying repetitive failures, selecting the failures to be investigated, determining root cause, selecting corrective actions for implementation, and monitoring subsequent system/component performance. Appendix A provides an example of the use of this methodology at an operating nuclear generating station.

**NUREG/CR-5666:** PROGRAMMATIC ROOT CAUSE ANALYSIS OF MAINTENANCE PERSONNEL PERFORMANCE PROBLEMS. INABA,K. XYZYX Information Corp. January 1991. 105pp. 9102110195. 56652:169.

This report presents a method for diagnosing the programmatic root causes of personnel perfor[illegible] problems in the maintenance of nuclear power plants. [illegible] ess uses normally available maintenance work orde[illegible] identify repeat maintenance caused by inadequate personnel performance. The primary emphasis of the analysis is on corrective maintenance, but the process will detect corrective maintenance actions caused by errors made during scheduled maintenance. Four logic trees are provided to help isolate the causes of such problems to program elements. The program elements consist of the technicians and their managers, as well as support elements and their managers, such as the people responsible for developing procedures and other documentation, developing and delivering training programs, etc. Because of the importance of management commitment to improve maintenance, emphasis is placed on identifying responsible managers, as a primary or co-cause of problems. A sample application of the process to a plant is also presented.

**NUREG/CR-5667:** INEL PERSONAL COMPUTER VERSION OF MACCS 1.5. JONES,K.R.; DOBBE,C.A.; KNUDSON,D.L. EG&G Idaho, Inc. (subs. of EG&G, Inc.). March 1991. 37pp. 9103260227. EGG-2634. 57152:307.

The MELCOR Accident Consequence Code System, Version 1.5 (MACCS 1.5), calculates potential consequences resulting from atmospheric releases of radioactive materials. Sandia National Laboratories developed the code for the U.S. Nuclear Regulatory Commission on a VAX/VMS mini computer. This report documents the Idaho National Engineering [illegible]atory conversion of MACCS 1.5 for compilation and execu[illegible] an 80386-based IBM or IBM-compatible personal compu[illegible]C). The resulting PC version of the code is available throu[illegible] the National Energy Software Center, Argonne National Laboratory, 9700 South Cass Avenue, Argonne, IL 60439.

**NUREG/CR-5668:** DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-5. OSBORNE,M.F.; LORENZ,R.A.; TRAVIS,J.R.; et al. Oak Ridge National Laboratory. October 1991. 62pp. 9111110280. ORNL/TM-11743. 59575:355.

Test VI-5, the fifth in a series of high-temperature fission product release tests, was conducted in a flowing mixture of hydrogen and helium. The test specimen was a 15.2-cm-long section of a fuel rod which had been irradiated to a burnup of ~42 MWd/kg. Using a hot cell-mounted test apparatus, the fuel rod was heated in an induction furnace under simulated LWR accident conditions to two test temperatures, 2000 K for 20 min and then 2700 K for an additional 20 min. Based on fission product inventories measured in the fuel or calculated by ORIGEN2, analyses of test components showed total releases from the fuel of ~100% for (85)Kr, (134)Cs, and (137)Cs; 18% for (125)Sb; and 57% for (154)Eu. Almost 3% of the iodine was collected in a volatile form. In the hydrogen atmosphere, the Zircaloy cladding melted, ran down, and reacted with the UO(2) and some of the fission products. The total mass released from the furnace to the collection system was ~0.681 g. The results from this test were compared with the CORSOR-M and ORNL diffusion release models.

**NUREG/CR-5669:** EVALUATION OF EXPOSURE LIMITS TO TOXIC GASES FOR NUCLEAR REACTOR CONTROL ROOM OPERATORS. MAHLUM,D.D.; SASSER,L.B. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1991. 68pp. 9108190290. PNL-7522. 58828:323.

Pacific Northwest Laboratory (PNL) has evaluated ammonia, chlorine, Halon (actually a generic name for several halogenated hydro-carbons), and sulfur dioxide for their possible effects during an acute two-minute exposure in order to derive recommendations for maximum exposure levels. To perform this evaluation, PNL conducted a search to find the most pertinent literature regarding toxicity in humans and in experimental animals. Much of the literature is at least a decade old, not an unexpected finding since acute exposures are less often performed now than they were a few years ago. In most cases, the studies did not specifically examine the effects of two-minute exposures, thus, extrapolations had to be made from studies of longer exposure periods. Whenever possible, PNL gave the greatest weight to human data, with experimental animal data serving to strengthen the conclusion arrived at from consideration of the human data. Although certain individuals show hypersensitivity to materials like sulfur dioxide, PNL has not attempted to factor this information into the recommendations. After the evaluation of the data in the literature, PNL held a small workshop. Major participants in this workshop were three consultants, all of whom were Diplomates of the American Board of Toxicology, and staff from the Nuclear Regulatory Commission. PNL's preliminary recommendations for two-minute exposure limits and the rationale for them were discussed and consensus reached on final recommendations. These recommendations are: (1) ammonia-300 to 400-ppm; (2) chlorine-30 ppm; (3) Halon 1301-5%; Halon 1211-2%; and (4) sulfur dioxide-100 ppm. Control room operators should be able to tolerate two-minute exposures to these levels, don fresh-air masks, and continue to operate the reactor if the toxic material is eliminated, or safely shut down the reactor if the toxic gas remains.

**NUREG/CR-5670:** MULTILOOP INTEGRAL SYSTEM TEST (MIST):MIST FACILITY FUNCTIONAL SPECIFICATION. HABIB,T.F.; KOKSAL,C.G.; MOSKAL,T.E.; et al. Babcock & Wilcox Co. April 1991. 327pp. 91052?0035. EPRI/NP-7165. 57825:198.

The Multiloop Integral System Test (MIST) is part of a multiphase program started in 1983 to address small-break loss-of-coolant accidents (SBLOCAs) specific to Babcock and Wilcox designed plants. MIST is sponsored by the U.S. Nuclear Regulatory Commission, the Babcock & Wilcox Owners Group, the Electric Power Research Institute, and Babcock and Wilcox. The unique features of the Babcock and Wilcox design, specifically the hot leg U-bends and steam generators, prevented the use of existing integral system data or existing integral facilities to address the thermal-hydraulic SBLOCA questions. MIST was specifically designed and constructed for this program, and an existing facility--the Once Through Integral System (OTIS)--was also used. Data from MIST and OTIS are used to benchmark the adequacy of system codes, such as RELAP5 and TRAC, for predicting abnormal plant transients. The MIST Functional Specification documents as-built design features, dimensions, instrumentation, and test approach. It also represents the scaling basis for the facility and serves to define the scope of work for the facility design and construction.

**NUREG/CR-5672 V01:** CHARACTERISTICS OF LOW-LEVEL RADIOACTIVE WASTE. Decontamination Waste.Annual Report For Fiscal Year 1990. MCISAAC,C.V.; AKERS,D.W. EG&G Idaho, Inc. (subs. of EG&G, Inc.). February 1991. 33pp. 9103040381. EGG-2635. 56860:290.

The objective of Project FIN A6359, "Characteristics of Low-Level Radioactive Waste: Decontamination Waste Forms," funded by the U.S. Nuclear Regulatory Commission, is to provide base-line data on the stability and leachability of solidified decontamination wastes that are generated at operating commercial nuclear power stations following the chemical decontamination of primary coolant systems. This work is being performed to assess the adequacy of tests identified in "Technical Position on Waste Forms," prepared by the NRC Low-Level Waste Management Branch, to meet the requirements of IOCFR61. As part of this project, samples of decontamination waste stream resins and cement waste forms were obtained from commercial nuclear power stations. During Fiscal Year 1990, samples from the FitzPatrick, Brunswick, and Peach Bottom nuclear stations were examined. Samples were subjected to the leach tests described in "Technical Position on Waste Forms" to assess the effects of the decontamination wastes on the stability and leachability of the waste forms. Demineralized water and four different synthetic leachates with pH ranging from 4.2 to 10.4 were used for the tests. The results of these tests are tabulated and preliminary analyses are presented.

**NUREG/CR-5677:** A UNIFIED INTERPRETATION OF ONE-FIFTH TO FULL SCALE THERMAL MIXING EXPERIMENTS RELATED TO PRESSURIZED THERMAL SHOCK. THEOFANOUS,T.G.; YAN,H. California, Univ. of, Santa Barbara, CA. April 1991. 679pp. 9105150314. 57712:162.

Thermal mixing in relation to Pressurized Thermal Shock has been examined experimentally throughout the world in a variety of scales. These include the CREARE-1/5, the IVO/IVO (NRC)-2/5, the PURDUE (UCSB)-1/2, the CREARE-1/2, the HDR-1/1 and the UPTF-1/1 test facilities. The Regional Mixing Model and the associated computer programs REMIX and NEWMIX are used to interpret these data, in this report, in a comprehensive fashion. These interpretations indicate that cooldown transients and degree of stratification can be predicted with confidence. Universal stratification solutions are also provided, in graphical form, and a simple procedure for handcalculation is also described.

**NUREG/CR-5681:** LOW-LEVEL WASTE SOURCE TERM MODEL DEVELOPMENT AND TESTING. SULLIVAN,T.M.; SUEN,C.J. Brookhaven National Laboratory. May 1991. 101pp. 9105300183. BNL-NUREG-52280. 57868:287.

The low-level waste source term model development project has adapted/developed two computer codes to predict the migration of radionuclides emplaced in shallow land burial facilities. The computer code FEMWATER is used to predict water flow and moisture content. The computer code BLT is used to predict container Breach, waste form Leaching, and contaminant Transport. Recent work on this project focused on two areas. One involved improvements to the leaching models incorporated in BLT. In particular, this report describes an additional model that was added to BLT which simulates the waste form using the method of finite differences and treats the contacting solution as a mixing bath. This model improves upon the previous models in BLT in three areas: (a) it treats the release processes of diffusion, dissolution, and surface rinse simultaneously; (b) it allows for partitioning between the waste form and solution; and (c) it permits solution feedback effects to influence diffusive releases. Verification studies of the finite difference/mixing bath model are discussed in detail. The second area of research involved comparing BLT model predictions to experimental data. This report presents the results of modeling laboratory scale wet/dry cycle leach experiments and lysimeter experiments conducted at Pacific Northwest Laboratory. Based on this modeling work, recommendations for future areas of study are given.

**NUREG/CR-5682:** SPECIFIC TOPICS IN SEVERE ACCIDENT MANAGEMENT. MEYER,J.F.; CHUNG,D.T.; PANCIERA,V.W.; et al. SCIENTECH, Inc. May 1991. 200pp. 9106040392. 57904:090.

This report examines five topical areas of concern to severe accident management. These areas are as follows: Human Factors, Accident Management During Shutdown, Information Needs, Long term Implications, and Uncertainties. The objective of this report is to assist the NRC in performing its research function and to provide guidance to the industry on accident management strategies, as well as accident management programs in general.

**NUREG/CR-5683:** LABORATORY TESTING OF CEMENT GROUTING OF FRACTURES IN WELDED TUFF. SHARPE,C.J.; DAEMEN,J.J.K. Arizona, Univ. of, Tucson, AZ. March 1991. 163pp. 9104220323. 57448:115.

The objective of this investigation is to experimentally determine the effectiveness of fracture sealing in welded tuff using ordinary portland cement and microfine cement grouts. Laboratory experiments have been performed on 17 tuff cylinders with three types of fractures: (1) tension-induced cracks, (2) natural fractures, and (3) sawcuts. Prior to grouting, the hydraulic conductivity of the intact rock and of the fractures is measured under a range of normal stresses. The surface topography of the fracture is mapped, and the results are used to determine aperture distributions across the fractures. Grouts are injected through axial boreholes at pressures of 0.3 to 4.1 MPa while holding fractures under a constant normal stress. Five group formulations have been tested. Bentonite (0 to 5 percent by weight) has been added to these grouts to increase their stability. Water-to- cement ratios range from 0.45 to 1.0. Permeability testing of grouted fractures is used to evaluate the effectiveness of fracture grouting. Post-test visual inspection of grout distribution confirms that permeability testing in an injection hole is not a reliable method to assess the effectiveness of grouting. Grout distribution is highly non- uniform.

**NUREG/CR-5684:** ANALYSES AND FIELD TESTS OF THE HYDRAULIC PERFORMANCE OF CEMENT GROUT BOREHOLE SEALS. GREER,W.B.; DAEMEN,J.J.K. Arizona, Univ. of, Tucson, AZ. April 1991. 527pp. 9104300026. 57557:185.

Three tests for determining the hydraulic properties of borehole seals are analyzed in detail. Two consist of monitoring the injection rate of water at constant pressure into one end of a seal and monitoring the collection rate into a free-draining zone at the other end. The third test is performed by shutting in the collection zone and monitoring the buildup in hydraulic head. One-dimensional and axisymmetric three-dimensional flow models are presented for analyzing test results. In the one-dimensional models, the seal is homogeneous and isotropic. In the axisymmetric models, the seal and rock mass are homogeneous and isotropic porous media. The equation for saturated, confined ground-water flow is assumed to apply. The hydraulic properties of the seal are expressed by its hydraulic conductivity and specific storage. In the axisymmetric models, the conductivity and specific storage of the rock mass are included in the formulation. Closed-form solutions are presented for the one-dimensional models. Numerical analysis with the axisymmetric models uses an available finite element code for ground-water flow. We examine the effects of variations in hydraulic parameters on the quantities measured in the tests (i.e., flow rates or head) and compare the one- dimensional and axisymmetric models. Methods are presented for obtaining the hydraulic properties of the seal and/or rock mass by analysis of test results. A fourth test, a tracer travel-time test, presents a means for detecting any high-velocity flow path through or around the seal. The test methods are applied to cement grout borehole seals from 10 to 36 cm in length and 10 cm in diameter in a nearly horizontal hole and in three vertical holes.

**NUREG/CR-5686:** EFFECTIVENESS OF FRACTURE SEALING WITH BENTONITE GROUTING. RAN,C.; DAEMEN,J.J.K. Arizona, Univ. of, Tucson, AZ. June 1991. 192pp. 9107080258. 58309:126.

Bentonite is known to have an extremely low permeability and a self-healing ability. It has therefore been selected as a major sealing component in several repository concepts. Bentonite grouts have the following advantages: (1) small particle size, can be injected into small fractures or voids, (2) suitable water absorption properties, can produce gels at low concentrations and (3) stable physical and chemical properties, may have considerable longevity. Bentonite fracture grouting tests are performed on a model made of circular acrylic plates with outer diameter of 30 cm and a central injection hole of 2.5 cm diameter. Suspensions with bentonite concentration of 15% to 31% have been injected into fractures with apertures of 9 to 90 microns under injection pressures less than 0.6 MPa. Grouting reduces the hydraulic conductivities of the fractures from the 10(-1) to the 10(-5) cm/s level. When the suspension is thin enough and the fracture is very small, channeling develops in the grouted fractures. Preliminary results indicate that the permeability of a grouted fracture does not increase with time in more than 125 days. The flow properties of bentonite suspensions, viscosity, shear stress, yield stress and gelation, are investigated. Water flow through ungrouted fractures and movement of water in bentonite grout are studied. The physical stability or bleeding capacity of bentonite suspensions is determined.

**NUREG/CR-5688:** MECHANICAL CHARACTERIZATION OF DENSELY WELDED APACHE LEAP TUFF. FUENKAJORN,K.; DAEMEN,J.J.K. Arizona, Univ. of, Tucson, AZ. June 1991. 125pp. 9107080255. 58309:001.

An empirical criterion is formulated to describe the compressive strength of densely welded Apache Leap tuff. It incorporates the effects of size, L/D ratio, loading rate and density variations, and improves the correlation between the test results and the failure envelope. Uniaxial and triaxial compressive strengths, Brazilian tensile strength and elastic properties of the densely welded brown unit of Apache Leap tuff have been determined using the ASTM standard test methods. All tuff samples are tested dry at room temperature with the core axis normal to the flow layers. The uniaxial compressive strength is $73.2 \pm 16.5$ MPa. The Brazilian tensile strength is $5.12 \pm 1.2$ MPa. The Young's modulus and Poisson's ratio are $22.6 \pm 5.7$ GPa and $0.20 \pm 0.03$. Smoothness and perpendicularity do not fully meet ASTM requirements for all samples, due to voids and inclusions on the sample surfaces and the sample preparation methods. The investigations of loading rate, L/D ratio and cyclic loading effects on compressive strength and of the size effect on tensile strength are not conclusive. The Coulomb strength criterion adequately represents the failure envelope of the tuff under confining pressures from 0 to 62 MPa. The tuff is highly heterogeneous as suggested by large variations in the results. The variability is probably caused by flow layers and by non-uniform distributions of inclusions, voids and degree of welding. Similar variability of properties has been reported elsewhere for the Topopah Spring tuff at Yucca Mountain.

**NUREG/CR-5689:** MEDICAL SCREENING REFERENCE MANUAL FOR SECURITY FORCE PERSONNEL AT FUEL CYCLE FACILITIES POSSESSING FORMULA QUANTITIES OF SPECIAL NUCLEAR MATERIALS. ARZINO,P.A.; BROWN,C.H. California State Univ., Hayward Foundation, Inc., Hayward, CA. September 1991. 30pp. 9110110233. 59358:097.

The recommendations contained throughout this NUREG were provided to the Nuclear Regulatory Commission (NRC) as medical screening information that could be used by physicians who are evaluating the parameters for the safe participation of guards, Tactical Response Team members (TRTs), and all other armed response personnel in physical fitness training and in physical performance standards testing. The information provided in this NUREG will help licensees determine if guards, TRTs, and other armed response personnel can effectively perform their normal and emergency duties without undue hazard to themselves, to fellow employees, to the plant site, and to the general public. The medical recommendations in this NUREG are similar in content to the medical standards contained in 10 CFR Part 1046 which, in part, specifies medical standards for the protective force personnel regulated by the Department of Energy. The guidelines contained in this NUREG are not requirements, and compliance is not required.

**NUREG/CR-5690:** PHYSICAL FITNESS TRAINING REFERENCE MANUAL FOR SECURITY FORCE PERSONNEL AT FUEL CYCLE FACILITIES POSSESSING FORMULA QUANTITIES OF SPECIAL NUCLEAR MATERIALS. ARZINO,P.A.; CAPLAN,C.S.; GOOLD,R.E. California State Univ., Hayward Foundation, Inc., Hayward, CA. September 1991. 39pp. 9110110218. 59358:059.

The recommendations contained throughout this NUREG are being provided to the Nuclear Regulatory Commission (NRC) as a reference manual which can be used by licensee management as they develop a program plan for the safe participation of guards, Tactical Response Team members (TRTs), and all other armed response personnel in physical fitness training and in physical performance standards testing. The information provided in this NUREG will help licensees determine if guards, TRTs, and other armed response personnel can effectively perform their normal and emergency duties without undue hazard to themselves, to fellow employees, to the plant site, and to the general public. The recommendations in this NUREG are similar in part to those contained within the Department of Energy (DOE) Medical and Fitness Implementation Guide which was published in March 1991. The guidelines contained in this NUREG are not requirements, and compliance is not required.

**NUREG/CR-5691:** INSTRUMENTATION AVAILABILITY FOR A PRESSURIZED WATER REACTOR WITH A LARGE DRY CONTAINMENT DURING SEVERE ACCIDENTS. ARCIERI,W.C.; HANSON,D.J. EG&G Idaho, Inc. (subs. of EG&G, Inc.). March 1991. 128pp. 9106120192. EGG-2638. 58062:133.

In support of the U.S. Nuclear Regulatory Commission (NRC) Accident Management Research Program, the availability of instruments to supply accident management information during a broad range of severe accidents is evaluated for a pressurized water reactor with a large dry containment. Results from this evaluation include the following: (a) identification of plant conditions that would impact instrument performance and information needs during severe accidents, (b) definition of envelopes of parameters that would be important in assessing the performance of plant instrumentation for a broad range of severe accident sequences, and (c) assessment of the availability of plant instrumentation during severe accidents.

**NUREG/CR-5692:** GENERIC RISK INSIGHTS FOR GENERAL ELECTRIC BOILING WATER REACTORS. TRAVIS,R.; TAYLOR,J. Brookhaven National Laboratory. CHUNG,J. Risk Application Branch. May 1991. 88pp. 9106210013. BNL-NUREG-52282. 58167:132.

A methodology has been developed to extract generic risk-based information from probabilistic risk assessments (PRAs) of General Electric boiling water reactors and apply the insights gained to plants that have not been subjected to a PRA. The available risk assessments (six plants) were examined to identify the most probable, i.e., dominant accident sequences at each plant. The goal was to include all sequences which represented at least 80% of core damage frequency. If the same plant specific dominant accident sequence appeared within this boundary in at least two plant PRAs, the sequence was considered, to be a representative sequence. Eight sequences met this definition. From these sequences, the most important component failures and human errors that contributed to each sequence have been prioritized. Guidance is provided to prioritize the representative sequences and modify selected basic events that have been shown to be sensitive to the plant specific design or operating variations of the contributing PRAs. This risk- based guidance can be used for utility and NRC activities including operator training, maintenance, design review, and inspections.

**NUREG/CR-5695:** A PROCESS FOR RISK-FOCUSED MAINTENANCE. LOFGREN,E.V.; COOPER,S.E.; KURTH,R.E.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). March 1991. 141pp. 9104080273. 57321:031.

This report presents a process for focusing maintenance resources on components that enable nuclear plant systems to perform their essential functions and on components whose failure may initiate challenges to safety systems, so as to have the greatest impact in decreasing risk. The process provides criteria, based on risk, for deciding which components are critical to risk and determining what maintenance activities are required to ensure reliable operation of those "risk-critical" components. Two approaches are provided for selection of risk-critical components. One approach uses the results of a Probabilistic Risk Assessment (PRA); the other is based on the methodology developed for this report, which has a basis in PRA although it does not use the results of a PRA study. Following identification of risk-critical components, both approaches use a single methodology for determining what maintenance activities are required to ensure reliable operation of the identified components. The report also provides demonstrations of application of the two approaches to selection of risk-critical components and demonstrations of application of the methodology for determining what maintenance activities are required to an active standby safety system, a normally operating system, and passive components.

**NUREG/CR-5696:** IRRADIATION EFFECTS ON CHARPY IMPACT AND TENSILE PROPERTIES OF LOW UPPER-SHELF WELDS,HSSI SERIES 2 AND 3. NANSTAD,R.K.; BERGGREN,R.G. Oak Ridge National Laboratory. August 1991. 240pp. 9110090295. ORNL/TM-11804. 59326:111.

The objective of the Second and Third Irradiation Series was to investigate the effects of irradiation on the ductile fracture toughness of seven commercially fabricated, low upper-shelf welds. All seven submerged-arc welds were fabricated with copper-coated wire and Linde 80 flux and had average bulk copper contents from 0.21 to 0.42% with nickel levels of about 0.6%. In addition to the fracture toughness specimens which were irradiated at nominally 288 degrees C, Charpy V-notch and tensile specimens were included in the capsules at available locations which were subject to wide variations in irradiation temperature and fluence. This report presents analyses of the Charpy impact and tensile test data. Analyses revealed a dependence of yield strength on irradiation temperature of -1.1 MPa/degrees C, while the Charpy impact energy dependencies were about -0.5 degrees C/degrees C for transition temperature shift and -0.06 J/degrees C for upper-shelf decrease. After adjustment to an irradiation temperature of 288 degrees C and normalization to a fluence of 8 x 10(18) neutrons/cm(2) (>1 MeV), the Charpy transition temperature shifts ranged from 59 to 123 degrees C while the upper-shelf energies ranged from 58 to 79 J.

**NUREG/CR-5697:** USE OF THICKNESS REDUCTION TO ESTIMATE VALUES OF K. IRWIN,G.R. Maryland, Univ. of, College Park, MD. * Oak Ridge National Laboratory. November 1991. 24pp. 9112310207. ORNLSUB797778/5. 60155:285.

Using results for two 152-mm-thick wide-plate tests at the National Institute of Standards and Technology, estimates of K were made using the residual thickness reduction near the plane of fracture. These results corresponded well to the average of K values for cleavage arrest and reinitiation obtained at Oak Ridge National Laboratory using generation-mode, dynamic-analysis computations.

**NUREG/CR-5701:** A PERFORMANCE ASSESSMENT METHODOLOGY FOR HIGH-LEVEL RADIOACTIVE WASTE DISPOSAL IN UNSATURATED,FRACTURED TUFF. GALLEGOS,D.P. Sandia National Laboratories. July 1991. 45pp. 9107230228. SAND91-0539. 58546:161.

Sandia National Laboratories, under contract to the U.S. Nuclear Regulatory Commission, has developed a methodology for performance assessment of deep geologic disposal of high-level nuclear waste. The applicability of this performance assessment methodology has been demonstrated for disposal in bedded salt and basalt; it has since been modified for assessment of repositories in unsaturated, fractured tuff. Changes to the methodology are primarily in the form of new or modified ground water flow and radionuclide transport codes. A new computer code, DCM3D, has been developed to model three-dimensional ground-water flow in unsaturated, fractured rock using a dual-continuum approach. The NEFTRAN II code has been developed to efficiently model radionuclide transport in time-dependent velocity fields, has the ability to use externally calculated pore velocities and saturations, and includes the effect of saturation-dependent retardation factors. In order to use these codes together in performance-assessment-type analyses, code-coupler programs were developed to translate DCM3D output into NEFTRAN II input. In addition to flow and transport codes, other portions of performance assessment methodology were evaluated as part of modifying the methodology for tuff. The scenario methodology developed under the bedded salt program, considered adequate, was not altered, but has been applied to tuff. An investigation of the applicability of uncertainty and sensitivity analysis techniques to non-linear models indicates that Monte Carlo simulation remains the most robust technique for these analyses. No changes have been recommended for the dose and health effects models, nor the biosphere transport models. Additionally, a number of outstanding, but unresolved, technical issues have been identified.

**NUREG/CR-5702:** ACCIDENT MANAGEMENT INFORMATION NEEDS FOR A BWR WITH A MARK I CONTAINMENT. CHIEN,D.N.; HANSON,D.J. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1991. 159pp. 9105220020. EGG-2639. 57824:001.

In support of the U.S. Nuclear Regulatory Commission Accident Management Research Program, information needs during severe accidents have been evaluated for Boiling Water Reactors (BWRs) with MARK I containments. This evaluation was performed using a methodology that identifies plant information needs necessary for personnel to: (a) diagnose that an accident is in progress, (b) select and implement strategies to prevent or mitigate the accident, and (c) monitor the effectiveness of these stogress, (b) select and implement strategies to prevent or mitigate the accident, and (c) monitor the effectiveness of these strategies. The information needs and capabilities identified are intended to form a basis for more comprehensive information needs assessments. These assessments will be performed during the analysis and development of specific strategies, which will be used in accident management prevention and mitigation.

**NUREG/CR-5703:** LOWER-BOUND INITIATION TOUGHNESS WITH A MODIFIED-CHARPY SPECIMEN. DALLY,J.W.; FOURNEY,W.L.; IRWIN,G.R.; et al. Maryland, Univ. of, College Park, MD. November 1991. 44pp. 9112310203. ORNL-SUB797778/7. 60155:311.

"Lower-bound" initiation toughness of A 533 B reactor grade steel was determined over the temperature range from 0 to 57 degrees C by using a modified-Charpy specimen. The lower-bound measurements were attained by utilizing the following procedures: (1) dynamic loading, (2) modification of the geometry of the specimen, and (3) axial precompression of the notch. The report describes in detail the key features of the modified geometry, the method of precompressing the specimens, and the strain-gage procedure. The dynamic initiation toughness K(Id), which correlates with the lower-bound toughness, was determined by analyzing strain-time records from the specimen. The results from a fractogaphic analysis were correlated with those from the strain-time analysis. An empirical correlation was developed relating K(I) to the energy absorbed (E(cvi)) during the fracture of the specimen. Finally, the lower-bound toughness from this study compared favorably with K(I) and K(Id) measurements from the same material established in other programs.

**NUREG/CR-5706:** POTENTIAL SAFETY-RELATED PUMP LOSS: AN ASSESSMENT OF INDUSTRY DATA.NRC Bulletin 88-04. CASADA,D.A. Oak Ridge National Laboratory. June 1991. 53pp. 9107010081. ORNL-6671. 58248:306.

This report documents the results of a study of the nuclear industry's response to NRC Bulletin 88-04. The work was conducted for the U.S. Nuclear Regulatory Commission (NRC) Nuclear Plant Aging Research Program. All written correspondence between utilities and the NRC was reviewed and classified. Major pump vendors were interviewed to discuss their perspectives on low-flow degradation of pumps. Individual sites were visited to review the details of system design and procedural controls relative to the Bulletin issues.

**NUREG/CR-5707:** APPLICATION OF CONTAINMENT AND RELEASE MANAGEMENT TO A PWR ICE-CONDENSER PLANT. NEOGY,P.; LEHNER,J.R. Brookhaven National Laboratory. July 1991. 104pp. 9108190283. BNL-NUREG-52286. 58828:219.

This report identifies and evaluates accident management strategies that are potentially of value in maintaining containment integrity and controlling the release of radioactivity following a severe accident at a pressurized water reactor with an ice condenser containment. The strategies are identified using a logic tree structure leading from the safety objectives and safety functions, through the mechanisms that challenge these safety functions, to the strategies. The strategies are applied to severe accident sequences which have one or more of the following characteristics: significant probability of core damage, high consequences, give rise to a number of potential challenges, and include the failure of important safety systems.

**NUREG/CR-5711:** ASSESSMENT OF UNCERTAINTIES IN MEASUREMENT OF PH IN HOSTILE ENVIRONMENTS CHARACTERISTIC OF NUCLEAR REPOSITORIES. KREIDER,K.G.; TARLOV,M.J.; HUANG,P.H. National Institute of Standards & Technology (formerly National Bureau of Standa. October 1991. 105pp. 9111110285. 59576:167.

This report focuses on evaluation and characteristics of sputtered thin film pH electrodes which can be used to assess the corrosivity of hot (100 degrees C) aqueous solutions present in nuclear repositories. Sputtered thin films have the advantages of high temperature capability, ruggedness, and low cost. The iridium oxide films were found to have a linear, 58mV/pH, response to changes in pH. They had little hysteresis but drifted approximately 0.2V over a period of two days exposure to pH 2-12 solutions. The films were found to be insensitive to interference from most ions such as alkali ions but had redox sensitivity to ferri-/ferrocyanide ions. Although special surface treatments were needed for the films for good adherence at 200 degrees C the films were not degraded after 20 hours exposure at pH 4, 7, and 10 at 200 degrees C. Ruthenium oxide sputtered films performed equally well to the iridium oxide films in parallel tests. The report also contains information on electrochemistry and testing of thin film electrodes and the characterization of the thin films by x-ray photoemission spectroscopy, ultraviolet photoemission spectroscopy, and ion scattering spectroscopy.

**NUREG/CR-5712:** MORECA: A COMPUTER CODE FOR SIMULATING MODULAR HIGH-TEMPERATURE GAS-COOLED REACTOR CORE HEATUP ACCIDENTS. BALL,S.J. Oak Ridge National Laboratory. October 1991. 70pp. 9112310198. ORNL/TM-11823. 60156:001.

The design features of the modular high-temperature gas-cooled reactor (MHTGR) have the potential to make it essentially invulnerable to damage from postulated core heatup accidents. This report describes the ORNL MORECA code, which was developed for analyzing postulated long-term core heatup scenarios for which active cooling systems used to remove afterheat following the accidents are not necessarily available. Simulations of long-term loss-of-forced-convection accidents, both with and without depressurization of the primary coolant, have shown that maximum core temperatures stay below the point at which any significant fuel failures and fission product releases are expected. Sensitivity studies also have been done to determine the effects of errors in the predictions due both to uncertainties in the modeling and to the assumptions about operational parameters. MORECA models the U.S. Department of Energy reference design of a standard MHTGR. This program was sponsored by the U.S. Nuclear Regulatory Commission to assist in the preliminary determinations of licensability of the reactor design.

**NUREG/CR-5713:** A REVIEW OF ENVIRONMENTAL CONDITIONS AND PERFORMANCE OF THE COMMERCIAL LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY NEAR SHEFFIELD,ILLINOIS. MURPHY,E.M.; BERGERON,M.P. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1991. 137pp. 9105300177. PNL-7621. 57869:028.

The Sheffield low-level radioactive waste disposal site is located about 5 km southwest of the town of Sheffield, Bureau County, in northwestern Illinois. Low-level radioactive waste was buried at the site between August 1967 and April 1978. The ground-water system beneath the Sheffield site can be conceptualized as containing two separate aquifer systems: a regional confined bedrock aquifer system and a local unconfined aquifer system in the shallow sequence of unconsolidated quaternary-aged sediments. The most significant hydrogeologic unit on the site is a pebbly-sand unit found within the Toulon Member of the Glasford Formation that grades into a coarse gravel with sand and pebbles east of the disposal site. In an area east of the site, a narrow, channel-like depression is filled with coarse, gravelly sand of the pebbly-sand unit of the Toulon Member, providing a hydraulic connection between the site and nearby strip-mine lake. Three major problems resulting from the waste burial at the Sheffield site include subsidence of trench covers, significant erosion, and elevated concentrations of tritium in the vadose zone and ground water at Sheffield.

**NUREG/CR-5714:** HYDROGEOLOGIC PERFORMANCE ASSESSMENT ANALYSIS OF THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY NEAR SHEFFIELD, ILLINOIS. BERGERON,M.P.; HOLFORD,D.J.; KEMNER,M.L.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1991. 143pp. 9105300168. PNL-7633. 57869:155.

A hydrogeologic performance assessment was conducted for the commercial low-level radioactive waste disposal site located about 3 mi southwest of the town of Sheffield, in Bureau County, northwestern Illinois. The site has 21 trenches, which contain about 900,000 m(3), of buried waste and about 60,000 Ci of nuclear by-product material. The disposal trenches cut through a complex series of Quaternary deposits, and are composed primarily of silts, clays, and sands. Ground water beneath the site, which ranges in depth from 1.5 to 14 m, generally moves in two directions: northeast to east toward a strip-mine lake and south to southeast toward small tributary channels belonging to Lawson Creek, which eventually drains into the strip-mine lake southeast of the site. The results in the performance assessment, which focused on the site ground-water pathway, suggest that tritium, Sr-90, and C-14 would be the only radionuclides released from the Sheffield site in any significant concentrations. A comparison of simulated tritium concentrations east of the site in the time frame of the burial history would suggest that model results are greater than the highest measured values by a factor of 2 or 3. The discrepancy between actual and predhat model results are greater than the highest measured values by a factor of 2 or 3. The discrepancy between actual and predicted concentrations likely reflects errors in the assumed tritium inventory estimates, availability in the inventory, and/or the actual release from the multitude of waste forms considered in the performance assessment. A comparison of transport results for Sr-9O and C-14 is not possible since neither has been detected in ground water near the site.

**NUREG/CR-5715:** REFERENCE MANUAL FOR THE CONTAIN 1.1 CODE FOR CONTAINMENT SEVERE ACCIDENT ANALYSIS. WASHINGTON,K.E.; MURATA,K.K.; GIDO,R.G.; et al. Sandia National Laboratories. July 1991. 260pp. 9108130345. SAND91-0835. 58781:001.

This report describes the phenomenological equations and the numerical procedures used by the CONTAIN 1.1 code to determine the conditions within nuclear power plant containment during a severe accident. The CONTAIN detailed models provide the capability to mechanistically calculate the containment internal thermalhydraulic conditions and the amount of radioactive matter that would be released to the environment if there were a leak from the containment. Note that the CONTAIN models can be verified by comparing the code calculations to experimental results. The models described include those to account for the flows of mass and energy between containment compartments, the exchange of energy between the atmosphere and heat structures, the thermodynamic conditions, the distributions of aerosols, the decay and transport of fission products, the deflagration of hydrogen and carbon monoxide, boiling water reactor suppression pool behavior, and engineered safety features, including a spray, fan coolers, and an ice condenser. These models are solved with implicit coupling, where appropriate, to obtain a stable and computationally efficient solution.

**NUREG/CR-5716:** MODEL VALIDATION AT THE LAS CRUCES TRENCH SITE. HILLS,R.G. New Mexico State Univ., Las Cruces, NM. WIERENGA,P.J. Arizona, Univ. of, Tucson, AZ. June 1991. 95pp. 9107080234. 58307:134.

A series of dynamic field experiments have been performed at the Las Cruces Trench site to provide data to test deterministic and stochastic models for water flow and solute transport in spatially variable unsaturated soils. Two experiments were performed to provide support for model validation efforts during Phase I of INTRAVAL (an international effort towards validation of geosphere models for transport of radionuclides) and a third experiment is currently underway to support the INTRAVAL Phase II efforts. The third experiment utilized different boundary and initial conditions and additional chemical tracers. The data from the third experiment along with model predictions from several modeling groups will be used to test models for water flow and solute transport during infiltration and redistribution. This report summarizes the Las Cruces Trench Site model validation efforts and presents the INTRAVAL Phase II validation plans. The Phase II validation strategy is discussed in detail.

**NUREG/CR-5717:** PACKAGING SUPPLIER INSPECTION GUIDE. STROMBERG,H.M.; GREGG,R.E.; KIDO,C.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1991. 53pp. 9107010119. EGG-2641. 58274:001.

This document is a guide for conducting quality assurance inspections of transportation packaging suppliers, where suppliers are defined as designers, fabricators, distributors, users, or owners of transportation packaging. This document can be used during an inspection to determine regulatory compliance within the requirements of 10 Code of Federal Regulations, Part 71, Subpart H (10 CFR 71.101-71.135). The guidance described in this document provides a framework for an inspection. It provides the inspector with the flexibility to adapt the methods and concepts presented here to meet the needs of the particular facility being inspected. The guide was developed to ensure a structured and consistent approach for inspections. The method treats each activity at a supplier facility as a separate entity (or functional element), and combines the activities within the framework of an "inspection tree." The method separates each functional element into several areas of performance and then identifies guidelines, based on regulatory requirements, to be used to qualitatively rate each area. This document was developed to serve as a field manual to facilitate the work of inspectors.

**NUREG/CR-5721:** VIDEO SYSTEMS FOR ALARM ASSESSMENT. GREENWOLL,D.A.; MATTER,J.C. Sandia National Laboratories. EBEL,P.E. BE, Inc. September 1991. 82pp. 9110100260. SAND91-0947. 59351:132.

The purpose of this report is to present technical information that should be useful to NRC licensees in designing closed-circuit television systems for video alarm assessment. There is a section on each of the major components in a video system: camera, lens, lighting, transmission, synchronization, switcher, monitor, and recorder. Each section includes information on component selection, procurement, installation, test, and maintenance. Considerations for system integration of the components are contained in each section. System emphasis is focused on perimeter intrusion detection and assessment systems. A glossary of video terms is included.

**NUREG/CR-5722:** INTERIOR INTRUSION DETECTION SYSTEMS. RODRIQUEZ,J.R.; MATTER,J.C. Sandia National Laboratories. DRY,B. BE, Inc. October 1991. 105pp. 9201060276. SAND91-0948. 60205:012.

The purpose of this report is to present technical information that should be useful to NRC licensees in designing interior intrusion detection systems. Interior intrusion sensors are discussed according to their primary applications: boundary-penetration detection, volumetric detection, and point protection. Information necessary for implementation of an effective interior intrusion detection system is presented, including principles of operation, performance characteristics, and guidelines for design, procurement, installation, testing, and maintenance. A glossary of sensor terms is included.

**NUREG/CR-5723:** SECURITY SYSTEM SIGNAL SUPERVISION. CHRITTON,M.R. BE, Inc. MATTER,J.C. Sandia National Laboratories. September 1991. 39pp. 9110100264. SAND91-0949. 59351:211.

The purpose of this report is to present technical information that should be useful to NRC licensees for understanding and applying line supervision techniques to security communication links. A review of security communication links is followed by detailed discussions of link physical protection and DC/AC static supervision and dynamic supervision techniques. Material is also presented on security for atmospheric transmission and video line supervision. A glossary of security communication line supervision terms is appended.

**NUREG/CR-5727:** CHLORIDE ION DIFFUSION IN LOW WATER-TO-SOLID CEMENT PASTES. CLIFTON,J.R.; KNAB,L.I.; GARBOCZI,E.J.; et al. National Institute of Standards & Technology (formerly National Bureau of Standa. June 1991. 31pp. 9107080248. NISTIR 4549. 58308:165.

Diffusion coefficients of 0.3 water to solids ratio (w/s) hydrated portland cement paste specimens were measured using a conventional diffusion cell. Specimens were made from both ASTM Type I and Type II portland cements and blends containing mineral admixtures (fly ash, granulated blastfurnace slag, or silica fume). The average diffusion coefficient for the portland cement paste specimens was 14x10(-13) m(2)/s. The diffusion coefficients for the specimens containing mineral admixtures were much more variable than those for the portland cement paste specimens. A probable cause of the variability in the test results was the presence of cracks observed in the test specimens. The effects of the depth of concrete cover over reinforcing steel and of the chloride ion diffusion coefficient on the service life of reinforced concrete exposed to chloride ions were predicted based on a diffusion model. Based on the model, the effect of the cover was shown to be proportional to the square of the cover depth. A 10-fold decrease in the diffusion coefficient of concrete was predicted to result in a 10- fold increase in the predicted service life. Based on the results of the present study, it is recommended that a new chloride diffusivity test should be developed which is applicable to concrete. candidate test method is proposed.

**NUREG/CR-5728:** EXPERIMENTS TO INVESTIGATE THE EFFECT OF FLIGHT PATH ON DIRECT CONTAINMENT HEATING (DCH) IN THE SURTSEY TEST FACILITY.The Limited Flight Path (LFP) Tests. ALLEN,M.D.; PILCH,M.; NICHOLS,R.T.; et al. Sandia National Laboratories. October 1991. 110pp. 9111110282. SAND91-1105. 59576:057.

The goal of the Limited Flight Path (LFP) test series was to investigate the effect of reactor subcompartment flight path length on direct containment heating (DCH). The test series consisted of eight experiments with nominal flight paths of 1, 2, or 8 m. A thermitically generated mixture of iron, chromium, and alumina simulated the corium melt of a severe reactor accident.

After thermite ignition, superheated steam forcibly ejected the molten debris into a 1:10 linear scale model of a dry reactor cavity. The blowdown steam entrained the molten debris and dispersed it into the Surtsey vessel. The vessel pressure, gas temperature, debris temperature, hydrogen produced by steam/metal reactions, debris velocity, mass dispersed into the Surtsey vessel, and debris particle size were measured to each experiment. The measured peak pressure for each experiment was normalized by the total amount of energy introduced into the Surtsey vessel; the normalized pressures increased with lengthened flight path. The debris temperature at the cavity exit was about 2320 K. Gas grab samples indicated that steam in the cavity reacted rapidly to form hydrogen, so the driving gas was a mixture of steam and hydrogen. In these experiments approximately 70% of the steam driving gas was converted to hydrogen. The total amount of hydrogen produced was a weak function of the total debris mass dispersed into the Surtsey vessel, indicating that most of the steam/metal reactions occurred in the reactor cavity.

**NUREG/CR-5729:** MULTIVARIABLE MODELING OF PRESSURE VESSEL AND PIPING J-R DATA. EASON,E.D.; WRIGHT,J.E.; NELSON,E.E. Modeling & Computer Services. May 1991. 118pp. 9106120180. MCS 910401. 58063:227.

Multivariable models were developed for predicting J-R curves from available data, such as material chemistry, radiation exposure, temperature, and Charpy V-notch energy. The present work involved collection of public test data, application of advanced pattern recognition tools, and calibration of improved multivariable models. Separate models were fitted for different material groups, including RPV welds, Linde 80 welds, RPV base metals, piping welds, piping base metals, and the combined database. Three different types of models were developed, involving different combinations of variables that might be available for applications: a Charpy model, a preirradiation Charpy model, and a copper-fluence model. In general, the best results were obtained with the preirradiation Charpy model. The copper-fluence model is recommended only if Charpy data are unavailable, and then only for Linde 80 welds. Relatively good fits were obtained, capable of predicting the values of J for pressure vessel steels to within a standard deviation of 13-18% over the range of test data. The models were qualified for predictive purposes by demonstrating their ability to predict validation data not used for fitting.

**NUREG/CR-5732 DRF FC:** IODINE CHEMICAL FORMS IN LWR SEVERE ACCIDENTS.Draft Report For Comment. BEAHM,E.C.; WEBER,C.F.; KRESS,T.S. Oak Ridge National Laboratory. July 1991. 104pp. 9107220272. ORNL/TM-11861. 58490:001.

Calculated data from seven severe accident sequences in light water reactor plants were used to assess the chemical forms of iodine in containment. In most of the calculations for the seven sequences, iodine entering containment from the reactor coolant system was almost entirely in the form of CsI with very small contributions of I or HI. The largest fraction of iodine in forms other than CsI was a total of 3.2% as I plus HI. Within the containment, the CsI will deposit onto walls and other surfaces, as well as in water pools, largely in the form of iodide (I-). The radiation-induced conversion of I- in water pools into I(2) is strongly dependent on pH. In systems where the pH was controlled above 7, little additional elemental iodine would be produced in the containment atmosphere. When the pH falls below 7, it may be assumed that it is not being controlled and large fractions of iodine as I(2) within the containment atmosphere may be produced.

**NUREG/CR-5734:** RECOMMENDATIONS TO THE NRC ON ACCEPTABLE STANDARD FORMAT AND CONTENT FOR THE FUNDAMENTAL NUCLEAR MATERIAL CONTROL (FNMC) PLAN REQUIRED FOR LOW-ENRICHED URANIUM ENRICHMENT FACILITIES. MORAN,B.W.; BELEW,W.L. Oak Ridge K-25 Site. HAMMOND,G.A.; et al. 21st Century Industries, Inc. November 1991. 53pp. 9201060264. K/ITP-415. 60201:001.

A new section, 10 CFR 74.33, has been added to the material control and accounting (MC&A) requirements of 10 CFR Part 74. This new section pertains to U.S. Nuclear Regulatory Commission (NRC) licensed uranium enrichment facilities that are authorized to produce and to possess more than one effective kilogram of special nuclear material (SNM) of low strategic significance. The new section is patterned after 10 CFR 74.31, which pertains to NRC licensees (other than production or utilization facilities licensed pursuant to 10 CFR Part 50 and waste disposal facilities) that are authorized to possess and use more than one effective kilogram of unencapsulated SNM of low strategic significance. Because enrichment facilities have the potential capability of producing SNM of moderate strategic significance, and also strategic SNM, certain performance objectives and MC&A system capabilities are required in 10 CFR 74.33 in addition to those contained in 10 CFR 74.31. This document recommends to the NRC information that the licensee or applicant should provide in the fundamental nuclear material control plan. This document also describes methods that should be acceptable for compliance with the general performance objectives. While this document is intended to cover various uranium enrichment technologies, the primary focus at this time is gas centrifuge and gaseous diffusion.

**NUREG/CR-5737:** HYDROGEOLOGIC PERFORMANCE ASSESSMENT ANALYSIS OF THE COMMERCIAL LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY NEAR WEST VALLEY,NEW YORK. BERGERON,M.P.; SMOOT,J.L.; KEMNER,M.L.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. June 1991. 110pp. 9107010079. PNL-7688. 58248:196.

A hydrogeologic performance assessment of the commercial low-level waste site near West Valley, New York, was performed for two pathways: a shallow lateral pathway where trench water can potentially migrate laterally through fractured and weathered till to nearby streams and a deep vertical pathway where leachate can migrate downward through unweathered till and laterally offsite in a lacustrine unit. Along the shallow pathway, little physical site evidence is available to indicate what the degree of lateral migration can be. Past modeling showed that overflowing trench water would migrate laterally some distance before migrating downward into the unweathered till. If water did reach a nearby stream, calculations show that decay, adsorption, and stream dilution would reduce leachate concentration to acceptable levels. Within the deep pathway, tritium and carbon-14 were the only radionuclides released in any significant concentrations. Predicted tritium levels are well below regulatory limits; however, predicted peak C-14 concentrations, while meeting the 25 mrem/yr limit using the drinking-water-only exposure scenario, exceed the limit for full garden scenario. Site information on C-14 release rates and geochemical behavior has considerable uncertainty and would need to be more fully evaluated in a licensing situation.

**NUREG/CR-5740:** NEW GAMMA-RAY BUILDUP FACTOR DATA FOR POINT KERNEL CALCULATIONS: ANS-6.4.3 STANDARD REFERENCE DATA. TRUBEY,D.K. Oak Ridge National Laboratory. August 1991. 152pp. 9108290255. ORNL/RSIC-49/R1. 58911:248.

An American Nuclear Society Standards Committee Working Group, identified as ANS-6.4.3, has developed a set of evaluated gamma-ray isotropic point-source buildup factors and attenuation coefficients for a standard reference data base. The largely unpublished set of buildup factors calculated with the moments method has been evaluated by recalculating key values with Monte Carlo, integral transport, and discrete ordinates methods. Additional buildup factor data were obtained from PALLAS code results. Attention has been given to frequently-neglected processes such as bremsstrahlung and the effect of introducing a tissue phantom behind the shield. The proposed draft standard, provided as an appendix, contains data for a source energy range from 15 keV to 15 MeV and for 22 elements and 3 mixtures (water, air, and concrete). The buildup factor data are also represented as coefficients for the G-P fitting function. Tables giving correction factors for multiple scattering in tissue are also provided.

**NUREG/CR-5742 V01:** FEASIBILITY ASSESSMENT OF A RISK-BASED APPROACH TO TECHNICAL SPECIFICATIONS.Executive Summary. ATEFI,B.; GALLAGHER,D.W. Science Applications International Corp. (formerly Science Applications, Inc.). May 1991. 12pp. 9106180006. SAIC-90/1400. 58130:129.

The first phase of the assessment concentrates on (1) identification of selected risk-based approaches for improving current technical specifications, (2) appraisal of characteristics of each approach, including advantages and disadvantages, and (3) recommendation of one or more approaches that might result in improving current technical specification requirements. The second phase of the work concentrates on assessment of the feasibility of implementation of a pilot program to study detailed characteristics of the preferred approach. The real time risk-based approach was identified as the preferred approach to technical specifications for controlling plant operational risk. There do not appear to be any technical or institutional obstacles to prevent initiation of a pilot program to assess the characteristics and effectiveness of such an approach.

**NUREG/CR-5742 V02:** FEASIBILITY ASSESSMENT OF A RISK-BASED APPROACH TO TECHNICAL SPECIFICATIONS.Main Report. ATEFI,B.; GALLAGHER,D.W. Science Applications International Corp. (formerly Science Applications, Inc.). May 1991. 84pp. 9106180007. SAIC-90/1400. 58130:043.

See NUREG/CR-5742,V01 abstract.

**NUREG/CR-5743:** APPROACHES TO LARGE SCALE UNSATURATED FLOW IN HETEROGENEOUS, STRATIFIED, AND FRACTURED GEOLOGIC MEDIA. ABABOU,R. Center for Nuclear Waste Regulatory Analyses. August 1991. 160pp. 9110090320. 59332:045.

This report develops a broad review and assessment of quantitative modeling approaches and data requirements for large-scale subsurface flow in a radioactive waste geologic repository. The data review includes discussions of controlled field experiments, existing contamination sites, and site-specific hydrogeologic conditions at Yucca Mountain. Local-scale constitutive models for the unsaturated hydrodynamic properties of geologic media are analyzed, with particular emphasis on the effect of structural characteristics of the medium. The report further reviews and analyzes large-scale hydrogeologic spatial variability from aquifer data, unsaturated soil data, and fracture network data gathered from the literature. Finally, various modeling strategies toward large-scale flow simulations are assessed, including direct high-resolution simulation, and coarse-scale simulation based on auxiliary hydrodynamic models such as single equivalent continuum and dual-porosity continuum. The roles of anisotropy, fracturing, and broad-band spatial variability are emphasized.

**NUREG/CR-5748:** RADIATION EMBRITTLEMENT OF THE NEUTRON SHIELD TANK FROM THE SHIPPINGPORT REACTOR. CHOPRA,O.K.; SHACK,W.J. Argonne National Laboratory. ROSINSKI,S. Sandia National Laboratories. October 1991. 47pp. 9111070090. ANL-91/23. 59549:025.

The irradiation embrittlement of Shippingport neutron shield tank (NST) material (A212-B) has been characterized. Irradiation increases the Charpy transition temperature (CTT) by 23-28 degrees C (41-50 degrees F) and decreases the upper-shelf energy. The shift in CTT is not as severe as that observed in high-flux isotope reactor (HFIR) surveillance samples. However, the actual value of the CTT is higher than that for the HFIR data. The increase in yield stress is 51 MPa (7.4 ksi), which is comparable to HFIR data. The NST material is weaker in the transverse than in the longitudinal orientation. Some effects of position across the thickness of the wall are also observed; the CTT shift is slightly greater for specimens from the inner region of the wall. Annealing studies indicate complete recovery from embrittlement after 1 h at 400 degrees C (752 degrees F). Although the weld metal is significantly tougher than the base metal, the shifts in CTT are comparable. The shifts in CTT for the Shippingport NST are consistent with the test and Army reactor data for irradiations at <232 degrees C (<450 degrees F) and show very good agreement with the results for HFIR A212-B steel irradiated in the Oak Ridge Research Reactor (ORR). The effects of irradiation temperature, fluence rate, and neutron flux spectrum are discussed. The results indicate that fluence rate has no effect on radiation embrittlement at rates as low as 2 x 10(8) n/cm(2)-s and at the low operating temperatures of the Shippingport NST, i.e., 55 degrees C (130 degrees F). This suggests that the accelerated embrittlement of HFIR surveillance samples is most likely due to the relatively higher proportion of thermal neutrons in the HFIR spectrum compared to that for the test reactors.

**NUREG/CR-5749:** TECTONIC DEFORMATION REVEALED IN BALDCYPRESS TREES AT REELFOOT LAKE,TENNESSEE. VANARSDALE,R.; STAHLE,D.; CLEAVELAND,M. Arkansas, Univ. of, Fayetteville, AR. July 1991. 19pp. 9107220288. 58489:252.

Tree-ring analyses of baldcypress (Taxodium distichum) from Reelfoot Lake, Tennessee, support historical accounts that the lake formed during the great New Madrid earthquakes in 1811-1812. Due to ground subsidence and permanent flooding, all of the bottomland hardwood trees within the impounded area were killed. However, many water tolerant baldcypress survived, and hundreds of 200 to 800 years old baldcypress outline the positions of former stream channels drowned by the subsidence. Dendrochronological analyses of multiple cores from 21 baldcypress in the lake reveal several pronounced growth responses to the 1811-1812 earthquakes. These responses include a great surge in radial growth during the decade following the earthquakes and a permanent reduction in wood density beginning in 1812. These and other growth responses to the 1811-1812 earthquakes may allow us to determine if there have been other large earthquakes in the Reelfoot basin during the late Holocene and may help date the formation of other suspected sunk lands in the New Madrid seismic zone.

**NUREG/CR-5757:** VERIFICATION OF PIPING RESPONSE CALCULATION OF SMACS CODE WITH DATA FROM SEISMIC TESTING OF AN IN-PLANT PIPING SYSTEM. SRINIVASAN,M.G.; KOT,C.A.; HSIEH,B.J. Argonne National Laboratory. September 1991. 207pp. 9110090275. ANL-91/25. 59330:106.

The objective of this effort was to evaluate the piping analysis part of the SMACS code for estimating the response of realistic piping systems subjected to multiple independent support accelerations. Test data from the experiments on an in-plant piping system were used for this purpose. Two support configurations were selected for the evaluation: one a 'stiff' configuration containing both struts and snubbers, and the other, a more flexible configuration with no snubbers. Described are the analytical modeling, calculations, and results of the posttest simulation of two tests each for both support configurations. Almost all the calculated peak response quantities were smaller than the corresponding test measurements. However, pipe displacements and bending stresses were better estimated than the pipe accelerations and support forces. The discrepancies are mainly attributable to the inability of the linear analysis to model the nonlinear behavior of the piping system.

**NUREG/CR-5758 V01:** FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports,CY 1990. DURBIN,N.; MURPHY,S.; FLEMING,T.; et al. Battelle Human Affairs Research Centers. August 1991. 67pp. 9109050292. PNL-7736. 58988:230.

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 3, 1990, to June 30, 1990, and from July 1, 1990, to December 31, 1990. During CY 1990, licensees reported that they conducted 278,209 tests for the presence of illegal drugs and alcohol. Of these tests, 2,409 (.87%) were positive. Positive test results varied by category of test and category of worker. The majority of positive test results (1,548) were obtained through pre-access testing. Of tests conducted on workers having access to the protected area, there were 550 positive tests from random testing, and 214 positive tests from for-cause testing. Followup testing of workers who had previously tested positive resulted in 65 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-5760:** REPORT ON ANNEALING OF THE NOVOVORONEZH UNIT 3 REACTOR VESSEL IN THE USSR. COLE,N.M.; FRIDERICHS,T. MPR Associates, Inc. July 1991. 78pp. 9108130307. MPR-1230. 58765:308.

A U.S. delegation attended the thermal annealing operation of the Novovoronezh Unit 3 reactor vessel in the USSR to evaluate the Soviet reactor vessel annealing technology and to determine its applicability to PWR reactors in the U.S. Operations observed and described in this report include reactor vessel sample cutting, preparations for annealing, installation of annealing apparatus, and initial heatup of the reactor vessel. The annealing operation witnessed has been developed to a routine operation and appears applicable to U.S. PWRs. Key areas requiring further work to confirm applicability to U.S reactors are discussed.

**NUREG/CR-5761:** AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE SALEM NUCLEAR POWER PLANT. PUGH,R.; GORE,B.F.; VO,T.V. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1991. 34pp. 9109050280. PNL-7518. 58988:336.

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. Salem was selected as the fifth 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 Salem plant.

**NUREG/CR-5763:** AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE CALLAWAY NUCLEAR POWER PLANT. MOFFITT,N.E.; GORE,B.F.; VO,T.V. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1991. 34pp. 9109050285. PNL-7725. 58988:300.

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. Callaway was selected as the eleventh 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 Callaway plant.

**NUREG/CR-5764:** AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE GINNA NUCLEAR POWER PLANT. PUGH,R.; GORE,B.F.; VO,T.V.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1991. 36pp. 91101002[illegible]. PNL-7594. 59336:307.

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. Ginna was selected as the eighth 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 Ginna plant.

**NUREG/CR-5765:** SPARC-90: A CODE FOR CALCULATING FISSION PRODUCT CAPTURE IN SUPPRESSION POOLS. OWCZARSKI,P.C.; BURK,K.W. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1991. 94pp. 9201060084. PNL-7723. 60197:001.

This report describes the technical bases and use of two, updated versions of a computer code initially developed to serve as a tool for calculating aerosol particle retention in boiling water reactor pressure suppression pools during severe accidents, SPARC-87 and SPARC-90. The most recent version is SPARC-90. The initial or prototype version (Owczarski, Postma, and Schreck 1985) was improved to include the following: rigorous treatment of local particle deposition velocities on the surface of oblate spherical bubbles, new correlations for hydrodynamic behavior of bubble swarms, models for aerosol particle growth, both mechanistic and empirical models for vent exit region scrubbing, specific models for hydrodynamics of bubble breakup at various vent types, and models for capture of vapor iodine species. A complete user's guide is provided for SPARC-90 (along with SPARC-87). A code description, code operating instructions, partial code listing, examples of the use of SPARC-90, and summaries of experimental data comparison studies also support the use of SPARC-90.

**NUREG/CR-5767:** THE BEHAVIOR OF SHALLOW FLAWS IN REACTOR PRESSURE VESSELS. ROLFE,S.T. Kansas, Univ. of, Lawrence, KS. * Oak Ridge National Laboratory. November 1991. 35pp. 9201060092. ORNLSUB90SH6401. 60197:084.

The objective of this report is to recommend those research investigations that are necessary to understand the phenomenon of shallow behavior as it affects fracture toughness so that the results can be used properly in the structural margin assessment of reactor pressure vessels (RPVs) with flaws. Preliminary test results of A 533 B steel show an elevated crack-tip opening displacement toughness similar to that observed for structural steels tested at the University of Kansas. Thus, the inherent resistance to fracture initiation of A 533 B steel with shallow flaws appears to be higher than that used in the current American Society of Mechanical Engineers design curves based on testing fracture-mechanics specimens with deep flaws. If this higher toughness of laboratory specimens with shallow flaws can be transferred to a higher resistance to failure in RPV design or analysis, then the actual margin of safety in nuclear vessels with shallow flaws would be greater than is currently assumed on the basis of deep-flaw test results. This report reviews those factors and makes recommendations of studies that are needed to assess the transferability of shallow-flaw toughness test results to the structural margin assessment of RPV with shallow flaws.

**NUREG/CR-5768:** ICE-CONDENSER AEROSOL TESTS. LIGOTKE,M.W.; ESCHBACH,E.J.; WINEGARDNER,W. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1991. 390pp. 9110100238. PNL-7765. 59334:117.

This report presents the results of an experimental investigation of aerosol particle transport and capture using a full-scale height and reduced-scale cross section test facility based on the design of the ice compartment of a pressurized water reactor (PWR) ice-condenser containment system. Results of 38 tests included thermal-hydraulic as well as aerosol particle data. Particle retention in the test section was greatly influenced by thermal-hydraulic and aerosol test parameters. Test-average decontamination factor (DF) ranged between 1.0 and 36 (retentions between ~0 and 97.2%). The measured test-average particle retentions for tests without and with ice and steam ranged between DF = 1.0 and 2.2 and DF = 2.4 and 36, respectively. In order of apparent importance, parameters that caused particle retention in the test section in the presence of ice were steam mole fraction (SMF), noncondensible gas flow rate (residence time), particle solubility, and inlet particle size. Ice-basket section noncondensible flows greater than 0.1 m(3)/s resulted in stable thermal stratification whereas flows less than 0.1 m(3)/s resulted in thermal behavior termed meandering with frequent temperature crossovers between flow channels.

**NUREG/CR-5771:** PROBABILITY AND CONSEQUENCES OF MISLOADING FUEL IN A PWR. DIAMOND,D.J.; HSU,C.-J.; MUBAYI,V. Brookhaven National Laboratory. August 1991. 74pp. 9110090313. BNL-NUREG-52294. 59331:331.

This report documents the results of a study into the frequency and consequences of misloading fresh fuel assemblies during the reloading of a pressurized water reactor. The consequences that were considered included: (i) loss of required shutdown margin, (ii) inadvertent criticality, and (iii) worker exposure within the plant given inadvertent criticality. Neutronic calculations were performed for different patterns of fresh fuel clustered together in a Combustion Engineering reactor. The fresh fuel considered had a high U-235 content and was assumed to be loaded without control element assemblies. The frequencies of misloading fresh fuel assemblies into these clustered patterns were calculated taking into account operator errors and equipment malfunctions that could occur during an offload/reload sequence. The study has improved our understanding of how difficult it is to misload fuel and has quantified the loss of shutdown margin and the frequency of occurrence for specific misloadings as well as the doses that might result from an inadvertent criticality.

**NUREG/CR-5773:** SELECTION OF MODELS TO CALCULATE THE LLW SOURCE TERM. SULLIVAN,T.M. Brookhaven National Laboratory. October 1991. 81pp. 9111070095. BNL-NUREG-52295. 59549:073.

Performance assessment of a LLW disposal facility begins with an estimation of the rate at which radionuclides migrate out of the facility (i.e., source term). The focus of his 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). In turn, many of these physical processes are influenced by the design of the disposal facility (e.g., infiltration of water). The complexity of the problem and the absence of appropriate data prevent development of an entirely mechanistic representation of radionuclide release from a disposal facility. Typically, a number of assumptions, based on knowledge of the disposal system, are used to simplify the problem. This document provides a brief overview of disposal practices and reviews existing source term models as background for selecting appropriate models for estimating the source term. The selection rationale and the mathematical details of the models are presented. Finally, guidance is presented for combining the inventory data with appropriate mechanisms describing release from the disposal facility.

**NUREG/CR-5777:** GLOBAL POSITIONING SYSTEM MEASUREMENTS OVER A STRAIN MONITORING NETWORK IN THE EASTERN TWO-THIRDS OF THE UNITED STATES. STRANGE,W.E. Commerce, Dept. of, National Oceanic & Atmospheric Administration. September 1991. 34pp. 9110080418. 59313:310.

A 45-station geodetic network was established in 1987 using global positioning system (GPS) technology to provide a means of monitoring strain and deformation in the central and eastern United States. Reduction of the initial epoch data showed that accuracies of 1 to 3 cm can be achieved for horizontal position, provided sufficient observations are available and there are four or more fiducial stations whose positions are known a priori, for example from Very Long Baseline Interferometry measurements. Accuracies obtained provide the ability to determine strain at the 1:10(7) to 1:10(8) level. Vertical positions are less accurate because of problems in modeling refraction and are determined at the 5 to 7 cm level. It is planned to remeasure this network at regular intervals in the coming years to place bounds on the strain occurring in the central and eastern United States. This network is also expected to serve as a reference network for more detailed monitoring networks in areas of high risk such as the New Madrid area. Future measurements are expected to provide more accurate results because of increased numbers of GPS satellites available and improved computation software. The improved software will also allow future upgrading of the accuracy of the 1987 observations.

**NUREG/CR-5778 V01:** NEW YORK/NEW JERSEY REGIONAL SEISMIC NETWORK.Annual Report For April 1989 - March 1990. SEEBER,L.; SIMPSON,D.; JOHNSON,D.; et al. Lamont-Doherty Geological Observatory. September 1991. 50pp. 9110080413. 59311:213.

Lamont-Doherty Geological Observatory (L-DGO) continued operating a 31-station seismic network covering parts of New York and New Jersey. The network is being transformed into sub-networks with stations radio telemetered to "smart" recording stations. The sub-network approach is capable of providing improved data at reduced cost. The major research effort during the period of this report was centered about the Saguenay earthquake sequence in Quebec. L-DGO collaborated with the Canadian Geologic Survey in monitoring aftershocks with temporary local stations. Analysis of data from the 1985 Ardsley earthquake in Westchester county continued with a Green's function deconvolution approach to resolve the dimensions of the rupture of the main shock (Mb=4.0) and of the largest aftershock (Mb=3.0). The results corroborate the 1/2-1 km diameter inferred for the rupture and suggest that the segmentation of the Dobbs Ferry fault and of similar faults in the Manhattan Prong may be controlling the size of historic earthquakes in the New York City region. Finally, a portable seismograph survey was carried out in Palco, Kansas, which showed clearly that seismicity at Palco was induced.

**NUREG/CR-5780:** SUMMARY OF A WORKSHOP ON SEVERE ACCIDENT MANAGEMENT FOR BWRS. KASTENBERG,W.E.; APOSTOLAKIS,G.; JAE,M.; et al. California, Univ. of, Los Angeles, CA. November 1991. 62pp. 9201060094. 60197:121.

Severe accident management can be defined as the use of existing and/or alternative resources, systems and actions to prevent or mitigate a core-melt accident. For each accident sequence and each combination of 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 instrument behavior during an accident. During the period September 26-28, 1990, a workshop was held at the University of California, Los Angeles, to address these uncertainties for Boiling Water Reactors. This report contains a summary of the workshop proceedings.

**NUREG/CR-5781:** SUMMARY OF A WORKSHOP ON SEVERE ACCIDENT MANAGEMENT FOR PWRS. KASTENBERG,W.E.; APOSTOLAKIS,G.; JAE,M.; et al. California, Univ. of, Los Angeles, CA. November 1991. 61pp. 9201060242. 60195:168.

Severe accident management can be defined as the use of existing and/or alternative resources, systems and actions to prevent or mitigate a core-melt accident. For each accident sequence and each combination of strategy, there may be several options available to the operator; and each involves phenomenological and operational considerations regarding uncertainty. Operational uncertainty includes operator, system and instrument behavior during severe accidents. During the period May 15-17, 1990, a workshop was held at the University of California, Los Angeles, to address these uncertainties for pressurized water reactors. This report contains a summary of the workshop proceedings.

**NUREG/CR-5784:** FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.A Review Of The First Year Of Program Performance And An Update Of The Technical Issues. DURBIN,N.; MOORE,C.; GRANT,T.; et al. Battelle Human Affairs Research Centers. September 1991. 199pp. 9110100248. PNL-7795. 59335:239.

This report presents an overview of the NRC licensees' implementation of the FFD program during the first full year of the program's operation and provides new information on a variety of FFD technical issues. The purpose of this document is to contribute to appropriate changes to the rule, to the inspection process, and to other NRC activities. It describes the characteristics of licensee programs, discusses the results of NRC inspections, updates technical information covered in previous reports, and identifies lessons learned during the first year. Overall, the experience of the first full year of licensees' FFD program operations indicates that licensees have functioning fitness-for-duty programs devoted to the NRC rule's performance objectives of achieving drug-free workplaces in which nuclear power plant personnel are not impaired as they perform their duties.

**NUREG/CR-5794:** GROUND-WATER FLOW AND TRANSPORT MODELING OF THE NRC-LICENSED WASTE DISPOSAL FACILITY, WEST VALLEY, NEW YORK. KOOL,J.B.; WU,Y.S. HydroGeoLogic, Inc. October 1991. 128pp. 9201060238. 60194:294.

This report describes a simulation study of groundwater flow and radionuclide transport from disposal pits at the NRC licensed waste disposal facility in West Valley, New York. A transient, precipitation driven, flow model of the near-surface fractured till layer and underlying unweathered till was developed and calibrated against observed inflow data into a recently constructed interceptor trench for the period March - May, 1990. The results suggest that lateral flow through the upper, fractured till layer may be more significant than indicated by previous, steady state flow modeling studies. A conclusive assessment of the actual magnitude of lateral flow through the fractured till could, however, not be made. A primary factor contributing to this uncertainty is the unknown contribution of vertical infiltration through the interceptor trench cap to the total trench inflow. The second part of the investigation involved simulation of the migration of Sr-90, Cs-137 and Pu-239 from the one of the fuel hull disposal pits. A first- order radionuclide leach rate with rate coefficient of 10(-6)/day was assumed to describe radionuclide release into the disposal pit. The simulations indicated that for wastes buried below the fractured till zone, no significant migration would occur. However, under the assumed conditions, significant lateral migration could occur for radionuclides present in the upper, fractured till zone.

**NUREG/CR-5795:** VALIDATION AND TESTING OF THE VAM2D COMPUTER CODE. KOOL,J.B.; WU,Y.S. HydroGeoLogic, Inc. October 1991. 120pp. 9201060183. 60207:315.

VAM2D is a two-dimensional, variably saturated flow and transport code, with applications for performance assessment of nuclear waste disposal. The studies presented in this report involve application of the VAM2D code to two diverse subsurface modeling problems. The first one involves modeling of infiltration and redistribution of water and solutes in an initially dry, heterogeneous field soil. This application involves detailed modeling over a relatively short, 9-month period. The second problem pertains to the application of VAM2D to the modeling of a waste disposal facility in a fractured clay, over much larger space and time scales and with particular emphasis on the applicability and reliability of using the equivalent porous medium approach for simulating flow and transport in fractured geologic media.

**NUREG/CR-5796:** STEAM GENERATOR OPERATING EXPERIENCE,UPDATE FOR 1989-1990. FRANK,L. Viking Systems International. December 1991. 124pp. 9201060109. 60198:146.

This report summarizes operational events and degradation mechanisms affecting pressurized water reactor steam generator integrity. It provides: (1) results of 1989 and 1990 steam generator inspections; (2) highlights prevalent problem areas; (3) improvements that have been made in nondestructive testing methods; (4) preventive measures; (5) repair techniques; and (6) replacement procedures. It describes the equipment of the three (3) major suppliers and discusses recent examinations of 76 plants. Major areas of concern are the steam generator degradation mechanisms that affect tube integrity or cause tube leakage and tube failure. These include: (1) intergranular attack; (2) intergranular stress corrosion cracking; (3) primary water stress corrosion cracking; (4) pitting; and (5) vibrational wear and fatigue. Also discussed are plugging, sleeving, heat treatment, peening, chemical cleaning and steam generator replacements. The current status of regulatory instruments and inspection guidelines for ensuring the steam generator integrity, is discussed with the highlights of steam generator research. New potential safety issues such as circumferential cracking and tube plug cracking are also discussed.

**NUREG/CR-5798:** PILOT PROGRAM TO ASSESS PROPOSED BASIC QUALITY ASSURANCE REQUIREMENTS IN THE MEDICAL USE OF BYPRODUCT MATERIAL. KAPLAN,E.; NELSON,K.; MEINHOLD,C.B. Brookhaven National Laboratory. October 1991. 73pp. 9110290323. BNL-NUREG-52303. 59455:273.

In January 1990, the Nuclear Regulatory Commission (NRC) proposed amendments to 10 CFR Part 35 that would require medical licensees using byproduct material to establish and implement a basic quality assurance program. A 60-day real-world trial of the proposed rules was initiated to obtain information beyond that generally found through standard public comment procedures. Volunteers from randomly selected institutions had opportunities to review the details of the proposed regulations and to implement these rules on a daily basis during the trial. The participating institutions were then asked to evaluate the proposed regulations based on their personal experiences. The pilot project sought to determine whether medical institutions could develop written quality assurance programs that would meet the eight performance-based objectives of proposed Section 35.35. It was found that licensees could develop acceptable QA programs under a performance- based approach, that most licensee programs did meet the proposed objectives, and that most written QA plans would require consultations with

NRC or Agreement State personnel before they would fully meet all objectives of proposed Section 35.35. This report describes the overall pilot program. The methodology used to select and assemble the group of participating licensees is presented.

**NUREG/CR-5808:** CALCULATION OF ABSORBED DOSES TO WATER POOLS IN SEVERE ACCIDENT SEQUENCES. WEBER,C.F. Oak Ridge National Laboratory. December 1991. 49pp. 9201090205. ORNL/TM-11970. 60243:266.

A methodology is presented for calculating the radiation dose to a water pool from the decay of uniformly distributed nuclides in that pool. Motivated by the need to accurately model radiolysis reactions of iodine, direct application is made to fission product sources dissolved or suspended in containment sumps or pools during a severe nuclear reactor accident. Two methods of calculating gamma absorption are discussed - one based on point-kernel integration and the other based on Monte Carlo techniques. Using least-squares minimization, the computed results are used to obtain a correlation that relates absorbed dose to source energy and surface-to-volume ratio of the pool. This correlation is applied to most relevant fission product nuclides and used to actually calculate transient sump dose rate in a pressurized-water- reactor severe accident sequence.

**NUREG/CR-5809 DRF FC:** AN INTEGRATED STRUCTURE AND SCALING METHODOLOGY FOR SEVERE ACCIDENT TECHNICAL ISSUE RESOLUTION.Draft Report For Comment. BOYACK,B.E.; HENRY,R.E.; MOODY,F.J.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). November 1991. 678pp. 9201060161. EGG-2659. 60205:117.

Recognizing the central importance of severe accident scaling issues, the United States Nuclear Regulatory Commission implemented a Severe Accident Scaling Methodology development program involving a lead laboratory contractor and a Technical Program Group to guide the development and to demonstrate its practicality via a challenging application. The Technical Program Group recognized that the Severe Accident Scaling Methodology was an integral part of a larger structure for technical issue resolution and, therefore, found the need to define and document this larger structure. The Integrated Structure for Technical Issue Resolution objectives and process are described in this document. The objectives of the Severe Accident Scaling Methodology are to (a) provide a scaling methodology that is systematic and practical, auditable and traceable, (b) provide the scaling rationale and similarity criteria, (c) provide a procedure for conducting comprehensive reviews of facility design, test conditions, and results, (d) ensure the prototypicality of the experimental data, and (e) quantify biases due to scale distortions or due to non-prototypical test conditions. The ability to provide similarity criteria that combine the system (top-down) and process (bottom-up) view points, is a key feature of the Severe Accident Scaling Methodology. This hierarchical, two- tiered scaling (H2TS) approach provides both sufficiency and efficiency. The Integrated Structure for Technical Issue Resolution and the Severe Accident Scaling Methodology have been tested and demonstrated, by their application to a postulated direct containment heating scenario. The Technical Program Group believes the results demonstrate that the methodology satisfies the stated objectives.

**NUREG/GR-0002:** CONTINUOUS COOLING THERMAL CYCLE EFFECTS ON SENSITIZATION IN STAINLESS STEEL. ATTERIDGE,D.G.; CEDENO,C.A. Oregon Graduate Institute of Science & Technology, Beaverton, OR. September 1991. 70pp. 9110110188. 59351:254.

Work for this study was directed towards quantifying sensitization development (defined as grain boundary chromium depletion) in high carbon Type 304 and 316 stainless steel (SS) subjected to linear heating to a given peak temperature followed by linear cooling through the sensitization development temperature range. The major variables investigated included: (1) heating rate; (2) peak temperature; (3) holding time at peak temperature; and (4) cooling rate. Change in sensitization was tracked using the electrochemical potentiokinetic reactivation (EPR) test. Continuous heating/cooling cycles were performed using a furnace or using a thermal cycle simulation machine (Gleeble). Sensitization was found to increase with increasing peak temperature until a "critical" peak temperature was reached. Sensitization was very low for all samples heated above this critical peak temperature. The critical peak temperature was 900 degrees C for high-carbon (0.06 wt%) 304 and varied from 950 to 1000 degrees C for high-carbon (0.06 wt%) 316 SS. Sensitization increased with decreasing cooling rate and appeared to decrease with increasing heating rate. The slowest heating rate used was equal to the fastest cooling rate tested. Results are discussed in terms of grain boundary chromium carbide nucleation and precipitation, and chromium depletion.

**NUREG/GR-0003:** EFFECT OF PRIOR DEFORMATION ON SENSITIZATION DEVELOPMENT IN STAINLESS STEEL DURING CONTINUOUS COOLING. SIMMONS,J.W.; ATTERIDGE,D.G.; BRUEMMER,S.M. Oregon Graduate Institute of Science & Technology, Beaverton, OR. September 1991. 109pp. 9110150279. 59362:172.

High-carbon Type 316 stainless steel (SS) specimens were subjected to linear continuous cooling in a computer-controlled Gleeble thermal simulator. The degree of sensitization (DOS) was quantitatively measured using the electrochemical potentiokinetic reactivation (EPR) test. Sensitization values for the thermal cycles employed in the investigation were predicted using Bruemmer's SSDOS sensitization prediction model. Prior deformation significantly enhanced the rate of DOS development in the Type 316 SS material. The DOS increased with increasing amounts of prior strain and decreasing cooling rates. Sensitization response was also sensitive to peak cycle temperatures. Continuous cooling sensitization development occurred primarily in the critical temperature range between about 900 and 750 degrees C. Peak cycle temperatures above 1000 retarded sensitization development during subsequent continuous cooling. Strain recovery at elevated temperatures played an important role in reducing the effectiveness of prior deformation in accelerating sensitization kinetics. Due to the effects of recovery, in certain cases, prior strain values of 20% were only as effective as 10% in increasing the rate of sensitization development. Limited transgranular carbide precipitation was observed in 20% prior strain samples but was not a significant factor in the present work. The SSDOS model consistently over predicted DOS development regardless of material condition.

**NUREG/GR-0006 DRF FC:** DEPOSITION: SOFTWARE TO CALCULATE PARTICLE PENETRATION THROUGH AEROSOL TRANSPORT LINES.Draft Report For Comment. ANAND,N.K.; MCFARLAND,A.R. Texas A&M Univ., College Station, TX. October 1991. 41pp. 9201060116. 60198:270.

In this report, models are presented for calculating aerosol particle penetration through straight tubes of arbitrary orientation, inlets, and elbows. An expression to calculate effective depositional velocities of particles on tube walls is derived. The concept of "maximum penetration" is introduced, which is the maximum possible penetration through a sampling line connecting any two points in a three-dimensional space. A procedure to predict optimum tube diameter for an existing transport line is developed. An interactive menu driven software entitled DEPOSITION has been developed to perform above said tasks. This code can either be used on a PC or on a mainframe. The use and illustration of the software is described in Appendix A of this report. A copy of the DEPOSITION software can be obtained from the Department of Energy's Energy Science and Technology Software Center, Oak Ridge, TN 37831-1020.

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NUREG/CR-5641: STUDY OF OPERATIONAL RISK-BASED CONFIGURATION CONTROL.

**VIETH,P.**
NUREG/CR-4599 V01 N1: SHORT CRACKS IN PIPING AND PIPING WELDS.Semiannual Report, March-September 1990.

**VO,T.V.**
NUREG/CR-4427: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE BYRON AND BRAIDWOOD NUCLEAR POWER PLANTS.
NUREG/CR-4469 V11: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semi-Annual Report, April-September 1989.
NUREG/CR-5467: RISK-BASED INSPECTION GUIDE FOR CRYSTAL RIVER UNIT 3 NUCLEAR POWER PLANT.
NUREG/CR-5761: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE SALEM NUCLEAR POWER PLANT.
NUREG/CR-5763: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE CALLAWAY NUCLEAR POWER PLANT.
NUREG/CR-5764: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE GINNA NUCLEAR POWER PLANT.

**VULIN,D.S.**
NUREG/CR-3469 V06: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.

**WAHI,K.**
NUREG/CR-3964 V02: TECHNIQUES FOR DETERMINING PROBABILITIES OF EVENTS AND PROCESSES AFFECTING THE PERFORMANCE OF GEOLOGIC REPOSITORIES.Suggested Approaches.

**WALSH,B.**
NUREG/CR-5606: A REVIEW OF THE SOUTH TEXAS PROJECT PROBABILISTIC SAFETY ANALYSIS FOR ACCIDENT FREQUENCY ESTIMATES AND CONTAINMENT BINNING.

**WALTON,J.C.**
NUREG/CR-5614: PERFORMANCE OF INTACT AND PARTIALLY DEGRADED CONCRETE BARRIERS IN LIMITING FLUID FLOW.

**WARD,L.W.**
NUREG/CR-5543: A SYSTEMATIC PROCESS FOR DEVELOPING AND ASSESSING ACCIDENT MANAGEMENT PLANS.

**WASHINGTON,K.E.**
NUREG/CR-5516: QUALITY ASSURANCE PROCEDURES FOR THE CONTAIN SEVERE REACTOR ACCIDENT COMPUTER CODE.
NUREG/CR-5715: REFERENCE MANUAL FOR THE CONTAIN 1.1 CODE FOR CONTAINMENT SEVERE ACCIDENT ANALYSIS.

**WASSEF,W.**
NUREG/CR-5561: ANALYSIS OF BELLOWS EXPANSION JOINTS IN THE SEQUOYAH CONTAINMENT.

**WATKINS,J.C.**
NUREG/CR-4063: AN INVESTIGATION OF CORE LIQUID LEVEL DEPRESSION IN SMALL BREAK LOSS-OF-COOLANT ACCIDENTS.
NUREG/CR-5558: GENERIC ISSUE 87: FLEXIBLE WEDGE GATE VALVE TEST PROGRAM. Phase II Results And Analysis.

**WATKINS,R.A.**
NUREG-1415 V04 N01: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report.April-September 1991.

**WEBB,E.**
NUREG/CR-3964 V02: TECHNIQUES FOR DETERMINING PROBABILITIES OF EVENTS AND PROCESSES AFFECTING THE PERFORMANCE OF GEOLOGIC REPOSITORIES.Suggested Approaches.

**WEBER,C.F.**
NUREG/CR-5732 DRF FC: IODINE CHEMICAL FORMS IN LWR SEVERE ACCIDENTS.Draft Report For Comment.
NUREG/CR-5608: CALCULATION OF ABSORBED DOSES TO WATER POOLS IN SEVERE ACCIDENT SEQUENCES.

**WEBSTER,C.S.**
NUREG/CR-5481: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-4.
NUREG/CR-5668: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-5.

**WEINSTEIN,E.**
NUREG-1441: LESSONS LEARNED FROM THE POST-EMERGENCY TABLETOP EXERCISE IN BATON ROUGE,LOUISIANA,ON AUGUST 28 AND SEPTEMBER 18, 1990.
NUREG-1442: POST-EMERGENCY RESPONSE RESOURCES GUIDE.Based On The Post-Emergency TABLETOP Exercise In Baton Rouge,Louisiana,On August 28 And September 18, 1990.

**WEISS,A.J.**
NUREG/CP-0114 V01: PROCEEDINGS OF THE EIGHTEENTH WATER REACTOR SAFETY INFORMATION MEETING.
NUREG/CP-0114 V02: PROCEEDINGS OF THE EIGHTEENTH WATER REACTOR SAFETY INFORMATION MEETING.
NUREG/CP-0114 V03: PROCEEDINGS OF THE EIGHTEENTH WATER REACTOR SAFETY INFORMATION MEETING.
NUREG/CP-0118: TRANSACTIONS OF THE NINETEENTH WATER REACTOR SAFETY INFORMATION MEETING.

**WESTRA,C.**
NUREG/CR-5756 V01: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports,CY 1990.

**WHEELER,T.A.**
NUREG/CR-5606: A REVIEW OF THE SOUTH TEXAS PROJECT PROBABILISTIC SAFETY ANALYSIS FOR ACCIDENT FREQUENCY ESTIMATES AND CONTAINMENT BINNING.

**WHITE,J.E.**
NUREG/CR-5648: TRANSPORT CALCULATIONS OF NEUTRON TRANSMISSION THROUGH STEEL USING ENDF/B-V,REVISED ENDF/B-V,AND ENDF/B-VI IRON EVALUATIONS.

**WICHNER,R.P.**
NUREG/CR-5647: FISSION PRODUCT PLATEOUT AND LIFTOFF IN THE MHTGR PRIMARY SYSTEM: A REVIEW.

**WIERENGA,P.J.**
NUREG/CR-5716: MODEL VALIDATION AT THE LAS CRUCES TRENCH SITE.

**WILKERSON,C.L.**
NUREG/CR-4670: RADIONUCLIDE DISTRIBUTIONS AND MIGRATION MECHANISMS AT SHALLOW LAND BURIAL SITES.Final Report Of PNL Research Investigations On The Distribution, Migration, And Containment Of Radionuclides At Maxey Flats, Kentucky.

**WILKOWSKI,G.M.**
NUREG/CR-4599 V01 N1: SHORT CRACKS IN PIPING AND PIPING WELDS.Semiannual Report, March-September 1990.

NUREG/CR-5128: EVALUATION AND REFINEMENT OF LEAK-RATE ESTIMATION MODELS.

**WILLIAMS,D.C.**
NUREG/CR-5630: PWR DRY CONTAINMENT PARAMETRIC STUDIES.

**WILLIAMS,M.L.**
NUREG/CR-5648: TRANSPORT CALCULATIONS OF NEUTRON TRANSMISSION THROUGH STEEL USING ENDF/B-V,REVISED ENDF/B-V,AND ENDF/B-VI IRON EVALUATIONS.

**WILSON,G.E.**
NUREG/CR-5809 DRF FC: AN INTEGRATED STRUCTURE AND SCALING METHODOLOGY FOR SEVERE ACCIDENT TECHNICAL ISSUE RESOLUTION.Draft Report For Comment.

**WILSON,J.N.**
NUREG-1368: PREAPPLICATION SAFETY EVALUATION REPORT FOR THE SODIUM ADVANCED FAST REACTOR (SAFR) LIQUID METAL REACTOR.

**WILSON,R.**
NUREG/CR-5784: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.A Review Of The First Year Of Program Performance And An Update Of The Technical Issues.

**WINEGARDNER,W.**
NUREG/CR-5768: ICE-CONDENSER AEROSOL TESTS.

**WOLFRAM,L.M.**
NUREG/CR-5520: PROCEDURES GUIDE FOR EXTRACTING AND LOADING PROBABILISTIC RISK ASSESSMENT DATA INTO MAR-D USING IRRAS 2.5.

**WOOD,R.S.**
NUREG-0327 R05: OWNERS OF NUCLEAR POWER PLANTS.

**WRIGHT,J.E.**
NUREG/CR-5729: MULTIVARIABLE MODELING OF PRESSURE VESSEL AND PIPING J-R DATA.

**WRIGHT,R.Q.**
NUREG/CR-5648: TRANSPORT CALCULATIONS OF NEUTRON TRANSMISSION THROUGH STEEL USING ENDF/B-V,REVISED ENDF/B-V,AND ENDF/B-VI IRON EVALUATIONS.

**WU,Y.-T.**
NUREG/CR-5639: UNCERTAINTY EVALUATION METHODS FOR WASTE PACKAGE PERFORMANCE ASSESSMENT.

**WU,Y.S.**
NUREG/CR-5352 R01: VAM2D - VARIABLY SATURATED ANALYSIS MODEL IN TWO DIMENSIONS. Version 5.2 With Hysteresis And Chain Decay Transport. Documentation And User's Guide.
NUREG/CR-5794: GROUND-WATER FLOW AND TRANSPORT MODELING OF THE NRC-LICENSED WASTE DISPOSAL FACILITY, WEST VALLEY, NEW YORK.
NUREG/CR-5795: VALIDATION AND TESTING OF THE VAM2D COMPUTER CODE.

**WULFF,W.**
NUREG/CR-5809 DRF FC: AN INTEGRATED STRUCTURE AND SCALING METHODOLOGY FOR SEVERE ACCIDENT TECHNICAL ISSUE RESOLUTION.Draft Report For Comment.

**WYNHOFF,N.C.**
NUREG/CR-5343: RADIONUCLIDE CHARACTERIZATION OF REACTOR DECOMMISSIONING WASTE AND SPENT FUEL ASSEMBLY HARDWARE.Progress Report.

**XING,L.**
NUREG/CR-5780: SUMMARY OF A WORKSHOP ON SEVERE ACCIDENT MANAGEMENT FOR BWRS.
NUREG/CR-5781: SUMMARY OF A WORKSHOP ON SEVERE ACCIDENT MANAGEMENT FOR PWRS.

**XIONG,L.X.**
NUREG/CR-5727: CHLORIDE ION DIFFUSION IN LOW WATER-TO-SOLID CEMENT PASTES.

**YAN,H.**
NUREG/CR-5423: THE PROBABILITY OF LINER FAILURE IN A MARK-I CONTAINMENT.

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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.

**EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA**

OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA, DIRECTOR

NUREG-0090 V13 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1990.

NUREG-0090 V13 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1990.

NUREG-0090 V14 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.January-March 1991.

NUREG-0090 V14 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1991.

NUREG-0090 V14 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1991.

NUREG-1022 R01 DR FC: EVENT REPORTING SYSTEMS 10 CFR 50.72 AND 50.73.Clarification Of NRC Systems And Guidelines for Reporting.Draft Report For Comment.

NUREG-1272 V05 N01: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA.1990 Annual Report - Power Reactors.

NUREG-1272 V05 N02: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA.1990 Annual Report - Nonreactors.

NUREG-1275 V06: OPERATING EXPERIENCE FEEDBACK REPORT - SOLENOID-OPERATED VALVE PROBLEMS.Commercial Power Reactors.

NUREG-1303 R01: INCIDENT INVESTIGATION MANUAL.

INCIDENT RESPONSE BRANCH

NUREG-1441: LESSONS LEARNED FROM THE POST-EMERGENCY TABLETOP EXERCISE IN BATON ROUGE,LOUISIANA,ON AUGUST 28 AND SEPTEMBER 18, 1990.

NUREG-1442: POST-EMERGENCY RESPONSE RESOURCES GUIDE.Based On The Post-Emergency TABLETOP Exercise in Baton Rouge,Louisiana,On August 28 And September 18, 1990.

**EDO - OFFICE OF INFORMATION RESOURCES MANAGEMENT & ARM (POST 861109)**

DIVISION OF COMPUTER & TELECOMMUNICATIONS SERVICES (POST 890205)

NUREG-0020 V15: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of December 31, 1990.(Gray Book I)

**EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS**

OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS

NUREG-0430 V10: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.July 1989 - June 1990.(Gray Book II)

DIVISION OF SAFEGUARDS & TRANSPORTATION (POST 870413)

NUREG-0383 V01 R14: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Report Of NRC Approved Packages.

NUREG-0383 V02 R14: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Certificates Of Compliance.

NUREG-0383 V03 R11: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Report Of NRC Approved Quality Assurance Programs For Radioactive Materials Packages.

NUREG-0525 R17: SAFEGUARDS SUMMARY EVENT LIST (SSEL).Pre-NRC Through December 31, 1990.

NUREG-0725 R07: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL.

NUREG-1321: TESTING STANDARDS FOR PHYSICAL SECURITY SYSTEMS AT CATEGORY I FUEL CYCLE FACILITIES.

NUREG-1322: ACCEPTANCE CRITERIA FOR THE EVALUATION OF CATEGORY I FUEL CYCLE FACILITY PHYSICAL SECURITY PLANS.

DIVISION OF HIGH-LEVEL WASTE MANAGEMENT (POST 870413)

NUREG-1439: STAFF TECHNICAL POSITION ON REGULATORY CONSIDERATIONS IN THE DESIGN AND CONSTRUCTION OF THE EXPLORATORY SHAFT FACILITY.

NUREG/CR-4735 V07: EVALUATION AND COMPILATION OF DOE WASTE PACKAGE TEST DATA. Biannual Report: February-July 1989.

DIVISION OF LOW-LEVEL WASTE MANAGEMENT & DECOMMISSIONING (POST 870413)

NUREG-1199 R02: STANDARD FORMAT AND CONTENT OF A LICENSE APPLICATION FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY.

NUREG-1200 R02: STANDARD REVIEW PLAN FOR THE REVIEW OF A LICENSE APPLICATION FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY.

NUREG-1293 R01: QUALITY ASSURANCE GUIDANCE FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY.

NUREG-1375 V02: SAFETY EVALUATION REVIEW OF THE PROTOTYPE LICENSE APPLICATION SAFETY ANALYSIS REPORT.Belowground Vault.

**U.S. NUCLEAR REGULATORY COMMISSION**

OFFICE OF THE GENERAL COUNSEL (POST 860701)

NUREG-0386 D05 R09: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission, Appeal Board And Licensing Board Decisions.July 1972 - September 1990.

NUREG-0386 D06: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission, Appeal Board And Licensing Decisions.July 1972 - December 1990.

NUREG-0980 V01 N01: NUCLEAR REGULATORY LEGISLATION.101st Congress.

NUREG-0980 V02 N01: NUCLEAR REGULATORY LEGISLATION.101st Congress.

OFFICE OF THE INSPECTOR GENERAL (POST 890417)

NUREG-1415 V03 N02: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report.October 1990 - March 1991.

NUREG-1415 V04 N01: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report.April-September 1991.

NRC - NO DETAILED AFFILIATION GIVEN

NUREG/CR-4063: AN INVESTIGATION OF CORE LIQUID LEVEL DEPRESSION IN SMALL BREAK LOSS-OF-COOLANT ACCIDENTS.

**EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)**

OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 860720)

NUREG-1266 V05: NRC SAFETY RESEARCH IN SUPPORT OF REGULATION - FY 1990.

NUREG-1369: PREAPPLICATION SAFETY EVALUATION REPORT FOR THE SODIUM ADVANCED FAST REACTOR (SAFR) LIQUID METAL REACTOR.

NUREG/CP-0037: PROCEEDINGS OF THE SEMINAR ON ASSESSMENT OF FRACTURE PREDICTION TECHNOLOGY: PIPING AND PRESSURE VESSELS.

NUREG/CP-0118: TRANSACTIONS OF THE NINETEENTH WATER REACTOR SAFETY INFORMATION MEETING.

DIVISION OF ENGINEERING (POST 870413)

NUREG-0975 V08: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING.Annual Report For FY 1990.

NUREG-1144 R02: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN.Status And Accomplishments.

NUREG-1377 R02: NRC RESEARCH PROGRAM ON PLANT AGING: LISTING AND SUMMARIES OF REPORTS ISSUED THROUGH JUNE 1991.

NUREG-1426 V01: COMPILATION OF REPORTS FROM RESEARCH SUPPORTED BY THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING. 1965 - 1990.

DIVISION OF REGULATORY APPLICATIONS (POST 870413)

NUREG-0713 V10: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES,1988.Twenty First Annual Report.

NUREG-0933 S12: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

NUREG-0933 S13: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

NUREG-1307 R02: REPORT ON WASTE BURIAL CHARGES.Escalation Of Decommissioning Waste Disposal Costs At Low-Level Waste Burial Facilities.

NUREG-1391: CHEMICAL TOXICITY OF URANIUM HEXAFLUORIDE COMPARED TO ACUTE EFFECTS OF RADIATION.Final Report.

NUREG-1400 DRFT FC: AIR SAMPLING IN THE WORKPLACE.Draft Report For Comment.

NUREG-1446: STANDARDS FOR PROTECTION AGAINST RADIATION - 10 CFR PART 20. A Comparison Of The Existing And Revised Rules.

WASTE MANAGEMENT BRANCH (POST 910830)

NUREG/CR-4918 V05: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Progress Report On Field Experiments At A Humid Region Site,Beltsville,Maryland.

DIVISION OF SAFETY ISSUE RESOLUTION (POST 880717)

NUREG-0933 S01-12: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

NUREG-1362: REGULATORY ANALYSIS FOR FINAL RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL.Final Report.

NUREG-1374: TECHNICAL FINDINGS RELATED TO GENERIC ISSUE 79.An Evaluation Of PWR Reactor Vessel Thermal Stress During Natural Convection Cooldown.

NUREG-1398: ENVIRONMENTAL ASSESSMENT FOR FINAL RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL.Final Report.

NUREG-1401 DRFT FC: REGULATORY ANALYSIS FOR GENERIC ISSUE 23: REACTOR COOLANT PUMP SEAL FAILURE.Draft Report For Comment.

NUREG-1407: PROCEDURAL AND SUBMITTAL GUIDANCE FOR INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR SEVERE ACCIDENT VULNERABILITIES.Final Report.

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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.

There were no NUREG/IA reports for 1991

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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.

**EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA**

OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA, DIRECTOR

- NUREG/CR-2000 V09N12: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of December 1990.
- NUREG/CR-2000 V10 N1: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of January 1991.
- NUREG/CR-2000 V10 N2: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of February 1991.
- NUREG/CR-2000 V10 N3: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of March 1991.
- NUREG/CR-2000 V10 N4: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of April 1991.
- NUREG/CR-2000 V10 N5: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of May 1991.
- NUREG/CR-2000 V10 N6: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of June 1991.
- NUREG/CR-2000 V10 N7: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of July 1991.
- NUREG/CR-2000 V10 N8: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of August 1991.
- NUREG/CR-2000 V10 N9: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of September 1991.
- NUREG/CR-2000 V10N10: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of October 1991.
- NUREG/CR-2000 V10N11: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of November 1991.

DIVISION OF SAFETY PROGRAMS (POST 870413)

- NUREG/CR-4674 V13: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1990 A STATUS REPORT.Main Report And Appendix A.
- NUREG/CR-4674 V14: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1990 A STATUS REPORT.Appendixes B And C.
- NUREG/CR-5456: ANALYSIS OF FLOW STRATIFICATION IN THE SURGE LINE OF THE COMANCHE PEAK REACTOR.

**EDO - OFFICE OF INFORMATION RESOURCES MANAGEMENT & ARM (POST 861109)**

OFFICE OF INFORMATION RESOURCES MANAGEMENT (POST 890205)

- NUREG/CR-2907 V09: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1988.

**EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS**

DIVISION OF SAFEGUARDS & TRANSPORTATION (POST 870413)

- NUREG/CR-5689: MEDICAL SCREENING REFERENCE MANUAL FOR SECURITY FORCE PERSONNEL AT FUEL CYCLE FACILITIES POSSESSING FORMULA QUANTITIES OF SPECIAL NUCLEAR MATERIALS.
- NUREG/CR-5690: PHYSICAL FITNESS TRAINING REFERENCE MANUAL FOR SECURITY FORCE PERSONNEL AT FUEL CYCLE FACILITIES POSSESSING FORMULA QUANTITIES OF SPECIAL NUCLEAR MATERIALS.
- NUREG/CR-5717: PACKAGING SUPPLIER INSPECTION GUIDE.
- NUREG/CR-5721: VIDEO SYSTEMS FOR ALARM ASSESSMENT.
- NUREG/CR-5722: INTERIOR INTRUSION DETECTION SYSTEMS.
- NUREG/CR-5723: SECURITY SYSTEM SIGNAL SUPERVISION.
- NUREG/CR-5734: RECOMMENDATIONS TO THE NRC ON ACCEPTABLE STANDARD FORMAT AND CONTENT FOR THE FUNDAMENTAL NUCLEAR MATERIAL CONTROL (FNMC) PLAN REQUIRED FOR LOW-ENRICHED URANIUM ENRICHMENT FACILITIES.

DIVISION OF HIGH-LEVEL WASTE MANAGEMENT (POST 870413)

- NUREG/CR-3964 V02: TECHNIQUES FOR DETERMINING PROBABILITIES OF EVENTS AND PROCESSES AFFECTING THE PERFORMANCE OF GEOLOGIC REPOSITORIES.Suggested Approaches.
- NUREG/CR-4735 V07: EVALUATION AND COMPILATION OF DOE WASTE PACKAGE TEST DATA. Biannual Report: February-July 1989.
- NUREG/CR-5522: A COMPARISON OF PARAMETER ESTIMATION AND SENSITIVITY ANALYSIS TECHNIQUES AND THEIR IMPACT ON THE UNCERTAINTY IN GROUND WATER FLOW MODEL PREDICTIONS.
- NUREG/CR-5537: APPROACHES FOR THE VALIDATION OF MODELS USED FOR PERFORMANCE ASSESSMENT OF HIGH-LEVEL NUCLEAR WASTE REPOSITORIES.
- NUREG/CR-5639: UNCERTAINTY EVALUATION METHODS FOR WASTE PACKAGE PERFORMANCE ASSESSMENT.

DIVISION OF LOW-LEVEL WASTE MANAGEMENT & DECOMMISSIONING (POST 870413)

- NUREG/CR-5432 V01: RECOMMENDATIONS TO THE NRC FOR SOIL COVER SYSTEMS OVER URANIUM MILL TAILINGS AND LOW-LEVEL RADIOACTIVE WASTES. Identification And Ranking Of Soils For Disposal Facility Covers.
- NUREG/CR-5432 V02: RECOMMENDATIONS TO THE NRC FOR SOIL COVER SYSTEMS OVER URANIUM MILL TAILINGS AND LOW-LEVEL RADIOACTIVE WASTES. Laboratory And Field Tests For Soil Covers.
- NUREG/CR-5432 V03: RECOMMENDATIONS TO THE NRC FOR SOIL COVER SYSTEMS OVER URANIUM MILL TAILINGS AND LOW-LEVEL RADIOACTIVE WASTES. Construction Methods And Guidance For Sealing Penetrations In Soil Covers.
- NUREG/CR-5539: A SELF-TEACHING CURRICULUM FOR THE NRC/SNL LOW-LEVEL WASTE PERFORMANCE ASSESSMENT METHODOLOGY.
- NUREG/CR-5713: A REVIEW OF ENVIRONMENTAL CONDITIONS AND PERFORMANCE OF THE COMMERCIAL LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY NEAR SHEFFIELD,ILLINOIS.
- NUREG/CR-5714: HYDROGEOLOGIC PERFORMANCE ASSESSMENT ANALYSIS OF THE LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY NEAR SHEFFIELD, ILLINOIS.
- NUREG/CR-5737: HYDROGEOLOGIC PERFORMANCE ASSESSMENT ANALYSIS OF THE COMMERCIAL LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY NEAR WEST VALLEY,NEW YORK.
- NUREG/CR-5773: SELECTION OF MODELS TO CALCULATE THE LLW SOURCE TERM.

**EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)**

OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 860720)

- NUREG/CR-5550: PASSIVE NONDESTRUCTIVE ASSAY OF NUCLEAR MATERIALS.

DIVISION OF ENGINEERING (POST 870413)

- NUREG/CR-3145 V09: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION. Annual Report,October 1989 - September 1990.
- NUREG/CR-4219 V07 N1: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For October 1989 - March 1990.
- NUREG/CR-4219 V07 N2: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For April-September 1990.
- NUREG/CR-4235: SELECTION OF SILICEOUS AGGREGATE FOR CONCRETE.
- NUREG/CR-4269: MODELS OF TRANSPORT PROCESSES IN CONCRETE.
- NUREG/CR-4295: BOND STRENGTH OF CEMENTITIOUS BOREHOLE PLUGS IN WELDED TUFF.
- NUREG/CR-4302 V02: AGING AND SERVICE WEAR OF CHECK VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.Aging Assessments And Monitoring Method Evaluations.

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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.

**MODELING & COMPUTER SERVICES**
NUREG/CR-5729: MULTIVARIABLE MODELING OF PRESSURE VESSEL AND PIPING J-R DATA.

**MPR ASSOCIATES, INC.**
NUREG/CR-5760: REPORT ON ANNEALING OF THE NOVOVORONEZH UNIT 3 REACTOR VESSEL IN THE USSR.

**NATIONAL INSTITUTE OF STANDARDS & TECHNOLOGY (FORMERLY NATIONAL BUREAU OF**
NUREG/CR-4235: SELECTION OF SILICEOUS AGGREGATE FOR CONCRETE.
NUREG/CR-4269: MODELS OF TRANSPORT PROCESSES IN CONCRETE.
NUREG/CR-4735 V07: EVALUATION AND COMPILATION OF DOE WASTE PACKAGE TEST DATA. Biannual Report: February-July 1989.
NUREG/CR-5711: ASSESSMENT OF UNCERTAINTIES IN MEASUREMENT OF PH IN HOSTILE ENVIRONMENTS CHARACTERISTIC OF NUCLEAR REPOSITORIES.
NUREG/CR-5727: CHLORIDE ION DIFFUSION IN LOW WATER-TO-SOLID CEMENT PASTES.

**NEW MEXICO STATE UNIV., LAS CRUCES, NM**
NUREG/CR-5716: MODEL VALIDATION AT THE LAS CRUCES TRENCH SITE.

**NEW MEXICO, UNIV. OF, ALBUQUERQUE, NM**
NUREG/CR-5660: STATIC AND SIMULATED SEISMIC TESTING OF THE TRG-7 THROUGH -16 SHEAR WALL STRUCTURES.

**OAK RIDGE K-25 SITE**
NUREG/CR-5734: RECOMMENDATIONS TO THE NRC ON ACCEPTABLE STANDARD FORMAT AND CONTENT FOR THE FUNDAMENTAL NUCLEAR MATERIAL CONTROL (FNMC) PLAN REQUIRED FOR LOW-ENRICHED URANIUM ENRICHMENT FACILITIES.

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NUREG/CR-2000 V10 N3: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of March 1991.
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NUREG/CR-2000 V10N11: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of November 1991.
NUREG/CR-4219 V07 N1: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For October 1989 - March 1990.
NUREG/CR-4219 V07 N2: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For April-September 1990.
NUREG/CR-4302 V02: AGING AND SERVICE WEAR OF CHECK VALVES USED IN ENGINEERED SAFETY-FEATURE SYSTEMS OF NUCLEAR POWER PLANTS.Aging Assessments And Monitoring Method Evaluations.
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NUREG/CR-5668: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-5.
NUREG/CR-5696: IRRADIATION EFFECTS ON CHARPY IMPACT AND TENSILE PROPERTIES OF LOW UPPER-SHELF WELDS,HSSI SERIES 2 AND 3.
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NUREG/GR-0002: CONTINUOUS COOLING THERMAL CYCLE EFFECTS ON SENSITIZATION IN STAINLESS STEEL.
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NUREG/CR-4551 V2R1P2: EVALUATION OF SEVERE ACCIDENT RISKS: QUANTIFICATION OF MAJOR INPUT PARAMETERS.Experts' Determination Of Containment Loads And Molten Core Containment Interaction Issues.
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NUREG/CR-4674 V14: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1990 A STATUS REPORT.Appendixes B And C.
NUREG/CR-4690 V01 R1: GENERIC COMMUNICATIONS INDEX.Listings Of Communications, 1971 - 1989.
NUREG/CR-5611: ISSUES AND APPROACHES FOR USING EQUIPMENT RELIABILITY ALERT LEVELS.
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NUREG/CR-5695: A PROCESS FOR RISK-FOCUSED MAINTENANCE.
NUREG/CR-5742 V01: FEASIBILITY ASSESSMENT OF A RISK-BASED APPROACH TO TECHNICAL SPECIFICATIONS.Executive Summary.
NUREG/CR-5742 V02: FEASIBILITY ASSESSMENT OF A RISK-BASED APPROACH TO TECHNICAL SPECIFICATIONS.Main Report.

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NUREG/CR-4893: TECHNICAL FINDINGS REPORT FOR GENERIC ISSUE 135.Steam Generator And Steam Line Overfill Issues.
NUREG/CR-5167: COST/BENEFIT ANALYSIS FOR GENERIC ISSUE 23: REACTOR COOLANT PUMP SEAL FAILURE.
NUREG/CR-5682: SPECIFIC TOPICS IN SEVERE ACCIDENT MANAGEMENT.

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NUREG/CR-5065: A SYSTEMATIC APPROACH TO REPETITIVE FAILURES.

**STANFORD UNIV., STANFORD, CA**

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**TECHNADYNE ENGINEERING CONSULTANTS, INC.**

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NUREG/CR-3964 V02: TECHNIQUES FOR DETERMINING PROBABILITIES OF EVENTS AND PROCESSES AFFECTING THE PERFORMANCE OF GEOLOGIC REPOSITORIES.Suggested Approaches.
NUREG/GR-0006 DRF FC: DEPOSITION: SOFTWARE TO CALCULATE PARTICLE PENETRATION THROUGH AEROSOL TRANSPORT LINES.Draft Report For Comment.

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NUREG/CR-5796: STEAM GENERATOR OPERATING EXPERIENCE,UPDATE FOR 1989-1990.

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NUREG/CR-4214 R1P2A1: HEALTH EFFECTS MODELS FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Modifications Of Models Resulting From Recent Reports On Health Effects Of Ionizing Radiation.Low LET Radiation.Part II: Scientific Bases For Health....

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## International Organization Index

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

There were no NUREG/IA reports for 1991

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# Licensed Facility Index

This index lists the facilities that were the subject of NRC staff or contractor reports. The facility names are arranged in alphabetical order. They are preceded by their Docket number and followed by the report number. If further information is needed, refer to the main citation by the NUREG number.

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This journal includes all formal reports in the NUREG series prepared by the NRC staff and contractors; proceedings of conferences and workshops; as well as 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.

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