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Vol. 15, No. 4

Regulatory and Technical Reports (Abstract Index Journal)

Annual Compilation for 1990

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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:

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

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

A detailed explanation of the entries precedes each index.

The bibliographic elements of the main citations are the following:

Staff Report

NUREG-0808: MARK II CONTAINMENT PROGRAM EVALUATION AND ACCEPTANCE CRITERIA. ANDERSON, C.J. Division of Safety Technology. August 1981. 90 pp. 8109140048. 09570:200.

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

Conference Report

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

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

Contractor Report

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

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

International Agreement Report

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

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

Availability of NRC Publications

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:

Superintendent of Documents
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NRC Report Codes

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

In addition to the NUREG and NUREG/CR codes, NUREG/CP is used for NRC-sponsored conference proceedings 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 V13 N11: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of October 30,1989.(Gray Book I) LOVELACE,W.H. Division of Computer & Telecommunications Services (Post 890205). January 1990. 543pp. 9002120335. 52579:130.

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

NUREG-0020 V13 N12: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of November 30,1989.(Gray Book I) LOVELACE,W.H. Division of Computer & Telecommunications Services (Post 890205). January 1990. 558pp. 9003070140. 52809:125.

See NUREG-0020,V13,N11 abstract.

NUREG-0020 V14 N01: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of December 31,1989.(Gray Book I) LOVELACE,W.H. Division of Computer & Telecommunications Services (Post 890205). February 1990. 543pp. 9003130117. 52885:090.

See NUREG-0020,V13,N11 abstract.

NUREG-0020 V14 N02: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of January 31,1990.(Gray Book I) LOVELACE,W.H. Division of Computer & Telecommunications Services (Post 890205). April 1990. 564pp. 9005040120. 53648:104.

See NUREG-0020,V13,N11 abstract.

NUREG-0020 V14 N03: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of February 28,1990.(Gray Book I) HARTFIELD,R.A. Division of Computer & Telecommunications Services (Post 890205). May 1990. 531pp. 9008070321. 54860:231.

See NUREG-0020,V13,N11 abstract.

NUREG-0040 V13 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1989.(White Book) * Division of Reactor Inspection & Safeguards (Post 870411). January 1990. 91pp. 9003070134. 52808:208.

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 1989 through December 1989.

NUREG-0040 V14 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1990. * Division of Reactor Inspection & Safeguards (Post 870411). July 1990. 213pp. 9008140499. 54924:276.

This periodical covers the results of inspections performed by NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from January 1990 through March 1990.

NUREG-0040 V14 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,April-June 1990.(White Book) * Division of Reactor Inspection & Safeguards (Post 870411). October 1990. 125pp. 9011090222. 55661:315.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from April 1990 through June 1990.

NUREG-0040 V14 N03: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,July-September 1990.(White Book) * Division of Reactor Inspection & Safeguards (Post 870411). November 1990. 226pp. 9011160225. 55755:032.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from July 1990 through September 1990.

NUREG-0090 V12 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1989. * Office for Analysis & Evaluation of Operational Data, Director. January 1990. 34pp. 9003070121. 52814:186.

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, 1989. For this reporting period, there were five abnormal occurrences. One abnormal occurrence took place at a licensed nuclear power plant and involved significant deficiencies associated with the containment recirculation sump at the Trojan facility. The other four abnormal occurrences took place under other NRC-issued licenses: the first involved a medical diagnostic misadministration; the second involved a medical therapy misadministration; the third involved a radiation overexposure of a radiographer; and the fourth involved a significant breakdown and careless disregard of the radiation safety program at three of a licensee's manufacturing facilities. The Agreement States reported no abnormal occurrences during the reporting period. The report also contains information that updates some previously reported abnormal occurrences.

NUREG-0090 V12 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1989. * Office for Analysis & Evaluation of Operational Data, Director. March 1990. 21pp. 9005040046. 53622:224.

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, 1989. For this reporting period, there were three abnormal occurrences, none involving a licensed nuclear power plant. Two of the abnormal occurrences took place under other NRC-issued licenses. The first involved a medical diagnostic misadministration and the second involved a medical therapy misadministration. The third abnormal occurrence was reported by an Agreement State (Louisiana) and involved an overexposure to an industrial radiographer. The report also contains information that updates a previously reported abnormal occurrence.

NUREG-0090 V13 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.January-March 1990. * Office for Analysis & Evaluation of Operational Data, Director. July 1990. 42pp. 9008160098. 54947:261.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health and safety and requires a quarterly report of such events to be made to Congress. This report covers the period January 1 through March 31, 1990. For this reporting period, there were 10 abnormal occurrences. One involved the loss of vital ac power with a subsequent reactor coolant system heat-up at the Vogtle Unit 1 power plant during shutdown. The event was investigated by an NRC Incident Investigation Team (IIT). The other nine abnormal occurrences involved nuclear material licensees and are described in detail under other NRC-issued licenses: eight of these involved medical therapy misadministrations; the other involved the receipt of an unshielded radioactive source at Amersham Corporation in Burlington, Massachusetts. The latter event was also investigated by an NRC IIT. No abnormal occurrences were reported by the Agreement States. The report also contains information that updates a previously reported abnormal occurrence.

NUREG-0090 V13 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1990. * Office for Analysis & Evaluation of Operational Data, Director. October 1990. 34pp. 9011200187. 55791:044.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health and safety and requires a quarterly report of such events to be made to Congress. This report covers the period April 1 through June 30, 1990. The report discusses six abnormal occurrences, none involving a nuclear power plant. There were five abnormal occurrences at NRC licensees: (1) deficiencies in brachytherapy program; (2) a radiation overexposure of a radiographer; (3) a medical diagnostic misadministration; (4) administration of I-131 to a lactating female with uptake by her infant; and (5) a medical therapy misadministration. An Agreement State (Arizona) reported an abnormal occurrence involving a medical diagnostic misadministration. The report also contains information that updates a previously reported abnormal occurrence.

NUREG-0304 V14 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1989. * Division of Freedom of Information & Publications Services (Post 890205). March 1990. 152pp. 9004030145. 53228:250.

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 V15 N01: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1990,January-March. * Division of Freedom of Information & Publications Services (Post 890205). May 1990. 53pp. 9006080185. 54074:156.

See NUREG-0304,V14,N04 abstract.

NUREG-0304 V15 N02: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Second Quarter 1990,April-June. * Division of Freedom of Information & Publications Services (Post 890205). August 1990. 49pp. 9010240391. 55495:277.

See NUREG-0304,V14,N04 abstract.

NUREG-0304 V15 N03: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Third Quarter 1990,July-September. * Division of Freedom of Information & Publications Services (Post 890205). December 1990. 44pp. 9012280235. 56257:334.

See NUREG-0304,V14,N04 abstract.

NUREG-0325 R13: U.S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS.April 1, 1990. * Ofc of Personnel (Post 870413). April 1990. 64pp. 9005070012. 53656:347.

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

NUREG-0325 R14: U.S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS.August 15, 1990. * Ofc of Personnel (Post 870413). September 1990. 62pp. 9010090054. 55333:027.

See NUREG-0325,R13 abstract.

NUREG-0383 V01 R13: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Report Of NRC Approved Packages. * Division of Safeguards & Transportation (Post 870413). October 1990. 475pp. 9011060134. 55624:341.

This directory contains a Report of NRC Approved Packages (Volume 1), Certificates of Compliance (Volume 2), and a Report of NRC Approved Quality Assurance Programs for Radioactive Materials Packages (Volume 3). The purpose of this directory is to make available a convenient source of information on Quality Assurance Programs and Packagings which have been approved by the U.S. Nuclear Regulatory Commission. Shipments of radioactive material utilizing these packagings must be in accordance with the provisions of 49 CFR 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 R13: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Certificates Of Compliance. * Division of Safeguards & Transportation (Post 870413). October 1990. 634pp. 9011060139. 55626:096.

See NUREG-0383,V01,R13 abstract.

NUREG-0383 V03 R10: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Report On NRC Approved Quality Assurance Programs For Radioactive Materials Packages. * Division of Safeguards & Transportation (Post 870413). October 1990. 177pp. 9011060128. 55624:164.
See NUREG-0383,V01,R13 abstract.

NUREG-0386 D05 R05: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission, Appeal Board And Licensing Board Decisions.July 1972 - September 1989. * Office of the General Counsel (Post 860701). March 1990. 660pp. 9004110232. 53351:319.
This Revision number 5 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 July 1, 1972 to September 30, 1989, interpreting the NRC's Rules of Practice in 10 CFR Part 2.

NUREG-0386 D05 R06: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission, Appeal Board And Licensing Board Decisions.July 1972 - December 1989. * Office of the General Counsel (Post 860701). June 1990. 672pp. 9007120220. 54473:203.
This Revision 6 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 July 1, 1972 to December 31, 1989, interpreting the NRC's Rules of Practice in 10 CFR Part 2.

NUREG-0386 D05 R07: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission, Appeal Board And Licensing Board Decisions.July 1972 - March 1990. * Office of the General Counsel (Post 860701). August 1990. 655pp. 9009120095. 55125:145.
This Revision Number 7 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 July 1, 1972 to March 31, 1990, interpreting the NRC's Rules of Practice in 10 CFR Part 2.

NUREG-0386 D05 R08: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission, Appeal Board And Licensing Board Decisions.July 1972 - June 1990. * Office of the General Counsel (Post 860701). November 1990. 682pp. 9012280169. 56226:027.
This Revision 8 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 July 1, 1972, to June 30, 1990, interpreting the NRC's Rules of Practice in 10 CFR Part 2.

NUREG-0430 V09 N02: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.July 1988 - June 1989.(Gray Book II) * Office of Nuclear Material Safety & Safeguards. April 1990. 18pp. 9005210145. 53861:345.
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 R16: SAFEGUARDS SUMMARY EVENT LIST (SSEL). * Division of Safeguards & Transportation (Post 870413). July 1990. 369pp. 9009070011. 55070:128.
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, 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 were obtained from official NRC reports.

NUREG-0540 V11 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. October 1-31,1989. * Division of Freedom of Information & Publications Services (Post 890205). January 1990. 326pp. 9002120142. 52581:253.
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 V11 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.NOVEMBER 1-30, 1989. * Division of Freedom of Information & Publications Services (Post 890205). March 1990. 335pp. 9004090177. 53309:236.
See NUREG-0540,V11,N10 abstract.

NUREG-0540 V11 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.DECEMBER 1-31,1989. * Division of Freedom of Information & Publications Services (Post 890205). April 1990. 230pp. 9005210149. 53887:204.
See NUREG-0540,V11,N10 abstract.

NUREG-0540 V12 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31,1990. * Division of Freedom of Information & Publications Services (Post 890205). April 1990. 332pp. 9005210255. 53889:146.
See NUREG-0540,V11,N10 abstract.

NUREG-0540 V12 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.February 1-28, 1990. * Division of Freedom of Information & Publications Services (Post 890205). June 1990. 304pp. 9006290112. 54324:008.
See NUREG-0540,V11,N10 abstract.

NUREG-0540 V12 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.March 1-31, 1990. * Division of Freedom of Information & Publications Services (Post 890205). July 1990. 453pp. 9008070384. 54847:012.
See NUREG-0540,V11,N10 abstract.

NUREG-0540 V12 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.April 1-30, 1990. * Division of Freedom of Information & Publications Services (Post 890205). July 1990. 322pp. 9008160107. 54949:016.
See NUREG-0540,V11,N10 abstract.

NUREG-0540 V12 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.May 1-31, 1990. * Division of Freedom of Information & Publications Services (Post 890205). July 1990. 414pp. 9008160102. 54947:322.
See NUREG-0540,V11,N10 abstract.

NUREG-0540 V12 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. June 1-30, 1990. * Division of Freedom of Information & Publications Services (Post 890205). August 1990. 333pp. 9009040041. 55050:227.
See NUREG-0540,V11,N10 abstract.

NUREG-0540 V12 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.July 1-31, 1990. * Division of Freedom of Information & Publications Services (Post 890205). September 1990. 364pp. 9009210224. 55232:198.

See NUREG-0540,V11,N10 abstract.

NUREG-0540 V12 N08: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. August 1-31,1990. * Division of Freedom of Information & Publications Services (Post 890205). October 1990. 319pp. 9010300245. 55557:001.

See NUREG-0540,V11,N10 abstract.

NUREG-0540 V12 N09: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.September 1-30,1990. * Division of Freedom of Information & Publications Services (Post 890205). November 1990. 254pp. 9011270230. 55855:109.

See NUREG-0540,V11,N10 abstract.

NUREG-0540 V12 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. October 1-31, 1990. * Division of Freedom of Information & Publications Services (Post 890205). December 1990. 266pp. 9012280250. 56269:270.

See NUREG-0540,V11,N10 abstract.

NUREG-0650 R01: PUBLISHING DOCUMENTS IN THE NUREG SERIES. * Division of Freedom of Information & Publications Services (Post 890205). December 1990. 150pp. 9101110201. 56344:148.

This report provides guidelines to staff, contractors, and grantees who prepare documents to be published in the NUREG series for the U.S. Nuclear Regulatory Commission (NRC). Adhering to these guidelines will improve the readability and promote the uniformity of NRC publications. In addition, adhering to these guidelines will ensure these authors comply with the NRC's publications policy and procedures. This report revises and supersedes NUREG-0650, "Technical Writing Style Guide," originally published in November 1979. For the convenience of users, it repeats information about NRC's preferred style for listing references, with minor changes, that was published in NUREG-1379, "NRC Editorial Style Guide." NUREG-1379 superseded Supplement 1 to NUREG-0650.

NUREG-0713 V09: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES 1987.Twentieth Annual Report. BROOKS,B.G. Division of Regulatory Applications (Post 870413). HAGEMEYER,D. Science Applications International Corp. (formerly Science Applications, Inc.). November 1990. 275pp. 9011270235. 55856:116.

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 1987. The bulk of the data presented in the report was obtained from annual radiation exposure reports submitted in accordance with the requirements of 10 CFR 20.407 and the technical specifications of nuclear power plants. Data on workers terminating their employment at certain NRC licensed facilities were obtained from reports submitted pursuant to 10 CFR 20.408. The 1987 annual reports submitted by about 460 licensees indicated that approximately 233,000 individuals were monitored, 211,000 of whom were monitored by nuclear power facilities. They incurred an average individual dose of 0.19 rem (cSv) and an average measurable dose of 0.39 rem (cSv). Termination radiation exposure reports were analyzed to reveal that about 86,300 individuals completed their employment with one or more of the 460 covered licensees during 1987. Some 81,600 of these individuals terminated from power reactor facilities, and about 8,700 of them were considered to be transient workers who received an average dose of 0.69 rem (cSv).

NUREG-0750 CI02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January 1,1980 Through December 31,1985. * Division of Freedom of Information & Publications Services (Post 890205). November 1989. 1,089pp. 9003160045. 52958:212.

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 V30 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-September 1989. * Division of Freedom of Information & Publications Services (Post 890205). February 1990. 50pp. 9004100439. 53350:238.

See NUREG-0750,CI02 abstract.

NUREG-0750 V30 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-December 1989. * Division of Freedom of Information & Publications Services (Post 890205). May 1990. 87pp. 9007120163. 54486:237.

See NUREG-0750,CI02 abstract.

NUREG-0750 V30 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1989.Pages 1-84. * Division of Freedom of Information & Publications Services (Post 890205). January 1990. 89pp. 9004030109. 53229:208.

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 V30 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1989. Pages 85-165. * Division of Freedom of Information & Publications Services (Post 890205). January 1990. 85pp. 9004030047. 53230:007.

See NUREG-0750,V30,N01 abstract.

NUREG-0750 V30 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1989. Pages 167-229. * Division of Freedom of Information & Publications Services (Post 890205). January 1990. 68pp. 9004030101. 53229:299.

See NUREG-0750,V30,N01 abstract.

NUREG-0750 V30 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1989. Pages 231-323. * Division of Freedom of Information & Publications Services (Post 890205). January 1990. 100PP. 9004030044. 53230:096.

See NUREG-0750,V30,N01 abstract.

NUREG-0750 V30 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1989. Pages 325-708. * Division of Freedom of Information & Publications Services (Post 890205). March 1990. 393pp. 9004100444. 53350:286.

See NUREG-0750,V30,N01 abstract.

NUREG-0750 V30 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1989. Pages 709-811. * Division of Freedom of Information & Publications Services (Post 890205). April 1990. 105pp. 9007120158. 54486:319.

See NUREG-0750,V30,N01 abstract.

NUREG-0750 V31 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January-March 1990. * Division of Freedom of Information & Publications Services (Post 890205). July 1990. 53pp. 9009170093. 55202:102.

See NUREG-0750,CI02 abstract.

NUREG-0750 V31 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January-June 1990. * Division of Freedom of Information & Publications Services (Post 890205). September 1990. 81pp. 9010230091. 55470:048.

See NUREG-0750,CI02 abstract.

NUREG-0750 V31 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1990. Pages 1-130. * Division of Freedom of Information & Publications Services (Post 890205). April 1990. 135pp. 9007120154. 54487:064.
See NUREG-0750,V30,N01 abstract.

NUREG-0750 V31 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1990. Pages 131-195. * Division of Freedom of Information & Publications Services (Post 890205). April 1990. 70pp. 9007120148. 54487:199.
See NUREG-0750,V30,N01 abstract.

NUREG-0750 V31 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1990.Pages 197-332. * Division of Freedom of Information & Publications Services (Post 890205). May 1990. 139pp. 9007120122. 54487:268.
See NUREG-0750,V30,N01 abstract.

NUREG-0750 V31 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1990.Pages 333-370. * Division of Freedom of Information & Publications Services (Post 890205). June 1990. 44pp. 9007120119. 54488:047.
See NUREG-0750,V30,N01 abstract.

NUREG-0750 V31 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1990.Pages 371-481. * Division of Freedom of Information & Publications Services (Post 890205). July 1990. 117pp. 9008070350. 54863:075.
See NUREG-0750,V30,N01 abstract.

NUREG-0750 V31 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1990.Pages 483-604. * Division of Freedom of Information & Publications Services (Post 890205). August 1990. 133pp. 9009040088. 55046:178.
See NUREG-0750,V30,N01 abstract.

NUREG-0750 V32 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-September 1990. * Division of Freedom of Information & Publications Services (Post 890205). December 1990. 36pp. 9101110235. 56342:204.
See NUREG-0750,CI02 abstract.

NUREG-0750 V32 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1990.Pages 1-55. * Division of Freedom of Information & Publications Services (Post 890205). August 1990. 62pp. 9009130045. 55194:041.
See NUREG-0750,V30,N01 abstract.

NUREG-0750 V32 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1990. Pages 56-128. * Division of Freedom of Information & Publications Services (Post 890205). October 1990. 80pp. 9010310170. 55579:263.
See NUREG-0750,V30,N01 abstract.

NUREG-0750 V32 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1990. Pages 129-199. * Division of Freedom of Information & Publications Services (Post 890205). November 1990. 77pp. 9011290035. 55880:026.
See NUREG-0750,V30,N01 abstract.

NUREG-0750 V32 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1990. Pages 201-331. * Division of Freedom of Information & Publications Services (Post 890205). December 1990. 140pp. 9101110238. 56342:064.
See NUREG-0750,V30,N01 abstract.

NUREG-0797 S22: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF COMANCHE PEAK STEAM ELECTRIC STATION,UNITS 1 AND 2.Docket Nos. 50-445 And 50-446.(Texas Utilities Electric Company,et al.) * Comanche Peak Project Division (890101-900602). January 1990. 369pp. 9003070163. 52814:220.

Supplement 22 to the Safety Evaluation Report related to the operation of the Comanche Peak Steam Electric Station, Units 1 and 2 (NUREG-0797), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Somervell County, Texas, approximately 40 miles southwest of Fort Worth, Texas. This supplement reports the status of certain issues that had not been resolved at the time of publication of the Safety Evaluation Report and Supplements 1, 2, 3, 4, 6, 12, and 21 to that report. This supplement also includes the evaluations for licensing items resolved since Supplement 21 was issued. Supplement 5 has been cancelled. Supplements 7 through 11 were limited to the staff evaluation of allegations investigated by the NRC Technical Review Team. Supplement 13 presented the staff's evaluation of the Comanche Peak Response Team (CPRT) Program Plan, which was formulated by the applicant to resolve various construction and design issues raised by sources external to the applicant. Supplements 14 through 20 presented the staff's evaluation of the applicant's Corrective Action Program and CPRT activities. Items identified in Supplements 7, 8, 9, 10, 11, 13, 14, and 15 through 20 are not included in this supplement, except to the extent that they affect the applicant's Final Safety Analysis Report.

NUREG-0797 S23: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF COMANCHE PEAK STEAM ELECTRIC STATION,UNITS 1 AND 2.Docket Nos. 50-445 And 50-446.(Texas Utilities Electric Company,et al.) * Comanche Peak Project Division (890101-900602). February 1990. 102pp. 9003070170. 52808:299.

Supplement 23 to the Safety Evaluation Report related to the operation of the Comanche Peak Steam Electric Station, Units 1 and 2 (NUREG-0797), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Somervell County, Texas, approximately 40 miles southwest of Fort Worth, Texas. This supplement reports the status of certain issues that had not been resolved at the time of publication of the Safety Evaluation Report and Supplements 1, 2, 3, 4, 6, 12, 21, and 22 to that report. This supplement also includes the evaluations for licensing items resolved since Supplement 22 was issued. Supplement 5 has been cancelled. Supplements 7 through 11 were limited to the staff evaluation of allegations investigated by the NRC Technical Review Team. Supplement 13 presented the staff's evaluation of the Comanche Peak Response Team (CPRT) Program Plan, which was fomulated by the applicant to resolve various construction and design issues raised by sources external to the applicant. Supplements 14 through 20 presented the staff's evaluation of the applicant's Corrective Action Program and CPRT activities. Items identified in Supplements 7, 8, 9, 10, 11, 13, and 15 through 20 are not included in this supplement, except to the extent that they affect the applicant's Final Safety Analysis Report.

NUREG-0797 S24: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF COMANCHE PEAK STEAM ELECTRIC STATION UNITS 1 AND 2.Docket Nos. 50-445 And 50-446.(Texas Utilities Electric Company,et al.) * Comanche Peak Project Division (890101-900602). April 1990. 174pp. 9005180159. 53861:137.

Supplement 24 to the Safety Evaluation Report related to the operation of the Comanche Peak Steam Electric Station, Units 1 and 2 (NUREG-0797), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Somervell County, Texas, approximately 40 miles southwest of Fort Worth, Texas. This supplement reports the status of certain issues that had not been resolved at the time of publication of the Safety Evaluation Report and Supplements 1, 2, 3, 4, 6, 12, 21, 22, and 23 to that report. This supplement also includes the evaluations for licensing items resolved since Supplement 23 was issued. Supplement 5 has been cancelled. Supplements 7 through 11 were limited to the staff evaluation of allegations investigated by the NRR Technical Review Team. Supplement 13 presented the staff's evaluation of the Comanche Peak Response Team (CPRT) Program Plan, which was formulated by the licensee to resolve various construction and design issues raised by sources external to the licensee. Supplements 14 through 20 presented the staff's evaluation of the licensee's Corrective Action Program and CPRT activities. Items identified in Supplements 7, 8, 9, 10, 11, 13, and 15 through 20 are not included in this supplement, except to the extent that they affect the licensee's Final Safety Analysis Report.

NUREG-0800 03.6.1 R2: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 2 To SRP Section 3.6.1,"Plant Design For Protection Against Postulated Piping Failures In Fluid Systems Outside... * Office of Nuclear Reactor Regulation, Director (Post 870411). October 1990. 29pp. 9010310161. 55629:063.

The revision of SRP Section 3.6.1 (Revision 2), Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment, eliminated jet impingement effects on essential equipment associated with the arbitrary one square foot break in the break exclusion (superpipe) zone of main steam and feedwater lines outside the containment, modified the background section of SPLB 3-1 to bring the background up to date, and clarified the staff's position on postulated single active failures concurrent with a postulated piping failure in one of two or more redundant trains of dual-purpose moderate-energy essential system as defined in BTP SLPB 3-1.

NUREG-0800 17.3 R00: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 0 To SRP Section 17.3, "Quality Assurance Program Description." * Division of Licensee Performance & Quality Evaluation (Post 870411). August 1990. 20pp. 9009070001. 55071:137.

SRP Section 17.3, "Quality Assurance Program Description," is a new section in Chapter 17. It puts in place a performance-oriented quality assurance program review plan that (1) minimizes the current fragmentation and overlap of the self-assessment function responsibilities, including safety committee activities, audits, and other independent assessments, (2) simplifies the format, clarifies the intent, and consolidates the text of the present SRP Sections 17.1 and 17.2, (3) places emphasis on management, performance/verification, and self-assessment, the three components of quality assurance, and (4) permits the use of up-to-date industry consensus standards.

NUREG-0837 V09 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. October-December 1989. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). March 1990. 326pp. 9004100322. 53343:076.

This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facility sites throughout the country for the fourth quarter of 1989.

NUREG-0837 V10 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. January-March 1990. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). June 1990. 227pp. 9007120185. 54470:336.

This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facilities throughout the country for the first quarter of 1990.

NUREG-0837 V10 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. April-June 1990. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). September 1990. 227pp. 9010090064. 55319:345.

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 1990.

NUREG-0837 V10 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. July-September 1990. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). December 1990. 230pp. 9101110198. 56356:001.

This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facilities throughout the country for the third quarter of 1990.

NUREG-0847 S05: 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). November 1990. 109pp. 9011260026. 55798:146.

Supplement No. 5 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. 4 was issued, and (2) matters that the staff had under review when Supplement No. 4 was issued.

NUREG-0896 S09: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SEABROOK STATION, UNITS 1 AND 2.Docket Nos. STN-50-443 And STN-50-444.(Public Service Company Of New Hampshire) * Division of Reactor Projects - I/II (Post 870411). March 1990. 166pp. 9004030159. 53229:042.

This report is Supplement No. 9 to the Safety Evaluation Report (SER) (NUREG-0896, March 1983) for the application filed by the Public Service Company of New Hampshire, et al., for licenses to operate Seabrook Station, Units 1 and 2 (Docket Nos. STN 50-443 and STN 50-444). It has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission and provides recent information on open items identified in the SER. The facility is located in Seabrook, New Hampshire. Subject to favorable resolution of the items discussed in this report, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

NUREG-0933 S11: A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R.; RIGGS,R.; MILSTEAD,W.; et al. Division of Regulatory Applications (Post 870413). July 1990. 180pp. 9008070346. 54862:255.

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

NUREG-0936 V08 N04: NRC REGULATORY AGENDA.Quarterly Report,October-December 1989. * Division of Freedom of Information & Publications Services (Post 890205). January 1990. 134pp. 9003070131. 52808:074.

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 V09 N01: NRC REGULATORY AGENDA.Quarterly Report,January-March 1990. * Division of Freedom of Information & Publications Services (Post 890205). April 1990. 145pp. 9005210250. 53889:001.

See NUREG-0936,V08,N04 abstract.

NUREG-0936 V09 N02: NRC REGULATORY AGENDA.Quarterly Report,April-June 1990. * Division of Freedom of Information & Publications Services (Post 890205). July 1990. 153pp. 9008140487. 54925:129.

See NUREG-0936,V08,N04 abstract.

NUREG-0936 V09 N03: NRC REGULATORY AGENDA.Quarterly Report,July-September 1990. * Division of Freedom of Information & Publications Services (Post 890205). October 1990. 151pp. 9011090217. 55661:164.

See NUREG-0936,V08,N04 abstract.

NUREG-0940 V08 N04: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,October-December 1989. * Ofc of Enforcement (Post 870413). March 1990. 289pp. 9005040056. 53623:190.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (October - December 1989) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions. Also included are a number of enforcement actions that had been previously resolved but not published in this NUREG. 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 V09 N01: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,January-March 1990. * Ofc of Enforcement (Post 870413). May 1990. 403pp. 9006290085. 54320:263.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January - March 1990) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions. Also included are a number of enforcement actions that had been previously resolved but not published in this NUREG. 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 V09 N02: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June 1990. * Ofc of Enforcement (Post 870413). September 1990. 502pp. 9009250062. 55242:170.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 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 V09 N03: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,July-September 1990. * Ofc of Enforcement (Post 870413). November 1990. 382pp. 9012110322. 56044:308.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (July - September 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-1021 R06: OPERATOR LICENSING EXAMINER STANDARDS. DEAN,W. Division of Licensee Performance & Quality Evaluation (Post 870411). June 1990. 315pp. 9006290065. 54321:339.

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

NUREG-1100 V06: BUDGET ESTIMATES.Fiscal Year 1991. * Division of Budget & Analysis (Post 890205). January 1990. 208pp. 9002120328. 52610:231.

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 year 1991.

NUREG-1125 V11: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS.1989 Annual. * ACRS - Advisory Committee on Reactor Safeguards. April 1990. 150pp. 9005210245. 53888:210.

This compilation contains 54 ACRS reports submitted to the Commission or to the Executive Director for Operations during calendar year 1989. 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 within project name by chronological order. Part 2 categorizes the reports by the most appropriate generic subject area and within subject area by chronological order.

NUREG-1145 V06: U.S. NUCLEAR REGULATORY COMMISSION 1989 ANNUAL REPORT. * Office of Administration (Post 890205). July 1990. 248pp. 9009040091. 55047:155.

This report covers the major activities, events, decisions, and planning that took place during fiscal year 1989 within the U.S. Nuclear Regulatory Commission (NRC) or involving the NRC.

NUREG-1150 V01: SEVERE ACCIDENT RISKS:AN ASSESSMENT FOR FIVE U.S. NUCLEAR POWER PLANTS.Final Summary Report. * Division of Systems Research (Post 880717). December 1990. 287pp. 9101110232. 56342:237.

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 1 of this report has three parts. Part I provides the background and objectives of the assessment and summarizes the methods used to perform the risk studies. Part II provides a summary of results obtained for each of the five plants studied. Part III provides perspectives on the results and discusses the role of this work in the larger context of the NRC staff's work.

NUREG-1150 V02: SEVERE ACCIDENT RISKS:AN ASSESSMENT FOR FIVE U.S. NUCLEAR POWER PLANTS.Appendices A,B, And C.Final Report. * Division of Systems Research (Post 880717). December 1990. 313pp. 9101110227. 56343:164.

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 2 of this report contains three appendices, providing greater detail on the methods used, an example risk calculation, and more detailed discussion of particular technical issues found important in the risk studies.

NUREG-1214 R05: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. ALLENSPACH,F.; NEASE,R. Division of Licensee Performance & Quality Evaluation (Post 870411). February 1990. 114pp. 9003070137. 52807:320.

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. These ratings are grouped by Region showing rating based on the revised Manual Chapter 0516 functional areas, then the pre-revised Manual Chapter 0516 functional areas and then for reactors under construction. Section 2 presents a chronological listing of all SALP report ratings for each operating facility. These ratings are also grouped by Region showing ratings based on the revised Manual Chapter 0516 functional areas and then the pre-revised Manual Chapter functional areas. 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 R06: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. ALLENSPACH,F.; WHARTON,R. Division of Licensee Performance & Quality Evaluation (Post 870411). August 1990. 114pp. 9008310213. 55043:056.

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-1232 V04: SAFETY EVALUATION REPORT ON TENNESSEE VALLEY AUTHORITY: WATTS BAR NUCLEAR PERFORMANCE PLAN. AULUCK,R. TVA Projects Division (890101-900602). January 1990. 84pp. 9003070159. 52809:041.

This safety evaluation report on the information submitted by the Tennessee Valley Authority in its Nuclear Performance Plan for the Watts Bar Nuclear Plant and in supporting documents has been prepared by the U.S. Nuclear Regulatory Commission staff. The plan addresses the plant-specific corrective actions as part of the recovery program for licensing of Unit 1. The staff will be monitoring and inspecting the implementation of the programs. The plan does not address all licensing matters that will be required for fuel load and operation of Unit 1. Those remaining licensing matters have been addressed in previous safety evaluations or will be addressed in accordance with routine NRC licensing practices.

NUREG-1266 V04: NRC SAFETY RESEARCH IN SUPPORT OF REGULATION - FY 1989. * Office of Nuclear Regulatory Research (Post 860720). April 1990. 61pp. 9005040092. 53622:113.

This report, the fifth 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 1989. 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 V04 N01: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA 1989 ANNUAL REPORT.Power Reactors. * Office for Analysis & Evaluation of Operational Data, Director. July 1990. 264pp. 9009040081. 55048:295.

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 1989. The report is published in two separate parts. NUREG-1272, Vol. 4, 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. This report also compiles the status of staff actions resulting from previous Incident Investigation Team (IIT) reports. NUREG-1272, Vol. 4, No. 2, covers nonreactors and presents a review of the events and concerns during 1989 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 V04 N02: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA 1989 ANNUAL REPORT.Nonreactors. * Office for Analysis & Evaluation of Operational Data, Director. July 1990. 68pp. 9009040059. 55048:199.

See NUREG-1272,V04,N01 abstract.

NUREG-1299 DRFT FC: STANDARD REVIEW PLAN FOR THE REVIEW OF LICENSE RENEWAL APPLICATIONS FOR NUCLEAR POWER PLANTS.Draft Report For Comment. * Office of Nuclear Reactor Regulation, Director (Post 870411). November 1990. 347pp. 9012110326. 56060:001.

The SRP-LR is to be used by the NRC staff when performing safety reviews of applications for the renewal of power reactor licenses. The use of the SRP-LR when reviewing license renewal applications provides a framework for the staff to determine whether or not (1) the application is sufficient to allow the timely renewal provisions of 10 CFR 2.109 to apply, (2) systems, structures, and components important to license renewal have been identified, (3) significant age-related degradation has been identified and its effects evaluated, and (4) programs for age-related degradation management have been or will be implemented such that the current licensing basis will be maintained during the renewal term. The draft SRP-LR has been developed to enable the staff to identify areas and issues requiring review, and provides acceptance criteria to assist the reviewers.

NUREG-1316: TECHNICAL FINDINGS AND REGULATORY ANALYSIS RELATED TO GENERIC ISSUE 70.Evaluation Of Power-Operated Relief Valve And Block Valve Reliability In PWR Nuclear Power Plants. KIRKWOOD,R. Division of Safety Issue Resolution (Post 880717). December 1989. 26pp. 9002120148. 52611:263.

This report summarizes work performed by the Nuclear Regulatory Commission staff to resolve Generic Issue 70, "Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Power-Operated Relief Valve and Block Valve Reliability." The report evaluates the reliability of PORVs and block valves and their safety significance in PWR nuclear power plants. The report identifies those safety-related functions that may be performed by PORVs and describes ways in which PORVs and block valves may be improved. This report also presents the regulatory analysis for Generic Issue 70.

NUREG-1333: MAINTENANCE APPROACHES AND PRACTICES IN SELECTED FOREIGN NUCLEAR POWER PROGRAMS AND OTHER U.S. INDUSTRIES: REVIEW AND LESSONS LEARNED. DEY,M. Division of Regulatory Applications (Post 870413). April 1990. 111pp. 9005070010. 53657:051.

The Commission published a Notice of Proposed Rulemaking on Maintenance of Nuclear Power Plants on November 28, 1988, spelling out NRC's expectations in maintenance. In preparing the proposed rule, the NRC reviewed maintenance practices in other countries and considered maintenance approaches in other industries in this country. As a result of the review of maintenance practices, it was concluded that certain practices in the following areas have been found to contribute significantly to effective maintenance: (1) systems approach; (2) effectiveness monitoring; (3) technician qualifications and motivation; and (4) maintenance organization.

NUREG-1339: RESOLUTION OF GENERIC SAFETY ISSUE 29:BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS. JOHNSON,R.E. Division of Safety Issue Resolution (Post 880717). June 1990. 23pp. 9008140466. 54927:158.

This report describes the U.S. Nuclear Regulatory Commission's (NRC's) Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants," including the bases for establishing the issue and its historical highlights. The report also describes the activities of the Atomic Industrial Forum (AIF) relevant to this issue, including its cooperation with the Materials Properties Council (MPC) to organize a task group to help resolve the issue. The Electric Power Research Institute, supported by the AIF/MPC task group, prepared and issued a two-volume document that provides, in part, the technical basis for resolving Generic Safety Issue 29. This report presents the NRC's review and evaluation of the two-volume document and NRC's conclusion that this document, in conjunction with other information from both industry and NRC, provides the bases for resolving this issue.

NUREG-1350 V02: NUCLEAR REGULATORY COMMISSION INFORMATION DIGEST.1990 Edition. OLIVE,K.L. Division of Budget & Analysis (Post 890205). March 1990. 117pp. 9004090296. 53315:278.

The Nuclear Regulatory Commission Information Digest provides summary information regarding the U.S. Nuclear Regulatory Commission, its regulatory responsibilities, and areas licensed by the Commission. This is an annual publication for the general use of the NRC Staff and is available to the public. The digest is divided into two parts: the first presents an overview of the U.S. Nuclear Regulatory Commission and the second provides data on NRC commercial nuclear reactor licensees and commercial nuclear power reactors worldwide.

NUREG-1352: ACTION PLANS FOR MOTOR-OPERATED VALVES AND CHECK VALVES. SCARBROUGH,T.G. Division of Engineering Technology (Post 890827). June 1990. 34pp. 9006290061. 54318:323.

The proper performance of motor-operated valves (MOVs) and check valves is necessary for the safe operation of a nuclear power plant. Problems have been experienced with these valves for many years. Currently, the U.S. Nuclear Regulatory Commission (NRC) and the nuclear industry have a number of activities under way to provide assurance that MOVs and check valves will successfully perform their safety functions when needed. The Mechanical Engineering Branch (EMEB) of the Office of Nuclear Reactor Regulation has been assigned the responsibility of coordinating NRC activities and monitoring industry activities regarding MOVs and check valves. To meet this responsibility, EMEB has prepared action plans to provide assurance of the proper performance of these valves. Through the action plans, the NRC staff will have an organized approach to resolve the concerns regarding the operability of MOVs and check valves in a timely manner.

NUREG-1362 DRFT FC: REGULATORY ANALYSIS FOR PROPOSED RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL.Draft Report For Comment. * Division of Safety Issue Resolution (Post 880717). July 1990. 213pp. 9008070329. 54862:042.

This regulatory analysis provides the supporting information for a proposed rule that will define 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 ratios is presented, and Alternative B is the most cost-beneficial safety review alternative.

NUREG-1363 V02: ATOMIC SAFETY AND LICENSING BOARD PANEL ANNUAL REPORT FISCAL YEAR 1989. COTTER,B.P. Atomic Safety and Licensing Board Panel. July 1990. 63pp. 9008070362. 54845:098.

In Fiscal Year 1989, the Atomic Safety and Licensing Board Panel (ASLBP) 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 ASLBP during the year.

NUREG-1372: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC ISSUE C-8, "MAIN STEAM ISOLATION VALVE LEAKAGE AND LCS FAILURE." GRAVES,C.C. Division of Safety Issue Resolution (Post 880717). June 1990. 29pp. 9006290054. 54321:306.

Generic Issue C-8 deals with staff concerns about public risk because of the incidence of leak test failures reported for main steam isolation valves (MSIVs) at boiling water reactors and the limitations of the leakage control systems (LCS) for mitigating the consequences of leakage from these valves. If the MSIV leakage is greatly in excess of the allowable value in the technical specifications, the LCS would be unavailable because of design limitations. The issue was initiated in 1983 to assess (1) the causes of MSIV leakage failures, (2) the effectiveness of the LCS and alternative mitigation paths, and (3) the need for additional regulatory action to reduce public risk. This report presents the regulatory analysis for Generic Issue C-8 and concludes that no new regulatory requirements are warranted.

NUREG-1377 R01: NRC RESEARCH PROGRAM ON PLANT AGING. LISTINGS AND SUMMARIES OF REPORTS ISSUED THROUGH MAY 1990. KONDIC,N.N.; HILL,E.L. Division of Engineering (Post 870413). July 1990. 62pp. 9008070371. 54845:036.

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 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 May 1990 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-1381: TECHNICAL SPECIFICATIONS, COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1.Docket No. 50-445,Appendix "A" To License No. NPF-28. * Comanche Peak Project Division (890101-900602). February 1990. 354pp. 9003120783. 52883:015.

The Technical Specifications for Comanche Peak Steam Electric Station, Unit 1 were prepared by the U.S. Nuclear Regulatory Commission. They set forth the limits, operating conditions, and other requirements applicable to a nuclear reactor facility, as set forth in Section 50.36 of Title 10 of the Code of Federal Regulations Part 50, for the protection of the health and safety of the public.

NUREG-1383: GUIDANCE ON THE APPLICATION OF QUALITY ASSURANCE FOR CHARACTERIZING A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE. Final Report. PITTIGLIO,C.L.; STARMER,R.J.; HEDGES,D. Division of Low-Level Waste Management & Decommissioning (Post 870413). October 1990. 17pp. 9010240396. 55495:259.

This document provides guidance to an applicant on meeting the Quality Control (QC) requirements of 10 CFR Part 61.12 for a Low-Level Waste (LLW) disposal facility. The QC requirements combined with the requirements for managerial controls and audits are the basis for developing a Quality Assurance (QA) program and for the guidance provided herein. The document specifically established QA guidance for site characterization activities necessary to meet the performance objectives of 10 CFR Part 61 and to limit exposure to or release of radioactivity.

NUREG-1386: TECHNICAL SPECIFICATIONS FOR SEABROOK STATION, UNIT 1. Appendix "A" To License No. NPF-86. * Division of Reactor Projects - I/II (Post 870411). March 1990. 419pp. 9004090200. 53307:001.

The Seabrook Station, Unit 1 Technical Specifications were prepared by the U.S. Nuclear Regulatory Commission to set forth the limits, operating conditions, and other requirements applicable to a nuclear reactor facility as set forth in Section 50.36 of 10 CFR Part 50 for the protection of the health and safety of the public.

NUREG-1390: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE TRIGA TRAINING AND RESEARCH REACTOR AT THE UNIVERSITY OF ARIZONA.Docket No. 50-113. (University of Arizona) * Division of Reactor Projects - III,IV,V & Special Projects (870411-901215). May 1990. 74pp. 9006080200. 54074:209.

This Safety Evaluation Report for the application filed by the University of Arizona for the renewal of Operating License R-52 to continue operating its research reactor at an increased operating power level has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located on the University of Arizona campus in Tucson, Arizona. The staff concludes that the reactor can continue to be operated by the University of Arizona without endangering the health and safety of the public.

NUREG-1391 DRFT FC: CHEMICAL TOXICITY OF URANIUM HEXAFLUORIDE RELATED TO RADIATION DOSES.Draft Report For Comment. MCGUIRE,S.A. Office of Nuclear Regulatory Research (Post 860720). April 1990. 16pp. 9005210238. 53861:326.

The chemical effects from large acute exposures to uranium hexafluoride are compared to the effects from acute radiation doses of 25 rems to the whole body and of 300 rems to the thyroid. The analysis concludes that an acute exposure to about 500 mg-min/m(3) of hydrogen fluoride is roughly equivalent, in terms of early effects, to an acute whole body dose of 25 rems. Similarly, an intake of about 10 mg of uranium in soluble form is roughly equivalent, in terms of early effects, to an acute whole body dose of 25 rems. The purpose of these analyses is to provide information for developing siting criteria for uranium enrichment plants based on chemical toxicity. These siting criteria are to be similar, in terms of health and safety impact on workers and members of the public, to the siting criteria in NRC regulations for nuclear power plants, which are based on radiation doses.

NUREG-1392: LEAKAGE OF AN IRRADIATOR SOURCE - THE JUNE 1988 GEORGIA RSI INCIDENT. * Office of Governmental & Public Affairs (Post 870413). SETSER,J.L. Georgia, State of. February 1990. 45pp. 9003190259. 53005:245.

On June 6, 1988, operators of a pool irradiator in Decatur, Georgia were prevented by a safety system from raising sources from the pool. Radiation levels of 60 millirem per hour at the surface of the pool water were found, indicative of a leak of one or more of the 252 Cs-137 source capsules used at the irradiator. Because of the concerns which arose out of this incident, the State of Georgia and the Conference of Radiation Control Program Directors, Inc. decided it should be reviewed in depth. Georgia Governor, The Honorable Joe Frank Harris, created an Incident Evaluation Task Force and charged it with collecting information on the incident, maintaining communications with the DOE Investigative Board and preparing a written report of lessons learned. Since the incident and responses to it are still ongoing, a final report of the task force is expected at a later date. A summary of the Task Force's First Interim Report has been prepared for persons needing an overview of the incident and lessons learned to date. The Conference established an Incident Review Team which agreed to assume the responsibility from the Georgia task force to discuss the role of the States in regulating irradiators. Its Interim Report provides a summary of Agreement States' views and recommendations an some of the issues raised by the incident.

NUREG-1393: THE INCINERATION OF LOW-LEVEL RADIOACTIVE WASTE.A Report For The Advisory Committee On Nuclear Waste. LONG,S.W. Advisory Committee on Nuclear Waste. June 1990. 75pp. 9007120183. 54469:053.

This report is a summary of the contemporary use of incineration technology as a method for volume reduction of LLW. It is intended primarily to serve as an overview of the technology for waste management professionals involved in the use or regulation of LLW incineration. It is also expected that organizations presently considering the use of incineration as part of their radioactive waste management programs will benefit by gaining a general knowledge of incinerator operating experience. Specific types of incineration technologies are addressed in this report, including designation of the kinds of wastes that can be processed, the magnitudes of volume reduction that are achievable in typical operation, and requirements for ash handling and off-gas filtering and scrubbing. A status listing of both U.S. and foreign incinerators provides highlights of activities at government, industry, institutional, and commercial nuclear power plant sites.

NUREG-1394: EMERGENCY RESPONSE DATA SYSTEM (ERDS) IMPLEMENTATION. JOLICOEUR,J.R. Office for Analysis & Evaluation of Operational Data, Director. April 1990. 53pp. 9006080154. 54077:312.

The U.S. Nuclear Regulatory Commission has begun implementation of the Emergency Response Data System (ERDS) to upgrade its ability to acquire data from nuclear power plants in the event of an emergency at the plant. ERDS provides a direct real-time transfer of data from licensee plant computers to the NRC Operations Center. The system has been designed to be activated by the licensee during an emergency which has been classified at an ALERT or higher level. The NRC portion of ERDS will receive the data stream, sort and file the data. The users will include the NRC Operations Center, the NRC Regional Office of the affected plant, and if requested, the States which are within the ten mile EPZ of the site. The currently installed Emergency Notification System will be used to supplement ERDS data. This report provides the minimum guidance for implementation of ERDS at licensee sites. It is intended to be used for planning implementation under the current voluntary program as well for providing the minimum standards for implementing the proposed ERDS rule.

NUREG-1395 DRFT: INDUSTRY PERCEPTIONS OF THE IMPACT OF THE U.S. NUCLEAR REGULATORY COMMISSION ON NUCLEAR POWER PLANT ACTIVITIES. Draft Report. DAVIS,A.S.; PEDERSON,C.D. Region 3 (Post 820201). March 1990. 180pp. 9004090218. 53313:190.

Teams of senior managers from the NRC surveyed licensee staff members representing 13 nuclear power utilities from across the country to obtain their candid views of the effectiveness and impact of NRC regulatory activities. Licensee comments addressed the full scope of NRC activities and the impact of agency actions on licensee resources, staff performance, planning and scheduling, and organizational effectiveness. The principal themes of the survey respondents' comments are that (1) licensees acquiesce to NRC requests to avoid poor ratings on NRC Systematic Assessment of Licensee Performance (SALP) reports and the consequent financial and public perception problems that result, even if the requests require the expenditure of significant resources on matters of marginal safety significance, and (2) NRC so dominates licensee resources through its existing and changing formal and informal requirements that licensees believe that their plants, though not unsafe, would have better reliability, and may even achieve a higher degree of safety, if licensees were freer to manage their own resources. This draft report does not attempt to defend any NRC position; endorse or refute licensee perceptions; or explain any action taken by NRC in fulfilling its responsibilities to protect the health and safety of the public. Senior NRC managers have made a preliminary evaluation of the information in this report and have made recommendations to address licensee concerns in some areas. The final evaluation and recommendations will be published at a later date as the final NUREG.

NUREG-1396: ENVIRONMENTAL ASSESSMENT OF THE THERMAL NEUTRON ACTIVITATION EXPLOSIVE DETECTION SYSTEM FOR CONCOURSE USE AT U.S. AIRPORTS. JONES,C.G. Division of Industrial & Medical Nuclear Safety (Post 870729). August 1990. 160pp. 9010090050. 55322:093.

This document is an environmental assessment of a system designed to detect the presence of explosives in checked airline baggage or cargo. The system is meant to be installed at the concourse or lobby ticketing areas of U.S. commercial airports and uses a sealed radioactive source of californium-252 to irradiate baggage items. The major impact of the use of this system arises from direct exposure of the public to scattered or leakage radiation from the source and to induced radioactivity in baggage items. Under normal operation and the most likely accident scenarios, the environmental impacts that would be created by the proposed licensing action would not be significant.

NUREG-1398 DRFT FC: ENVIRONMENTAL ASSESSMENT FOR PROPOSED RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL.Draft Report For Comment. * Division of Safety Issue Resolution (Post 880717). July 1990. 57pp. 9008070358. 54845:250.

The possible environmental effects of promulgating nuclear power plant license renewal standards by the proposed 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 operating conditions of plants in ways that 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-1399: TECHNICAL SPECIFICATIONS,COMANCHE PEAK STEAM ELECTRIC STATION,UNIT 1.Docket No. 50-445.Appendix "A" To License NPF-87.(Texas Utilities Electric) * Comanche Peak Project Division (890101-900602). April 1990. 360pp. 9006080150. 54081:109.

The Technical Specifications for Comanche Peak Steam Electric Station, Unit 1 were prepared by the U.S. Nuclear Regulatory Commission. They set forth the limits, operating conditions, and other requirements applicable to a nuclear reactor facility, as set forth in Section 50.36 of Title 10 of the Code of Federal Regulations Part 50, for the protection of the health and safety of the public.

NUREG-1402: CLOSEOUT OF NRC BULLETIN 88-05:NONCONFORMING MATERIALS SUPPLIED BY PIPING SUPPLIES,INC.,AT FOLSOM,NEW JERSEY,AND WEST JERSEY MANUFACTURING COMPANY AT WILLIAMSTOWN,NEW JERSEY. * Division of Engineering Technology (Post 890827). May 1990. 79pp. 9006110010. IEB-88-005. 54083:185.

This report documents the activities that led to the closeout of U.S. Nuclear Regulatory Commission (NRC) Bulletin 88-05, which was issued on May 6, 1988. The bulletin required that licensees submit information on materials supplied by Piping Supplies, Inc. (PSI) and West Jersey Manufacturing Company (WJM), and requested that they (1) ensure that these materials complied with the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code (ASME Code) and design specifications or were suitable for their intended service or (2) replace such materials. Supplements 1 and 2 were issued on June 15 and August 3, 1988, respectively. In Supplement 2, another affiliated supplier, Chews Landing Metal Manufacturers, Incorporated (CLM), was identified. The staff concluded that (1) the analytical procedures used to qualify the nonconforming parts and the analysis results provide an adequate basis for resolving the staff's concerns regarding fittings and flanges; (2) even though the materials supplied by PSI, WJM, and CLM with falsified certified material test reports do not meet the ASME Code, their use is an acceptable alternative in accordance with Section 50.55a(a)(3)(ii) of Title 10 of the Code of Federal Regulations; (3) activities in response to Bulletin 88-05 regarding fittings and flanges can be closed for all operating plants; and (4) licensees should evaluate the use of product forms other than fittings and flanges.

NUREG-1403: SAFETY EVALUATION REPORT RELATED TO THE FULL-TERM OPERATING LICENSE FOR DRESDEN NUCLEAR POWER STATION,UNIT 2.Docket No. 50-237.(Commonwealth Edison Company) * Division of Reactor Projects - III,IV,V & Special Projects (870411-901215). October 1990. 39pp. 9010230057. 55472:277.

The Safety Evaluation Report for the full-term operating license application filed by Commonwealth Edison Company for the Dresden Nuclear Power Station, Unit 2, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Grundy County, Illinois. Subject to favorable resolution of the items discussed in this report, the staff concludes that the facility can continue to be operated without endangering the health and safety of the public.

NUREG-1404: LICENSEE USE OF TACTICAL EXERCISE RESULTS. SAWYER,C.; BROWN,C. Division of Safeguards & Transportation (Post 870413). April 1990. 11pp. 9005210234. 53861:312.

On November 10, 1988 the Nuclear Regulatory Commission (NRC) amended its physical security requirements in 10 CFR Part 73 for fuel facilities possessing formula quantities of strategic special nuclear material. The amendments to 10 CFR 73.46(b) require, among other things, that licensees carry out performance evaluations through tactical response exercises. The exercises are intended to demonstrate the guard force state of readiness and to test the effectiveness of delay mechanisms, alarm and communication systems, response times, deployment of response forces, firing skills (simulated), tactical maneuvers, etc. The purpose of this document is to set forth criteria, acceptable to the NRC staff, which will enable a licensee to use the results of an exercise to determine whether additional training or security improvements are needed.

NUREG-1405: INADVERTENT SHIPMENT OF A RADIOGRAPHIC SOURCE FROM KOREA TO AMERSHAM CORPORATION,BURLINGTON,MASSACHUSETTS. * NRC - No Detailed Affiliation Given. May 1990. 177pp. 9006110005. 54085:017.

Amersham Corporation, Burlington, Massachusetts, a licensee of the U.S. Nuclear Regulatory Commission (NRC), authorized to manufacture and distribute iridium-192 and cobalt-60 source assemblies for use in radiography equipment, received a shipment of 14 source changers on March 8, 1990, that were being returned from their product distributor, NDI Corporation in Seoul, Korea. One source changer contained a small sealed source in an unshielded location. Amersham employees retrieved the source, secured it in a hot cell, and notified NRC's Region I. Subsequently, NRC dispatched an Incident Investigation Team to perform a comprehensive review of this incident and determine the potential for exposure to those who handled the source changer and to members of the general public. This report describes the incident and the methodology used in the investigation and presents the Team's findings and conclusions.

NUREG-1407 DRFT FC: PROCEDURAL AND SUBMITTAL GUIDANCE FOR INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR SEVERE ACCIDENT VULNERABILITIES.Draft Report For Comment. * Division of Safety Issue Resolution (Post 880717). July 1990. 41pp. 9008070367. 54845:307.

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 (IPEEE). The guidance for reporting the results of the individual plant examination of internal events (IPE) is presented in NUREG-1335.

NUREG-1409: BACKFITTING GUIDELINES. ALLISON,D.P.; CONRAN,J.H.; TROTTIER,C.A. Office for Analysis & Evaluation of Operational Data, Director. July 1990. 102pp. 9008070378. 54846:270.

The backfitting process is the process by which the U.S. Nuclear Regulatory Commission (NRC) decides whether to issue new or revised requirements or staff positions to licensees of nuclear power reactor facilities. Requirements for proper justification of backfits and information requests are provided by two NRC rules (Title 10, Code of Federal Regulations, Sections 50.109 and 50.54(f)). NRC procedures include the charter of the committee to Review Generic Requirements, NRC Manual Chapter 0514, and individual office procedures. Three types of backfits are recognized. Cost-justified substantial safety improvements require backfit analyses and findings of (1) substantial safety improvement and (2) justified costs. Compliance exceptions and adequate protection exceptions do not require findings of substantial safety improvements and costs are not considered. However, they are still backfits and require documented evaluations to support the use of the exceptions. Information requests (as opposed to backfits) require an analysis of the burden to be imposed to ensure that they are justified in view of the potential safety significance of the information requested.

NUREG-1410: LOSS OF VITAL AC POWER AND THE RESIDUAL HEAT REMOVAL SYSTEM DURING MID-LOOP OPERATIONS AT VOGTLE UNIT 1 ON MARCH 20, 1990. * Ofc of the Executive Director for Operations. June 1990. 522pp. 9006290080. 54319:101.

On March 20, 1990, the Vogtle Electric Generating Plant Unit 1, located in Burke County, Georgia, about 25 miles southeast of Augusta, experienced a loss of all safety (vital) ac power. The plant was in cold shutdown with reactor coolant level lowered to "mid-loop" for various maintenance tasks. Both the containment building personnel hatch and equipment hatch were open. One emergency diesel generator and one reserve auxiliary transformer were out of service for maintenance, with the remaining reserve auxiliary transformer supplying both Unit 1 safety buses. A truck in the low voltage switchyard backed into the support column for an offsite power feed to the reserve auxiliary transformer which was supplying safety power. The insulator broke, a phase-to-ground fault occurred, and the feeder circuit breakers for the safety buses opened. The operable emergency diesel generator started automatically because of the undervoltage condition on the safety bus, but tripped off after about 1 minute. About 20 minutes later the diesel generator load sequencer was reset, causing the diesel generator to start a second time. The diesel generator started again, operated for about 1 minute, and tripped off. The diesel generator was restarted in the manual emergency mode 36 minutes after the loss of power. The generator remained on line and provided power to its safety bus. During the 36 minutes without safety bus power, the reactor coolant system temperature rose from about 90°F to 136°F. This report documents the results of an

Incident Investigation Team sent to Vogtle by the Executive Director for Operations of the U.S. Nuclear Regulatory Commission to determine what happened, identify the probable causes, and make appropriate findings and conclusions.

NUREG-1411: RESPONSE TO PUBLIC COMMENTS RESULTING FROM THE PUBLIC WORKSHOP ON NUCLEAR POWER PLANT LICENSE RENEWAL. * Office of Nuclear Regulatory Research (Post 860720). July 1990. 75pp. 9008070323. 54845:348.

On October 13, 1989, the U.S. Nuclear Regulatory Commission (NRC) issued an Advance Notice of Proposed Rulemaking on nuclear power plant license renewal. The notice presented the NRC's preliminary regulatory philosophy and approach for developing license renewal regulations and solicited comments on a number of technical and policy issues. It also announced plans for a public workshop to discuss the issues and to receive comments and information. The workshop was held on November 13-14, 1989, in Reston, Virginia. This document reports on the NRC's response to the public comments from the workshop and written comments on the workshop topics received shortly after the workshop. (The proceedings of the workshop were reported in NUREG/CP-0108).

NUREG-1412 DRFT FC: FOUNDATION FOR THE ADEQUACY OF THE LICENSING BASES.A Supplement To The Statement Of Considerations For The Proposed Rule On Nuclear Power Plant License Renewal (10 CFR Part 54).Draft Report For Comment. * Office of Nuclear Reactor Regulation, Director (Post 870411). July 1990. 106pp. 9008070326. 54860:125.

In order to limit the Commission's license renewal decision to consideration of whether age-related degradation has been adequately addressed, the Part 54 rulemaking must make a generic finding for all nuclear power plants that the findings of reasonable assurance of adequate protection for issuance of an operating license continue to be true at the time of the renewal application and accordingly need not be made anew at the time of license renewal. This analysis describes the regulatory processes that form the basis for such a finding. This document discusses how the licensing process has evolved in major safety issue areas under existing regulatory processes that have ensured continued adequacy of the licensing bases of all operating plants. The document presents the described regulatory processes as the Commission's reasons for considering it unnecessary to re-review an operating plant's licensing basis, except for age-related degradation concerns, at the time of license renewal. This report is a supplement to the Statement of Considerations for the Nuclear Regulatory Commission's proposed rule (10 CFR Part 54) that would establish the criteria and standards governing nuclear power plant license renewal.

NUREG-1415 V03 N01: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report.April-September 1990. GLENN,W.L.; WATKINS,R.A. Office of Inspector General (Post 890417). October 1990. 30pp. 9101110208. 56344:117.

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-1417: SAFETY EVALUATION REPORT RELATED TO HYDROGEN CONTROL OWNERS GROUP ASSESSMENT OF MARK III CONTAINMENT. LI,C.Y.; KUDRICK,J.A. Division of Systems Technology (Post 890827). October 1990. 44pp. 9010300241. 55629:017.

Title 10 of the Code of Federal Regulations (10 CFR), Section 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors," requires that systems be provided to control hydrogen concentration in the containment atmosphere following an accident to ensure that containment integrity is maintained. The purpose of this report is to provide regulatory guidance to licensees with Mark III containments with regard to demonstrating compliance with 10 CFR 50.44, Sections (c)(3)(vi) and (c)(3)(vii). In this report, the staff provides its evaluation of the generic methodology proposed by the Hydrogen Control Owners Group. This generic methodology is documented in Topical Report HGN-112-NP, "Generic Hydrogen Control Information for BWR/6 Mark III Containments." In addition, the staff has recommended that the vulnerability to interruption of power to the hydrogen igniters be evaluated further on a plant-specific basis as part of the individual plant examination of the plants with Mark III containments.

NUREG-1418: CHARACTERISTICS OF LOW-LEVEL RADIOACTIVE WASTE DISPOSED DURING 1987 THROUGH 1989. ROLES,G.W. Division of Low-Level Waste Management & Decommissioning (Post 870413). December 1990. 629pp. 9012280240. 56268:001.

This report presents the volume, activity, and radionuclide distributions in low-level radioactive waste (LLW) disposed during 1987 through 1989 at the commercial disposal facilities located near Barnwell, SC, Richland, WA, and Beatty, NV. The report has been entirely assembled from descriptions of waste provided in LLW shipment manifests. Individual radionuclide distributions are listed as a function of waste class, of general industry, and of waste stream. In addition, information is presented about disposal of wastes containing chelating agents, about use of solidification media, about the distribution of radiation levels at the surfaces of waste containers, and about the distribution of waste container sizes. Considerably more information is presented about waste disposed at the Richland and Beatty disposal facilities than at the Barnwell disposal facility.

NUREG-1420: SPECIAL COMMITTEE REVIEW OF THE NUCLEAR REGULATORY COMMISSION'S SEVERE ACCIDENT RISKS REPORT (NUREG-1150). KOUTS,H.J.C.; APOSTOLAKIS,G.; BIRKHOFER,E.H.; et al. Office of Nuclear Regulatory Research (Post 860720). August 1990. 90pp. 9009110129. 55124:136.

In April 1989, the Nuclear Regulatory Commission's (NRC) Office of Nuclear Regulatory Research (RES) published a draft report "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150. This report updated, extended and improved upon the information presented in the 1974 "Reactor Safety Study," WASH-1400. Because the information in NUREG-1150 will play a significant role in implementing the NRC's Severe Accident Policy, its quality and credibility are of critical importance. Accordingly, the Commission requested that RES conduct a peer review of NUREG-1150 to ensure that the methods, safety insights and conclusions presented are appropriate and adequately reflect the current state of knowledge with respect to reactor safety. To this end, RES formed a special committee in June of 1989 under the provisions of the Federal Advisory Committee Act. The Committee, composed of a group of recognized national and international experts in nuclear reactor safety, was charged with preparing a report reflecting their review of NUREG-1150 with respect to the adequacy of the methods, data, analysis and conclusions it set forth. The report which precedes reflects the results of this peer review.

NUREG-1423 V01: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON NUCLEAR WASTE.July 1988 - June 1990. * Advisory Committee on Nuclear Waste. August 1990. 115pp. 9010230048. 55472:352.

This compilation contains 37 reports issued by the Advisory Committee on Nuclear Waste (ACNW) during the first two years of its operation. The reports were submitted to the Chairman or to the Executive Director for Operations, U.S. Nuclear Regulatory Commission (NRC). Topics include the NRC analysis of the U.S. Department of Energy Site Characterization Plan for the high-level radioactive waste repository, the standards promulgated by the U.S. Environmental Protection Agency for the disposal of high-level waste, the NRC policy statement on Below Regulatory Concern, technical documents prepared by the NRC staff relative to the decommissioning of nuclear power plants, the stabilization of uranium mill tailings piles, and environmental monitoring. 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. Included in an appendix is a listing of references to related reports on nuclear waste matters that were issued by the Advisory Committee on Reactor Safeguards prior to the establishment of the ACNW.

NUREG-1424: SAFETY EVALUATION REPORT RELATED TO THE FULL-TERM OPERATING LICENSE FOR PALISADES NUCLEAR PLANT.Docket No. 50-255. (Consumers Power Company) MASCIANTONIO,A. Division of Reactor Projects - III,IV,V & Special Projects (870411-901215). November 1990. 43pp. 9012110312. 56056:146.

This safety evaluation report relates to the issuance of a full-term operating license (FTOL) for the Palisades Nuclear Generating Plant in response to an application filed by the Consumers Power Company. The report provides an update of the status of the Systematic Evaluation Program (SEP) issues that were not fully resolved in the SEP Integrated Plant Safety Assessment Report (NUREG-0820 and NUREG-0820 Supplement 1). This report also addresses the status of requirements stemming from the accident at Three Mile Island Unit 2, unresolved safety issues, and other important plant- specific issues that have not yet been resolved. The staff has evaluated the issues related to the conversion of the provisional operating license to a full-term operating license and concluded that the facility can continue to be operated without endangering the health and safety of the public.

NUREG-1425: WELDING AND NONDESTRUCTIVE EXAMINATION ISSUES AT SEABROOK NUCLEAR STATION.An Independent Review Team Report. SPESSARD,R.L.; COLEY,J.; CROWLEY,W.; et al. Ofc of the Executive Director for Operations. July 1990. 404pp. 9008140491. 54925:282.

In response to congressional concerns about the adequacy of the welding and nondestructive examination (NDE) programs at the Seabrook Nuclear Station, NRC senior management established an independent review team (IRT) to conduct an assessment. The IRT focused on the quality of the finished hardware and associated records, as well as on the adequacy of the overall quality assurance program as applied to the fabrication and NDE programs for pipe welds. This report documents the findings of that investigation.

NUREG/CP-0105 V01: PROCEEDINGS OF THE SEVENTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. March 1990. 499pp. 9004030082. 53226:322.

This three-volume report contains 84 papers out of the 111 that were presented at the Seventeenth Water Reactor Safety Information Meeting held at the Holiday Inn Crowne Plaza, Rockville, Maryland, during the week of October 23-25, 1989. 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 ten different papers presented by researchers from France, Germany, Japan, Norway and the United Kingdom. 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-0105 V02: PROCEEDINGS OF THE SEVENTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. March 1990. 525pp. 9004090184. 53310:211.

See NUREG/CP-0105,V01 abstract.

NUREG/CP-0105 V03: PROCEEDINGS OF THE SEVENTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. March 1990. 536pp. 9004090193. 53308:060.

See NUREG/CP-0105,V01 abstract.

NUREG/CP-0107: PROCEEDINGS OF THE UNITED STATES NUCLEAR REGULATORY COMMISSION SECURITY TRAINING SYMPOSIUM.Meeting The Challenge-Firearms And Explosives Recognition And Detection. * Division of Safeguards & Transportation (Post 870413). September 1990. 308pp. 9010230103. 55470:272.

These conference proceedings have been prepared in support of the U.S. Nuclear Regulatory Commission's Security Training Symposium on "Meeting the Challenge-Firearms and Explosives Recognition and Detection," November 28 through 30, 1989 in Bethesda, Maryland. This document contains the edited transcripts of the guest speakers. It also contains some of the speaker's formal papers that were distributed and some of the slides that were shown at the symposium (Appendix A).

NUREG/CP-0108: PUBLIC WORKSHOP ON NUCLEAR POWER PLANT LICENSE RENEWAL. HUGHES,A.A.; LIGON,D.M.; SETH,S.S. MITRE Corp. April 1990. 78pp. 9005020348. MTR-90W00013. 53589:136.

On 13 October 1989, the U.S. Nuclear Regulatory Commission (NRC) issued an Advance Notice of Proposed Rulemaking on nuclear power plant license renewal. The notice presented the NRC's preliminary regulatory philosophy and approach for developing license renewal regulations, and solicited comments on a number of technical and policy issues. It also announced plans for a public workshop to discuss the issues and to receive comments and information. Representatives from 89 organizations attended the workshop, held on 13-14 November 1989, in Reston, Virginia. Subsequently, 12 organizations submitted written comments to the NRC. This report provides a summary of both workshop and written comments.

NUREG/CP-0109: PROCEEDINGS OF THE SEMINAR ON LEAK-BEFORE-BREAK.Further Developments In Regulatory Policies And Supporting Research. WILKOWSKI,G.M. Battelle Memorial Institute, Columbus Laboratories. CHAO,K.-S. Taiwan Power Co. February 1990. 352pp. 9003190303. 53043:231.

The fourth in a series of International Leak-Before-Break (LBB) Seminars supported in part by the U.S. Nuclear Regulatory Commission was held at the National Central Library in Taipei, Taiwan on May 11 and 12, 1989. The seminar updated the international policies and supporting research on LBB. Attendees included representatives from regulatory agencies, electric utility representatives, nuclear power plant fabricators, research organizations, and academic institutions. Regulatory policy was the subject of presentations by Mr. G. Arlotto (U.S. NRC, U.S.A.), Dr. B. Jarman (AECB, Canada), Dr. P. Milella (ENEA-DISP, Italy), Dr. C. Faidy (EDF/Septen, France), and Dr. K. Takumi (NUPEC, Japan). A paper by Mr. K. Wichman and Mr. S. Lee of the U.S. NRC Office of Nuclear Reactor Regulation is included as background material to these proceedings; it discusses the history and status of LBB applications in U.S. nuclear power plants. In addition, several papers on the supporting research programs described regulatory policy or industry standards for flaw evaluations, e.g., the ASME Section XI code procedures. Supporting research programs were reviewed on the first and second day by several participants from Taiwan, U.S., Japan, Canada, Italy, and France.

NUREG/CP-0110: PROCEEDINGS OF THE INTERNATIONAL WORKSHOP ON NEW DEVELOPMENTS IN OCCUPATIONAL DOSE CONTROL AND ALARA IMPLEMENTATION AT NUCLEAR POWER PLANTS AND SIMILAR FACILITIES. BAUM,J.W.; DIONNE,B.J.; KHAN,T.A. Brookhaven National Laboratory. February 1990. 534pp. 9004090207. BNL-NUREG-52226. 53312:016.

This report contains summaries of papers and discussions presented at the International Workshop on New Developments in Occupational Dose Control and ALARA Implementation at Nuclear Power Plants and Similar Facilities held at Brookhaven National Laboratory, Upton, New York, September 18-21, 1989. The purpose of this workshop was to bring together scientists, engineers, regulators, and administrators who are involved with occupational dose control at nuclear facilities to exchange information on recent developments from their countries. The eleven countries represented included: Canada, Finland, France, Germany, Italy, Japan, Luxembourg, Sweden, Switzerland, the United Kingdom, and the United States of America. This workshop was sponsored jointly by the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy, in cooperation with the Organization for Economic Cooperation and Development, Nuclear Energy Agency.

NUREG/CP-0111: PROCEEDINGS OF THE SYMPOSIUM ON INSERVICE TESTING OF PUMPS AND VALVES.Held At The Hyatt Regency Hotel,Washington,DC, August 1-3, 1989. * EG&G Idaho, Inc. (subs. of EG&G, Inc.). October 1990. 478pp. 9011060112. EGG-2609. 55623:046.

The 1989 symposium on Inservice Testing of Pumps and Valves, jointly sponsored by the Board on Nuclear Codes and Standards of the American Society of Mechanical Engineers and by the Nuclear Regulatory Commission, provided a forum for the discussion of current programs and methods for inservice testing at nuclear power plants. The symposium also provided an opportunity to discuss the need to improve inservice testing in order to ensure the reliable performance of pumps and valves. The participation of industry representatives, regulators and consultants resulted in the discussion of a broad spectrum of ideas and perspectives regarding the improvement of inservice testing of pumps and valves at nuclear power plants.

NUREG/CP-0112 V01: PROCEEDINGS OF THE THIRD INTERNATIONAL ATOMIC ENERGY AGENCY SPECIALISTS' MEETING ON SUBCRITICAL CRACK GROWTH.Opening Session And Technical Session I.Held At Moscow,USSR,May 14-17,1990. CULLEN,W.H. Materials Engineering Associates, Inc. * Argonne National Laboratory. August 1990. 312pp. 9009200013. ANL-90/22. 55214:324.

This report is a compilation of papers which were presented at the Third IAEA Specialists' Meeting on Subcritical Crack Growth, held in Moscow, USSR, on May 14-17, 1990. Volume 1 contains the welcoming remarks and attendance records, as well as the contributed papers for Session I, covering Corrosion Fatigue. Volume 2 contains the contributed papers for Sessions II, III, and IV, covering Stress-Corrosion Cracking, Test Methods, Models and Mechanisms, and summaries of National Programs in Argentina and the United Kingdom. Included as well are the Conclusions and Recommendations (Session V) developed by the organizing committee of the meeting, and discussed and approved by the participants.

NUREG/CP-0112 V02: PROCEEDINGS OF THE THIRD INTERNATIONAL ATOMIC ENERGY AGENCY SPECIALISTS' MEETING ON SUBCRITICAL CRACK GROWTH.Technical Sessions II, III, And IV And Recommendations And Conclusions Session V.Held At Moscow,USSR,May 14-17, 1990. CULLEN,W.H. Materials Engineering Associates, Inc. * Argonne National Laboratory. August 1990. 219pp. 9009200015. ANL-90/22. 55215:276.

See NUREG/CP-0112,V01 abstract.

NUREG/CP-0113: TRANSACTIONS OF THE EIGHTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Office of Nuclear Regulatory Research (Post 860720). October 1990. 218pp. 9010230093. 55471:220.

This report contains summaries of papers on reactor safety research to be presented at the 18th Water Reactor Safety Information Meeting at the Holiday Inn Crowne Plaza in Rockville, Maryland, October 22-24, 1990. 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 the electric utilities, the Electric Power Research Institute (EPRI), 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-0672 ADD04: TECHNOLOGY, SAFETY AND COSTS OF DECOMMISSIONING A REFERENCE BOILING WATER REACTOR POWER STATION.Comparison Of Two Decommissioning Cost Estimates Developed For The Same Commercial Nuclear Reactor Power Station. KONZEK,G.J.; SMITH,R.I. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1990. 73pp. 9012280246. 56257:259.

This study presents the results of a comparison of a previous decommissioning cost study by Pacific Northwest Laboratory (PNL) and a recent decommissioning cost study by TLG Engineering, Inc., for the same commercial nuclear power reactor station. The purpose of this comparative analysis on the same plant is to determine the reasons why subsequent estimates for similar plants by others were significantly higher in cost and external occupational radiation exposure (ORE) than the PNL study. The primary purpose of the original study by PNL (NUREG/CR-0672) was to provide information on the available technology, the safety considerations, and the probable costs and ORE for the decommissioning of a large BWR power station at the end of its operating life. This information was intended for use as background data and bases in the modification of existing regulations and in the development of new regulations pertaining to decommissioning activities. It was also intended for use by utilities in planning for the decommissioning of their nuclear power stations. The TLG study was performed in 1989 for the same plant, Washington Public Power Supply System's Unit 2 (WNP-2), that PNL used as its reference plant in its 1980 decommissioning study. Areas of agreement and disagreement are identified, and reasons for the areas of disagreement are discussed.

NUREG/CR-1667: RISK METHODOLOGY FOR GEOLOGIC DISPOSAL OF RADIOACTIVE WASTE. Scenario Selection Procedure. CRANWELL,R.M.; GUZOWSKI,R.V.; CAMPBELL,J.E.; et al. Sandia National Laboratories. April 1990. 114pp. 9005210151. SAND80-1429. 53887:090.

This report contains the description of a procedure for selecting scenarios that are potentially important to the isolation of high level radioactive wastes in deep geologic formations. In this report, the term scenario is used to represent a set of naturally occurring and/or human-induced conditions that represent realistic future states of the repository, geologic systems, and ground-water flow systems that might affect the release and transport of radionuclides from the repository to humans. The scenario selection procedure discussed in this report is demonstrated by applying it to the analysis of a hypothetical waste disposal site containing a bedded-salt formation as the host medium for the repository. A final set of 12 scenarios is selected for this site.

NUREG/CR-2000 V08N12: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of December 1989. * Oak Ridge National Laboratory. January 1990. 70pp. 9002120184. ORNL/NSIC-200. 52611:131.

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 V09 N1: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of January 1990. * Oak Ridge National Laboratory. February 1990. 112pp. 9004090292. ORNL/NSIC-200. 53315:166.
See NUREG/CR-2000,V08,N12 abstract.

NUREG/CR-2000 V09 N2: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of February 1990. * Oak Ridge National Laboratory. March 1990. 120pp. 9004090225. ORNL/NSIC-200. 53314:122.
See NUREG/CR-2000,V08,N12 abstract.

NUREG/CR-2000 V09 N3: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of March 1990. * Oak Ridge National Laboratory. April 1990. 83pp. 9005210152. ORNL/NSIC-200. 53886:325.
See NUREG/CR-2000,V08,N12 abstract.

NUREG/CR-2000 V09 N4: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of April 1990. * Oak Ridge National Laboratory. May 1990. 83pp. 9006080270. ORNL/NSIC-200. 54074:072.
See NUREG/CR-2000,V08,N12 abstract.

NUREG/CR-2000 V09 N5: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of May 1990. * Oak Ridge National Laboratory. June 1990. 96pp. 9007120242. ORNL/NSIC-200. 54468:009.
See NUREG/CR-2000,V08,N12 abstract.

NUREG/CR-2000 V09 N6: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of June 1990. * Oak Ridge National Laboratory. July 1990. 89pp. 9008140508. ORNL/NSIC-200. 54927:069.
See NUREG/CR-2000,V08,N12 abstract.

NUREG/CR-2000 V09 N7: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of July 1990. * Oak Ridge National Laboratory. August 1990. 101pp. 9009110143. ORNL/NSIC-200. 55194:181.
See NUREG/CR-2000,V08,N12 abstract.

NUREG/CR-2000 V09 N8: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of August 1990. * Oak Ridge National Laboratory. September 1990. 88pp. 9010090032. ORNL/NSIC-200. 55319:255.
See NUREG/CR-2000,V08,N12 abstract.

NUREG/CR-2000 V09 N9: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of September 1990. * Oak Ridge National Laboratory. October 1990. 114pp. 9011150041. ORNL/NSIC-200. 55737:071.
See NUREG/CR-2000,V08,N12 abstract.

NUREG/CR-2000 V09N10: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of October 1990. * Oak Ridge National Laboratory. November 1990. 96pp. 9012100347. ORNL/NSIC-200. 55990:215.
See NUREG/CR-2000,V08,N12 abstract.

NUREG/CR-2000 V09N11: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of November 1990. * Oak Ridge National Laboratory. December 1990. 73pp. 9101160011. ORNL/NSIC-200. 56428:060.
See NUREG/CR-2000,V08,N12 abstract.

NUREG/CR-2331 V09 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Progress Report,July-September 1989. WEISS,A.J. Brookhaven National Laboratory. February 1990. 104pp. 9003070194. BNL-NUREG-51454. 52812:282.
This progress report describes current activities and technical progress in the programs at Brookhaven National Laboratory sponsored by the Division of Regulatory Applications, Division of Engineering, Division of Safety Issue Resolution, and Division of Systems Research of the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research following the reorganization in July 1988. The previous reports have covered the period October 1, 1976 through June 30, 1989.

NUREG/CR-2331 V09 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.October-December 1989. WEISS,A.J. Brookhaven National Laboratory. June 1990. 140pp. 9007120223. BNL-NUREG-51454. 54466:291.
This progress report describes current activities and technical progress in the programs at Brookhaven National Laboratory sponsored by the Division of Regulatory Applications, Division of Engineering, Division of Safety Issue Resolution, and Division of Systems Research of the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research following the reorganization in July 1988. The previous reports have covered the period October 1, 1976 through September 30, 1989.

NUREG/CR-2601 ADD01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE LIGHT WATER REACTORS FOLLOWING POSTULATED ACCIDENTS. Re-Evaluation Of The Cleanup Cost For The Boiling Water Reactor (BWR) Scenario 3 Accident From NUREG/CR-2601. KONZEK,G.J.; SMITH,R.I. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1990. 54pp. 9012280218. 56256:018.
The estimated costs for post-accident cleanup at the reference BWR (developed previously in NUREG/CR-2601, "Technology, Safety, and Costs of Decommissioning Reference Light Water Reactors Following Postulated Accidents") are updated to January 1989 dollars in this report. A simple formula for escalating post-accident cleanup costs is also presented. Accident cleanup following the most severe accident described in NUREG/CR-2601 (i.e., the Scenario 3 accident) is estimated to cost from $1.22 to $1.44 billion, in 1989 dollars, for assumed escalation rates of 4% or 8% in the years following 1989. The time to accomplish cleanup remained unchanged from the 6.3 years originally estimated. No reanalysis of current information on the technical aspects of TMI-2 cleanup has been performed. Only the cost of inflation has been evaluated since the original PNL analysis was completed.

NUREG/CR-2850 V09: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1987. BAKER,D.A. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1990. 187pp. 9009120104. PNL-4221. 55194:282.
Population radiation dose commitments have been estimated from reported radionuclide releases from commercial power reactors operating during 1987. Fifty-year dose commitments from a one-year exposure were calculated from both liquid and atmospheric releases for four population groups (infant, child, teen-ager and adult) residing between 2 and 80 km from each of 70 sites. This report tabulates the results of these calculations, showing the dose commitments for both liquid and airborne pathways for each age group and organ. Also included

for each of the sites is a histogram showing the fraction of the total population within 2 to 80 km around each site receiving various average dose commitments from the airborne pathways. The total dose commitments (from both liquid and airborne pathways) for each site ranged from a high of 15 person-rem to a low of 0.0016 person-rem for the sites with plants operating throughout the year with an arithmetic mean of 1.1 person-rem. The total population dose for all sites was estimated at 78 person-rem for the 150 million people considered at risk.

NUREG/CR-3145 V08: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION.Annual Report,October 1988 - September 1989. YOUNG,C.J.; LAY,T.; JACOBSON,J. Michigan, Univ. of, Ann Arbor, MI. February 1990. 62pp. 9003070254. 52802:257.

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. Three local earthquakes have been recorded during this report period, with magnitudes ranging from 0.7 to 2.2. This may represent a return to a more normal local level of seismicity after an anomalously low level in the 2 years following the occurrence of the m(b) = 4.5 earthquake, in St. Marys, Ohio on July 12, 1986. If the quiescent period reflected a release of most of the accumulated crustal strain by the St. Marys event, then the increased level of seismicity may mark the beginning of a new phase of strain accumulation. Four regional events were also well recorded by the array stations during this year. All of these events occurred in regions with well-established histories of seismicity. Their magnitudes range from 3.1 to 4.4.

NUREG/CR-3444 V07: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION, WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE. SOO,P.; MILIAN,L.W. Brookhaven National Laboratory. July 1990. 74pp. 9008070400. BNL-NUREG-51699. 54844:188.

Studies were carried out to investigate if simulated decontamination reagent/resin waste combinations could give rise to gas generation and thermal excursions during dewatering events. The results of temperature measurements and visual observations are given. Some limited work was also carried out to determine if gamma irradiation of ion-exchange resins causes structural changes and losses in ion-exchange capacity. In addition, the corrosion of various container materials in simulated decontamination resin waste was studied. In particular, the effects of gamma irradiation were quantified.

NUREG/CR-3469 V05: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA. KHAN,T.A.; TAN,H.; BAUM,J.W.; et al. Brookhaven National Laboratory. September 1990. 95pp. 9010090029. BNL-NUREG-51708. 55319:160.

One of the functions of the ALARA Center is to collect and disseminate information on dose reduction at nuclear power plants. This is the fifth 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 278 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-3668: MINET CODE DOCUMENTATION. VAN TUYLE,G.J.; NEPSEE,T.C.; GUPPY,J.G. Brookhaven National Laboratory. December 1989. 321pp. 9002120129. BNL-NUREG-51742. 52582:219.

The MINET computer code, developed for the transient analysis of fluid flow and heat transfer, is documented in this four-part reference. In Part 1, the MINET models, which are based on a momentum integral network method, are described. The various aspects of utilizing the MINET code are discussed in Part 2, The User's Manual. The third part is a code description, detailing the basic code structure and the various subroutines and functions that make up MINET. In Part 4, example input decks, as well as recent validation studies and applications of MINET are summarized.

NUREG/CR-3873: EFFECT OF WELDING CONDITIONS ON TRANSFORMATION AND PROPERTIES OF HEAT-AFFECTED ZONES ON LWR VESSEL STEELS. LUNDIN,C.D.; MOHAMMED,S. Tennessee, Univ. of, Knoxville, TN. * Oak Ridge National Laboratory. November 1990. 148pp. 9012100339. ORNLSUB787637/1. 55991:106.

The continuous cooling transformation and isothermal transformation behaviors were determined for SA-508 and SA-533 materials for conditions pertaining to the standard heat treatment and for the coarse-grained region of the heat-affected zone. The resulting diagrams help to select welding conditions that produce the most favorable microconstituent. Recommendations are provided regarding weld energy input and postweld heat treatment conditions to optimize heat-affected zone toughness. The reheat cracking tendency for both steels was evaluated by metallographic studies of simulated heat-affected zone structures subjected to postweld heat treatment cycles and simultaneous restraint. Both SA-533 grade B class 1, and SA-508 class 2 steels cracked intergranularly. The stress rupture parameter showed that SA-508 class 2 was more susceptible to reheat cracking than SA-533 grade B class 1. Cold cracking tests (Battelle Test and University of Tennessee modified hydrogen susceptibility test) indicated that a higher preheat temperature is required for SA-508 class 2 to avoid cracking than is required for SA-533 grade B class 1. Further, the hydrogen susceptibility tests showed that the SA-508 is more susceptible to hydrogen embrittlement than the SA-533.

NUREG/CR-3950 V06: FUEL PERFORMANCE ANNUAL REPORT FOR 1988. BAILEY,W.J. Battelle Memorial Institute, Pacific Northwest Laboratory. WU,S. Reactor Systems Branch. March 1990. 175pp. 9005020352. PNL-5210. 53589:223.

This annual report, the eleventh in a series, provides a brief description of fuel performance during 1988 in commercial nuclear power plants. Brief summaries of fuel design changes, fuel surveillance programs, fuel operating experience and trends, fuel problems, high-burnup fuel experience, and items of general significance are provided. References to additional, more detailed information and related NRC evaluations are included.

NUREG/CR-4214 R01 PI: HEALTH EFFECTS MODELS FOR NUCLEAR POWER PLANT ACCIDENT CONSEQUENCE ANALYSIS.Low LET Radiation.Part I:Introduction, Integration And Summary. EVANS,J.S. Harvard School of Public Health, Boston, MA. * Sandia National Laboratories. January 1990. 133pp. 9002120313. SAND85-7185. 52597:294.

This report provides dose-response models intended to be used in estimating the radiological health effects of nuclear power plant accidents. Models of early and continuing effects, cancers and thyroid nodules, and genetic effects are provided. Two-parameter Weibull hazard functions are recommended for estimating the risks of early and continuing health effects. Three potentially lethal early effects--the hematopoietic, pulmonary and gastrointestinal syndromes--are considered. In addition, models are provided for assessing the risks of several nonlethal early and continuing effects--including prodromal vomiting and diarrhea, hypothyroidism and radiation thyroiditis, skin burns, reproductive effects, and spontaneous abortions. Linear and linear-quadratic models are recommended for estimating cancer risks. Parameters are given for analyzing the risks of seven types of cancer in adults-- leukemia, bone, lung, breast, gastrointestinal, thyroid and "other". The category "other" cancers, is intended to reflect the combined risks of multiple myeloma, lymphoma, and cancers of the bladder, kidney, brain, ovary, uterus and cervix. Models of childhood cancers due to "in utero" exposure are also provided. For most cancers, both incidence and mortality are addressed. The models of cancer risk are derived largely from information summarized in BEIR III---with some adjustment to reflect more recent studies. The effect of the revised dosimetry in Hiroshima and Nagasaki has not been considered. Linear and linear-quadratic models are also recommended for assessing genetic risks. Five classes of genetic disease--dominant, x-linked, aneuploidy, unbalanced translocations and multifactorial diseases--are considered. In addition, the impact of radiation-induced genetic damage on the incidence of peri-implantation embryo losses is discussed. The uncertainty in modeling radiological health risks is addressed by providing central, upper, and lower estimates of all model parameters. Data are provided which should enable analysts to consider the timing and severity of each type of health risk.

NUREG/CR-4219 V06 N2: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For April-September 1989. PENNELL,W.E. Oak Ridge National Laboratory. August 1990. 96pp. 9011090211. ORNL/TM-9593. 55660:156.

The Heavy-Section Steel Technology (HSST) Program studies concern all areas of the technology of materials fabricated into thick-section, primary-coolant containment systems of light-water-cooled nuclear power reactors. The focus is on the behavior and structural integrity of steel reactor pressure vessels (RPVs) containing cracklike flaws. During this period, analytical efforts included examining the influence of high crack-arrest toughness on RPV integrity and an increased emphasis on evaluating large international structural experiments. Two areas of NRC topical support were continued: (1) the evaluation of mechanisms for enhanced low-temperature, low-flux irradiation embrittlement that may affect the integrity of RPV supports; and (2) an overall assessment of low upper-shelf (LUS) welds in RPVs with special emphasis on reevaluating ductile tearing criteria. The first four stub-panel crack-arrest tests were performed. Posttest material characterization was performed for clad-plate and wide-plate Series 2 test materials. Statistical analyses were performed on the data from the Fifth HSST Irradiation Series on the study of K(Ic) shifts. Analysis of the irradiated fracture-toughness testing was completed for the Seventh HSST Irradiation Series on cladding. Detailed planning was begun for the next pressurized-thermal-shock experiment, PTSE-4, to examine the extent of ductile tearing and its interaction with cleavage fracture in an LUS weld metal.

NUREG/CR-4469 V08: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report,October 1987 - March 1988. DOCTOR,S.R.; DEFFENBAUGH,J.; GOOD,M.S.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1989. 67pp. 9002120167. PNL-5711. 52603:122.

The Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors (NDE Reliability) Program at the Pacific Northwest Laboratory was established by the Nuclear Regulatory Commission to determine the reliability of current inservice inspection (ISI) techniques and to develop recommendations that will ensure a suitably high inspection reliability. The objectives of this program include determining the reliability of ISI performed on the primary systems of commercial light-water reactors (LWRs); using probabilistic fracture mechanics analysis to determine the impact of NDE unreliability on system safety; and evaluating reliability improvements that can be achieved with improved and advanced technology. A final objective is to formulate recommended revisions to ASME Code and Regulatory requirements, based on material properties, service conditions, and NDE capabilities and uncertainties. The program scope is limited to ISI of the primary systems including the piping, vessel, and other inspected components. This is a progress report covering the programmatic work from October 1987 through March 1988.

NUREG/CR-4469 V09: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report,April-September 1988. DOCTOR,S.R.; DEFFENBAUGH,J.; GOOD,M.S.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1989. 115pp. 9002120296. PNL-5711. 52597:100.

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 inspected components. This is a progress report covering the programmatic work from April 1988 through September 1988.

NUREG/CR-4469 V10: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report,October 1988 - March 1989. DOCTOR,S.R.; DEFFENBAUGH,J.; GOOD,M.S.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1990. 96pp. 9010230079. PNL-5711. 55472:181.

The Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors (NDE Reliability) Program at the Pacific Northwest Laboratory was established by the Nuclear Regulatory Commission to determine the reliability of current inservice inspection (ISI) techniques and to develop recommendations that will ensure a suitably high inspection reliability. The objectives of this program include determining the reliability of ISI performed on the primary systems of commercial light-water reactors (LWRs); using probabilistic fracture mechanics analysis to determine the impact of NDE unreliability on system safety; and evaluating reliability improvements that can be achieved with improved and advanced technology. A final objective is to formulate recommended revisions to ASME Code and Regulatory requirements, based on material properties, service conditions, and NDE uncertainties. The program scope is limited to ISI of the primary systems including the piping, vessel, and other components inspected in accordance with Section XI of the ASME Code. This is a progress report covering the programmatic work from October 1988 through March 1989.

NUREG/CR-4525: CLOSEOUT OF IE BULLETIN 84-03:REFUELING CAVITY WATER SEAL. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. June 1990. 88pp. 9007120236. PARAMETER IE155. 54468:336.

Documentation is provided in this report to close IE Bulletin 84-03 on the subject of refueling cavity water seals. The bulletin was issued on August 24, 1984, to all power reactor facilities except Fort St. Vrain. Because the excluded plant is gas-cooled and graphite-moderated, the subject of the bulletin does not apply to that plant. The event causing the safety concern was failure of the pneumatic seals containing water in the refueling cavity at Haddam Neck on August 21, 1984. The cavity had been flooded in preparation for refueling, but when the seals failed, water level decreased from full level just below the operating floor down to the level of the reactor vessel flange. On May 17, 1988, another refueling cavity seal leakage (about three feet in water level) occurred at Surry 1, which was in the middle of a refueling and maintenance outage. Evaluation of utility responses and NRC/Region inspection reports in accordance with a specific criterion shows that the bulletin is closed for 114 (97%) of the 118 affected facilities. It is concluded that the concerns of the bulletin have been resolved through design and procedure reviews and corrective actions taken by licensees. Follow-up items are proposed for use by NRC Region II in closing the bulletin for the four (4) remaining facilities with open bulletin status. NRC Region II verifies that the bulletin will not be closed for these four TVA facilities until construction is completed. Background information is supplied in the Introduction and Appendix A.

NUREG/CR-4550 V01 R1: ANALYSIS OF CORE DAMAGE FREQUENCY: INTERNAL EVENTS METHODOLOGY. ERICSON,D.M. ERC Environmental & Energy Services, Inc. WHEELER,T.A.; SYPE,T.T.; et al. Sandia National Laboratories. January 1990. 482pp. 9002120309. SAND86-2084. 52576:007.

NUREG-1150 examines the risk to the public from a selected group of nuclear power plants. This report describes the methodology that evolved as the internal event core damage frequencies for four plants were generated in support of NUREG-1150. The objective is to perform an analysis that closely approximates a state-of-the-art Level I Probabilistic Risk Assessment (PRA). Therefore, in principle, it is similar to those used in previous PRAs. However, this methodology, based upon previous studies and using analysts experienced in these techniques, allows the analysis to be focused upon selected areas. With this approach only the most important systems and failure modes are emphasized and modeled in detail, and the data and human reliability analyses are simplified. An analysis employing this methodology (exclusive of external reviews) can be completed in nine to twelve months using two or three full-time experienced systems analysts and part-time personnel in other areas, such as data analysis and human reliability analysis. This is significantly faster and less expensive than previous analyses, but even so, most of the insights that are obtained by the more expensive studies are still provided.

NUREG/CR-4550 V3R1P1: ANALYSIS OF CORE DAMAGE FREQUENCY:SURRY,UNIT 1,INTERNAL EVENTS. BERTUCIO,R.C.; JULIUS,J.A. EI Services, Inc. * Sandia National Laboratories. April 1990. 484pp. 9006080175. SAND86-2084. 54078:003.

This document contains the accident sequence analyses of internally initiated events for the Surry Nuclear Station, Unit 1. This is one of the five plant analyses conducted as part of the NUREG-1150 effort by the Nuclear Regulatory Commission. NUREG-1150 documents the risk of a selected group of nuclear power plants. The work performed and described here is an extensive reanalysis of that published in November 1986 as NUREG/CR-4550, Volume 3. It addresses comments from numerous reviewers and significant changes to the plant systems and procedures made since the first report. The uncertainty analysis and presentation of results are also much improved. The context and detail of this report are directed toward PRA practitioners who need to know how the work was performed and the details for use in further studies. The mean core damage frequency at Surry was calculated to be 4.0E-5 per year, with a 95% upper bound of 1.3E-4 and 5% lower bound of 6.8E-6 per year. Station blackout type accidents (loss of all AC power) were the largest contributors to the core damage frequency, accounting for approximately 68% of the total. The next type of dominant contributors were Loss of Coolant Accidents (LOCAs). These sequences account for 15% of core damage frequency. No other type of sequence accounts for more than 10% of core damage frequency.

NUREG/CR-4550 V3R1P2: ANALYSIS OF CORE DAMAGE FREQUENCY: SURRY,UNIT 1,INTERNAL EVENTS APPENDICES. BERTUCIO,R.C.; JULIUS,J.A. EI Services, Inc. * Sandia National Laboratories. April 1990. 702pp. 9006080169. SAND86-2084. 54079:127.

See NUREG/CR-4550,V03,R01,P01 abstract.

NUREG/CR-4550 V3R1P3: ANALYSIS OF CORE DAMAGE FREQUENCY: SURRY POWER STATION,UNIT 1 EXTERNAL EVENTS. BOHN,M.P.; LAMBRIGHT,J.A.; DANIEL,S.L.; et al. Sandia National Laboratories. December 1990. 393pp. 9101090373. SAND86-2084. 56298:285.

This report presents the analysis of external events (earthquakes, fires, floods, etc.) performed for the Surry Power Station as part of the USNRC-sponsored NUREG-1150 program. Both the internal and external events analyses make full use of recent insights and developments in risk assessment methods. In addition, the external event analyses make use of newly-developed simplified methods. As a first step, a screening analysis was performed which showed that all external events were negligible except for fires and seismic events. Subsequent detailed analysis of fires resulted in a total (mean) core damage frequency of 1.13E-5 per year. The seismic analysis resulted in a total (mean) core damage frequency of 1.16E-4 per year using hazard curves developed by Lawrence Livermore National Laboratory and 2.50E-5 per year using hazard curves developed by the Electric Power Research Institute. Uncertainty analyses were performed, and dominant components and sources of uncertainty were identified.

NUREG/CR-4550 V4R1P3: ANALYSIS OF CORE DAMAGE FREQUENCY: PEACH BOTTOM,UNIT 2 EXTERNAL EVENTS. LAMBRIGHT,J.A.; BOHN,M.P.; DANIEL,S.L.; et al. Sandia National Laboratories. December 1990. 471pp. 9101090410. SAND86-2084. 56313:086.

This report presents the analysis of external events (earthquakes, fires, floods, etc.) performed for the Peach Bottom Atomic Power Station as part of the USNRC-sponsored NUREG-1150 program. Both the internal and external events analyses make full use of recent insights and developments in risk assessment methods. In addition, the external event analyses make use of newly-developed simplified methods. As a first step, a screening analysis was performed which showed that all external events were negligible except for fires and seismic events. Subsequent detailed analysis of fires resulted in a total (mean) core damage frequency of 1.95E-5 per year. The seismic analysis resulted in a total (mean) core damage frequency of 7.66E-5 per year using hazard curves developed by Lawrence Livermore National Laboratory and 3.09E-6 per year using hazard curves developed by the Electric Power Research Institute. Uncertainty analyses were performed, and dominant components and sources of uncertainty were identified.

NUREG/CR-4550 V5R1P1: ANALYSIS OF CORE DAMAGE FREQUENCY: SEQUOYAH,UNIT 1,INTERNAL EVENTS. BERTUCIO,R.C.; BROWN,S.R. EI Services, Inc. * Sandia National Laboratories. April 1990. 422pp. 9005210262. SAND86-2084. 53890:118.

This document contains the accident sequence analyses of internally initiated events for the Sequoyah, Unit 1 nuclear power plant. This is one of the five plant analyses conducted as part of the NUREG-1150 effort by the Nuclear Regulatory Commission (NRC). NUREG-1150 documents the risk of a selected group of nuclear power plants. The work performed and described here is an extensive reanalysis of that published in February 1987 as NUREG/CR-4550, Volume 5. It addresses comments from numerous reviewers and significant changes to the plant systems and procedures made since the first report. The uncertainty analysis and presentation of results are also much improved. The mean core damage frequency at Sequoyah was calculated to be 5.7E-5 per year, with a 95 percent upper bound of 1.8E-4 and 5 percent lower bound of 1.2E-5 per year. Loss-of-coolant type accidents were the largest contributors to core damage frequency, accounting for approximately 62 percent of the total. The next most dominant type of accidents were station blackout (loss of all AC power), which account for 26 percent of core damage frequency.

NUREG/CR-4550 V5R1P2: ANALYSIS OF CORE DAMAGE FREQUENCY: SEQUOYAH,UNIT 1,INTERNAL EVENTS APPENDICES. BERTUCIO,R.C.; BROWN,S.R. EI Services, Inc. * Sandia National Laboratories. April 1990. 686pp. 9005210142. SAND86-2084. 53862:001.

This document contains the appendices for the accident sequence analyses of internally initiated events for the Sequoyah, Unit 1 nuclear power plant. This is one of the five plant analyses conducted as part of the NUREG-1150 effort by the Nuclear Regulatory Commission (NRC). NUREG-1150 will document the risk of a selected group of nuclear power plants. The work performed and described here is an extensive reanalysis of that published in February 1987 as NUREG/CR-4550, Volume 5. It addresses comments from numerous reviewers and significant changes to the plant systems and procedures made since the first report. The uncertainty analysis and presentation of results are also much improved. The mean core damage frequency at Sequoyah was calculated to be the 5.7E-5 per year, with a 95 percent upper bound of 1.8E-4 and 5 percent lower bound of 1.2E-5 per year. Loss-of-coolant type accidents were the largest contributors to core damage frequency, accounting for approximately 62 percent of the total. The next most dominant type of accidents were station blackout (loss of all AC power), which account for 26 percent of core damage frequency.

NUREG/CR-4550 V7 R01: ANALYSIS OF CORE DAMAGE FREQUENCY: ZION, UNIT 1, INTERNAL EVENTS. SATTISON,M.B.; HALL,K.W. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1990. 402pp. 9006290164. EGG-2554. 54325:223.

This document contains the accident sequence analyses of internally initiated events for the Zion Unit 1 Nuclear Power Plant. This is one of the five plant analyses conducted as part of the NUREG-1150 effort for the Nuclear Regulatory Commission. The work performed and described here is an extensive reanalysis of the work published in October 1986 as NUREG/CR-4550 Volume 7. It addresses comments from numerous reviewers and provides significantly more detailed modeling of most aspects of the Zion plant. The mean core damage frequency at Zion was calculated to be 3.4E-4 per year, with a 95% upper bound of 8.4E-4 per year and a 5% lower bound of 1.1E-4 per year. Note that this range is based only on the statistical treatment of six specific issues. Reactor coolant pump seal loss-of-coolant accidents (LOCAs) were the largest contributors to the core damage frequency, accounting for approximately 85% of the total.

NUREG/CR-4551 V2R1P1: EVALUATION OF SEVERE ACCIDENT RISKS: QUANTIFICATION OF MAJOR INPUT PARAMETERS.Expert Opinion Elicitation On In-Vessel Issues. HARPER,F.T.; BREEDING,R.J.; BROWN,T.D.; et al. Sandia National Laboratories. December 1990. 286pp. 9101090429. SAND86-1309. 56297:075.

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 initiation by events, both internal to the power station and external to the power station, were 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 In-Vessel Expert Panel.

NUREG/CR-4551 V2R1P7: EVALUATION OF SEVERE ACCIDENT RISKS: QUANTIFICATION OF MAJOR INPUT PARAMETERS.MACCS Input. SPRUNG,J.L. Sandia National Laboratories. ROLLSTIN,J.A. Gram, Inc. HELTON,J.C.; et al. Arizona State Univ., Tempe, AZ. December 1990. 252pp. 9101090454. SAND86-1309. 56298:001.

Estimation of offsite accident consequences is the customary final step in a probabilistic assessment of the risks of severe nuclear reactor accidents. Recently, the Nuclear Regulatory Commission reassessed the risks of severe accidents at five U.S. power reactors (NUREG-1150). Offsite accident consequences for NUREG-1150 source terms were estimated using the MELCOR Accident Consequence Code System (MACCS). Before these calculations were performed, most MACCS input parameters were reviewed, and for each parameter reviewed, a best-estimate value was recommended. This report presents the results of these reviews. Specifically, recommended values and the basis for their selection are presented for MACCS atmospheric and biospheric transport, emergency response, food pathway, and economic input parameters. Dose conversion factors and health effect parameters are not reviewed in this report.

NUREG/CR-4551 V3R1P1: EVALUATION OF SEVERE ACCIDENT RISKS: SURRY UNIT 1.Main Report. BREEDING,R.J.; HELTON,J.C.; MURFIN,W.B.; et al. Sandia National Laboratories. October 1990. 511pp. 9011260018. SAND86-1309. 55805:027.

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 the Surry Power Station, Unit 1. This power plant, located in southeastern Virginia, is operated by the Virginia Electric Power Corp. The emphasis in this risk analysis was not on determining a "so-called" point estimate of risk. Rather, it was to determine the distribution of risk, and to discover the uncertainties that account for the breadth of this distribution. Off-site risk initiation by events, both internal to the power station and external to the power station were assessed.

NUREG/CR-4551 V3R1P2: EVALUATION OF SEVERE ACCIDENT RISKS: SURRY UNIT 1. Appendices. BREEDING,R.J.; HELTON,J.C.; MURFIN,W.B.; et al. Sandia National Laboratories. October 1990. 386pp. 9011260020. SAND86-1309. 55804:001.

See NUREG/CR-4551,V03,R01,P01 abstract.

NUREG/CR-4551 V4R1P1: EVALUATION OF SEVERE ACCIDENT RISKS: PEACH BOTTOM,UNIT 2. Main Report. PAYNE,A.C.; BREEDING,R.J.; JOW,H-N.; et al. Sandia National Laboratories. December 1990. 713pp. 9101090481. SAND86-1309. 56301:036.

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 the Peach Bottom Atomic Power Station, Unit 2. This power plant, located in southeastern Pennsylvania, is operated by the Philadelphia Electric Company. The emphasis in this risk analysis was not on determining a "so-called" point estimate of risk. Rather, it was to determine the distribution of risk, and to discover the uncertainties that account for the breadth of this distribution. Off-site risk initiated by events both internal and external to the power station were assessed.

NUREG/CR-4551 V4R1P2: EVALUATION OF SEVERE ACCIDENT RISKS: PEACH BOTTOM,UNIT 2. Appendices. PAYNE,A.C.; BREEDING,R.J.; JOW,H-N.; et al. Sandia National Laboratories. December 1990. 438pp. 9101090470. SAND86-1309. 56299:318.

See NUREG/CR-4551,V04,R01,P01 abstract.

NUREG/CR-4551 V5R1P1: EVALUATION OF SEVERE ACCIDENT RISKS: SEQUOYAH,UNIT 1.Main Report. GREGORY,J.J.; HIGGINS,S.J.; et al. Sandia National Laboratories. MURFIN,W.B. Technadyne Engineering Consultants, Inc. December 1990. 307pp. 9101140293. SAND86-1309. 56377:303.

In support of the U.S. Nuclear Regulatory Commission'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 has completed a revised calculation of the risk to the general public from severe accidents at the Sequoyah Power Station, Unit 1. This power plant, located in southeastern Tennessee, is operated by the Tennessee Valley Authority. 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 risks from initiating events internal to the power station were assessed.

NUREG/CR-4551 V5R1P2: EVALUATION OF SEVERE ACCIDENT RISKS: SEQUOYAH,UNIT 1. Appendices. GREGORY,J.J.; HIGGINS,S.J.; et al. Sandia National Laboratories. MURFIN,W.B. Technadyne Engineering Consultants, Inc. December 1990. 464pp. 9101140299. SAND86-1309. 56376:179.

See NUREG/CR-4551,V05,R01,P01 abstract.

NUREG/CR-4551 V6R1P1: EVALUATION OF SEVERE ACCIDENT RISKS: GRAND GULF,UNIT 1.Main Report. BROWN,T.D.; BREEDING,R.J.; JOW,H.-N.; et al. Sandia National Laboratories. December 1990. 305pp. 9101140307. SAND86-1309. 56375:234.

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 the Grand Gulf Nuclear Station, Unit 1. This power plant, located in Port Gibson, Mississippi, is operated by the System Energy Resources, Inc. (SERI). 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 internal to the power plant was assessed.

NUREG/CR-4551 V6R1P2: EVALUATION OF SEVERE ACCIDENT RISKS: GRAND GULF,UNIT 1. Appendices. BROWN,T.D.; BREEDING,R.J.; JOW,H.-N.; et al. Sandia National Laboratories. December 1990. 533pp. 9101140337. SAND86-1309. 56378:250.

See NUREG/CR-4551,V06,R01,P01 abstract.

NUREG/CR-4554 V06: SCANS (SHIPPING CASK ANALYSIS SYSTEM):A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR SHIPPING CASK DESIGN REVIEW.Volume 6 - Theory Manual Buckling Of Circular Cylindrical Shells. LO,T.; MOK,G.C.; CHINN,D.J. Lawrence Livermore National Laboratory. February 1990. 76pp. 9004030139. UCID-20674. 53228:173.

A computer system called SCANS (Shipping Cask ANalysis System) is being developed for the staff of the U.S. Nuclear Regulatory Commission to perform confirmatory licensing review analyses. SCANS can handle problems associated with impact, heat transfer, thermal stress, internal or external pressure loads, and lead slump. A new capability implemented in SCANS is buckling analysis of the steel shells of a spent fuel shipping cask during a postulated impact with an unyielding surface. Three sets of buckling analysis formulas are included: (1) Code Case N-284 of the ASME Boiler and Pressure Vessel Code, (2) American Petroleum Institute Bulletin 2U an upgrade of N-284 that includes test results available after N-284 was written in 1979, (3) formulas frequently used by the piping and pressure vessel industry and formulas recommended by the Structural Stability Research Council. To be compatible with the ASME Code, the first set is recommended for use in shipping cask evaluation and this set is implemented in SCANS. The second and third sets are recommended references for SCANS users.

NUREG/CR-4554 V07: SCANS (SHIPPING CASK ANALYSIS SYSTEM) A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR SHIPPING CASK DESIGN REVIEW.Volume 7: Theory Manual Puncture Of Shipping Casks. LO,T. Lawrence Livermore National Laboratory. February 1990. 65pp. 9004030060. UCID-20674. 53226:100.

Under current regulatory requirements, a shipping cask should be designed for a series of hypothetical accident conditions. These test conditions include a 40-inch free drop of the cask onto a 6-inch diameter puncture pin. In this study, existing puncture test data was examined. Simple formulas based on test data were proposed for puncture evaluation of shipping casks. Dynamic and static nonlinear finite element analyses were performed to correlate analyses with the existing test data. In this analytical approach, three puncture failure prediction methods were proposed and their applicability was evaluated. The analytical approach provides an alternative to testing. Both laminated and solid wall shipping casks were analysed. In the study of laminated casks, the effects of the inner shell on the puncture of the outer shell were examined, as were the effects of material strength of the puncture pin. The study of geometric scaling of casks indicated that the normalized incipient puncture energy is insensitive to variations in the scale factors. This conclusion indicates that the proposed analytical approach of combining finite element analysis and failure prediction methods is consistent with the similarity principles in tests, except when local vibration is excited during the puncture process. Further study of this local vibration is needed.

NUREG/CR-4624 V06: RADIONUCLIDE RELEASE CALCULATIONS FOR SELECTED SEVERE ACCIDENT SCENARIOS.Supplemental Calculations. DENNING,R.S.; LEONARD,M.T.; CYBULSKIS,P.; et al. Battelle Memorial Institute, Columbus Laboratories. August 1990. 385pp. 9009200004. BMI-2139. 55?16:135.

The results of source term calculations are reported. These calculations were performed in support of the NUREG-1150 study. Analyses were performed for three plants: Peach Bottom, a Mark I, boiling water reactor; Surry, a subatmospheric containment, pressurized water reactor; and Sequoyah, a subatmospheric containment, pressurized water reactor. Complete source term results are presented for the following sequences: short term station blackout with failure of the ADS system in the Peach Bottom plant; station blackout with pump seal LOCA in the Surry plant; station blackout with a pump seal LOCA in the Sequoyah plant; and a very small break with loss of ECC and spray recirculation in the Sequoyah plant. In addition, some partial analyses were performed which did not require running all of the modules of the Source Term Code Package. Thermal-hydraulic calculations were performed for the Surry and Sequoyah plants to evaluate the effects of alternative emergency operating procedures involving primary and secondary depressurization. For the Surry plant, calculations were performed of radionuclide transport through the primary system during accident-induced failure of steam generator tubes.

NUREG/CR-4639 V01 R1: NUCLEAR COMPUTERIZED LIBRARY FOR ASSESSING REACTOR RELIABILITY (NUCLARR).Summary Description. GERTMAN,D.I.; GILMORE,W.E.; GALYEAN,W.J.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1990. 29pp. 9006080202. EGG-2458. 54074:042.

The Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR) is an automated data base management system for storing and processing human error probability and hardware component failure rate data. The NUCLARR system software resides on an IBM (or compatible) personal microcomputer. NUCLARR can be accessed by the end user to furnish data suitable for input in human and/or hardware reliability analysis to support a variety of risk assessment activities. The NUCLARR system is documented in a five-volume series of reports. Volume 1 of this series is the Summary Description, which presents an overview of the data management system, including a description of data collection, data qualification, data structure, and taxonomies. Programming activities, procedures for processing data, a user's guide, and hard copy data manual are presented in Volumes 2 through 5, NUREG/CR-4639.

NUREG/CR-4639 V4P1R2: NUCLEAR COMPUTERIZED LIBRARY FOR ASSESSING REACTOR RELIABILITY (NUCLARR).User's Guide.Part 1: Overview Of NUCLARR Data Retrieval. GILMORE,W.E.; GENTILLON,C.D.; GERTMAN,D.I.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). October 1990. 40pp. 9010250026. EGG-2458. 55496:050.

The Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR) is an automated data base management system for processing and storing human error probability and hardware component failure data. The NUCLARR system software resides on an IBM (or compatible) personal microcomputer. NUCLARR can be used by the end user to furnish data inputs for both human and hardware reliability analysis in support of a variety of risk assessment activities. The NUCLARR system is documented in a five-volume series of reports. Volume 4: User's Guide is presented in three parts. Part 1: Overview of NUCLARR Data Retrieval, provides an introductory overview to the system's capabilities and procedures for data retrieval. Part 2: Guide to Operations contains the instructions and basic procedures for using the NUCLARR software. Part 3: NUCLARR System Description provides in-depth discussion of the design characteristics and special features of the NUCLARR software.

NUREG/CR-4639 V4P2R2: NUCLEAR COMPUTERIZED LIBRARY FOR ASSESSING REACTOR RELIABILITY (NUCLARR).User's Guide.Part 2: Guide To Operations. GILMORE,W.E.; GENTILLON,C.D.; GERTMAN,D.I.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). October 1990. 141pp. 9010250024. EGG-2458. 55496:083.

See NUREG/CR-4639,V04,P01,R02 abstract.

NUREG/CR-4639 V4P3R2: NUCLEAR COMPUTERIZED LIBRARY FOR ASSESSING REACTOR RELIABILITY (NUCLARR).User's Guide.Part 3: NUCLARR System Description. GILMORE,W.E.; GENTILLON,C.D.; GERTMAN,D.I.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). October 1990. 234pp. 9010250020. EGG-2458. 55496:224.

See NUREG/CR-4639,V04,P01,R02 abstract.

NUREG/CR-4639 V5P1R3: NUCLEAR COMPUTERIZED LIBRARY FOR ASSESSING REACTOR RELIABILITY (NUCLARR).Data Manual.Part 1: Summary Description. GILBERT,B.G.; REECE,W.J.; GERTMAN,D.I.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). December 1990. 24pp. 9101180027. EGG-2458. 56465:304.

The Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR) is an automated data management system for processing and storing human error probability and hardware component failure data. The NUCLARR system software resides on an IBM (or compatible) personal computer. NUCLARR can furnish an end user with data inputs for both human and hardware reliability analysis in support of a variety of risk assessment activities. The NUCLARR system is documented in a five-volume series of reports. Volume 5: "Data Manual" provides a hard-copy representation of all data and related information available with the NUCLARR system software. This document is organized in three sections: Part 1 is the summary description, which presents an overview of the NUCLARR system and data processing procedures. Part 2 contains all data and information relevant to the human error probability (HEP) side of NUCLARR. Data and information for the hardware component failure data (HCFD) side are presented within Part 3.

NUREG/CR-4639 V5P2R3: NUCLEAR COMPUTERIZED LIBRARY FOR ASSESSING REACTOR RELIABILITY (NUCLARR).Data Manual.Part 2: Human Error Probability (HEP) Data. GILBERT,B.G.; REECE,W.J.; GERTMAN,D.I.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). December 1990. 456pp. 9101180037. EGG-2458. 56437:073.

See NUREG/CR-4639,V05,P01,R03 abstract.

NUREG/CR-4639 V5P3R3: NUCLEAR COMPUTERIZED LIBRARY FOR ASSESSING REACTOR RELIABILITY (NUCLARR).Data Manual.Part 3: Hardware Component Failure Data (HCFD). GILBERT,B.G.; REECE,W.J.; GERTMAN,D.I.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). December 1990. 179pp. 9101180025. EGG-2458. 56448:328.

See NUREG/CR-4639,V05,P01,R03 abstract.

NUREG/CR-4659 V03: SEISMIC FRAGILITY OF NUCLEAR POWER PLANT COMPONENTS (PHASE II).Switchgear, I&C Panels (NSSS) And Relays. BANDYOPADHYAY, HOFMAYER,C.H.; KASSIR,M.K.; et al. Brookhaven National Laboratory. February 1990. 73pp. 9004030128. BNL-NUREG-52007. 53228:101.

As part of the Component Fragility Program which was initiated in FY 1985, three additional equipment classes have been evaluated. This report contains the fragility results and discussions on these equipment classes which are switchgear, I&C panels and relays. Both low and medium voltage switchgear assemblies have been considered and a separate fragility estimate for each type is provided. Test data on cabinets from the nuclear instrumentation/neutron monitoring system, plant/process protection system, solid state protective system and engineered safeguards test system comprise the BNL data base for I&C panels (NSSS). Fragility levels have been determined for various failure modes of switchgear and I&C panels, and the deterministic results are presented in terms of test response spectra. In addition, the test data have been evaluated for estimating the respective probabilistic fragility levels which are expressed in terms of a median value, an uncertainty coefficient, a randomness coefficient and an HCLPF value. Due to a wide variation of relay design and the fragility level, a generic fragility level cannot be established for relays.

NUREG/CR-4661: CLOSEOUT OF IE BULLETIN 85-03: MOTOR-OPERATED VALVE COMMON MODE FAILURES DURING PLANT TRANSIENTS DUE TO IMPROPER SWITCH SETTINGS. FOLEY,W.J.; DEAN,R.S.; STEINBRECHER,H.; et al. PARAMETER, Inc. February 1990. 108pp. 9003120769. PARAMETER IE158. 52882:267.

Documentation is provided in this report for the closeout of IE Bulletin 85-03. The purpose of the bulletin was to request licensees to develop and implement programs to ensure that the switch settings on certain safety-related, motor-operated valves are selected, set, and maintained so as to ensure reliable valve operation when valves are subjected to maximum differential pressures expected during normal operation and the design basis events. The report includes documentation and status of the review of Action Item e of the bulletin completed up to the issuance of Generic Letter 89-10 on June 28, 1989. Licensee actions for Action Item e were completed satisfactorily for 102 (86%) of the 119 facilities for which actions were required. Satisfactory completion of the remaining 17 facilities is ensured by the generic letter. In the report conclusions, the interrelationship of the generic letter and the bulletin is presented. Background information is supplied in the introduction and Appendix A.

NUREG/CR-4667 V07: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April-September 1988. SHACK,W.J.; KASSNER,T.F.; PARK,J.Y.; et al. Argonne National Laboratory. March 1990. 53pp. 9007120135. ANL-89/40. 54470:206.

This report summarizes work performed by Argonne National Laboratory on environmentally assisted cracking in light water reactors during the six months from April to September 1988. The stress corrosion cracking (SCC) of Types 316NG and 304 stainless steels (SSs) was investigated by means of slow-strain-rate and fracture-mechanics crack-growth-rate tests in high-temperature water. The effects of load ratio and water chemistry on the crack growth behavior of Type 316NG and sensitized Type 304 SS were determined in long-term fracture-mechanics tests. The influence of organic impurities on the SCC of sensitized Type 304 SS was also investigated. Fatigue tests were conducted on Type 316NG SS in air and simulated boiling water reactor water at 288 degrees C to assess the degree of conservatism in the ASME Code Section III fatigue design curves. An ongoing investigation of the susceptibility of several heats of different grades of low-alloy ferritic steels to transgranular SCC in slow-strain-rate and fracture-mechanics tests was continued.

NUREG/CR-4667 V08: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1988 - March 1989. KASSNER,T.F.; PARK,J.Y.; RUTHER,W.E.; et al. Argonne National Laboratory. June 1990. 53pp. 9007120144. ANL-90/4. 54468:287.

This report summarizes work performed by Argonne National Laboratory on environmentally assisted cracking in light water reactors during the six months from October 1988 to March 1989. The effects of load ratio on stress corrosion cracking (SCC) of Types 316NG, 304, and CF-3M cast stainless steels (SSs) were investigated by fracture-mechanics crack-growth-rate (CGR) tests in high-temperature water. The influence of organic impurities on the SCC of Type 316NG SS was also investigated in long-term CGR tests. Tests to determine the susceptibility of 4-in.-diameter Types 316NG and 304 SS pipe weldments to SCC in simulated BWR environments have been conducted. The influence of carbonate at concentrations between 0.1 and 3300 ppm on the SCC behavior of sensitized Type 304 SS in deoxygenated water (less than 5 ppb) was determined in constant-extension- rate-tensile (CERT) tests. Fatigue tests were conducted on Type 316NG SS in air and BWR environments to assess the degree of conservatism in the ASME Code Section III fatigue design curves. CGR tests to determine susceptibility to SCC are being conducted on A533-Gr B low-alloy ferritic steel in simulated BWR environments.

NUREG/CR-4668: DAMAGED FUEL EXPERIMENT DF-1.Results And Analyses. GASSER,R.D.; FRYER,C.P.; GAUNTT,R.O.; et al. Sandia National Laboratories. January 1990. 269pp. 9003120772. SAND86-1030. 52884:009.

A series of in-pile experiments addressing LWR severe fuel damage phenomena has been conducted in the Annular Core Research Reactor (ACRR) at Sandia National Laboratories. The ACRR Debris Formation and Relocation (DF) experiments are quasi-separate effects tests that provide a data base for the development and verification of models for LWR severe core damage accidents. The first experiment in this series, DF-1, was performed on March 15, 1984, and the results are presented in this report. The DF-1 experiment examined the effects of low initial clad oxidation conditions on fuel damage and relocation processes. The DF-1 test assembly consisted of a nine-rod square-matrix bundle that employed PWR-type fuel rods with a 0.5-m fissile length. The fuel rods were composed of 10% enriched U0(2) pellets within a zircaloy-4 cladding. Steam flowed through the test bundle at flow rates varying between 0.5 and 3 g/s, and the ACRR maintained a peak power level of 1.5 MW during the high temperature oxidation phase of the test inducing about 8.5 kW fission power and about 20 kW peak oxidation power in the assembly. Visual observation showed early clad relocation and partial blockage formation at the grid spacer location accompanied by production of a dense aerosol. Posttest cross sections show liquefaction losses of fuel in excess of 10 volume percent, as well as large fractional losses of cladding material from the upper two-thirds of the bundle. The quantity of hydrogen measured during the test was consistent with the observed magnitude of cladding oxidation. Oxidation driven heating rates of 25 K/s and peak temperatures in excess of 2525 K were observed. The analyses, interpretation, and application of these results to severe fuel damage accidents are discussed.

NUREG/CR-4671: THE DF-4 FUEL DAMAGE EXPERIMENT IN ACRR WITH A BWR CONTROL BLADE AND CHANNEL BOX. GAUNTT,R.O.; GASSER,R.D. Sandia National Laboratories. OTT,L.J. Oak Ridge National Laboratory. November 1989. 397pp. 9003070275. SAND86-1443. 52806:283.

The DF-4 test was an experimental investigation into the melt progression behavior of boiling water reactor (BWR) core components under high temperature severe core damage conditions. In this study 14 zircaloy clad U0(2) fuel rods, and representations of the zircaloy fuel canister and stainless steel/B(4)C control blade were assembled into a 0.5 m long test bundle. The test bundle was fission heated in a flowing steam environment, using the Annular Core Research Reactor at Sandia Laboratories, simulating the environmental conditions of an uncovered BWR core experiencing high temperature damage as a result of residual fission product decay heating. The experimental results provide information on the thermal response of the test bundle components, the rapid exothermic oxidation of the zircaloy fuel cladding and canister, the production of hydrogen from metal-steam oxidation, and the failure behavior of the progressively melting bundle components. This information is provided in the form of thermocouple data, steam and hydrogen flow rate data, test bundle fission power data and visual observation of the damage progression. In addition to BWR background information, this document contains a description of the experimental hardware with details on how the experiment was instrumented and diagnosed, a description of the test progression, and a presentation of the on-line measurements. Also in this report are the results of a thermal analysis of the fueled test section of the experiment demonstrating an overall consistency in the measurable quantities from the test. A discussion of the results is provided.

NUREG/CR-4674 V09: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1988 A STATUS REPORT.Main Report And Appendix A. MINARICK,J.W.; CLETCHER,J.W.; BLAKE,A.A. Oak Ridge National Laboratory. February 1990. 169pp. 9003070234. ORNL/NOAC-232. 52813:323.

Thirty-two operational events with conditional probabilities of core damage of 1.0 x 10(-6) or higher occurring at commercial light-water reactors during 1988 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 1984-1987 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 evaluation process.

NUREG/CR-4674 V10: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1988 A STATUS REPORT.Appendixes B And C. MINARICK,J.W.; CLETCHER,J.W.; BLAKE,A.A. Oak Ridge National Laboratory. February 1990. 473pp. 9003070184. ORNL/NOAC-232. 52813:073.

See NUREG/CR-4674,V09 abstract.

NUREG/CR-4674 V11: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1989 A STATUS REPORT.Main Report And Appendix A. MINARICK,J.W.; CLETCHER,J.W.; COPINGER,D.A.; et al. Oak Ridge National Laboratory. August 1990. 177pp. 9009130048. ORNL/NOAC-232. 55192:068.

Thirty operational events with conditional probabilities of core damage of 1.0 x 10(-6) or higher occurring at commercial light-water reactors during 1989 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 1984-1988 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 V12: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1989 A STATUS REPORT.Appendices B And C. MINARICK,J.W.; CLETCHER,J.W.; COPINGER,D.A.; et al. Oak Ridge National Laboratory. August 1990. 516pp. 9009130058. ORNL/NOAC-232. 55192:245.

See NUREG/CR-4674,V11 abstract.

NUREG/CR-4691 V01: MELCOR ACCIDENT CONSEQUENCE CODE SYSTEM (MACCS).Volume 1: User's Guide. CHANIN,D.I.; SPRUNG,J.L.; RITCHIE,L.T.; et al. Sandia National Laboratories. February 1990. 507pp. 9005040033. SAND86-1562. 54503:136.

This report describes the MACCS computer code. The purpose of this code is to simulate the impact of severe accidents at nuclear power plants on the surrounding environment. MACCS has been developed for the U.S. Nuclear Regulatory Commission to replace the previous CRAC2 code and it incorporates many improvements in modeling flexibility in comparison to CRAC2. The principal phenomena considered in MACCS are atmospheric transport, mitigative actions based on dose projection, dose accumulation by a number of pathways including food and water ingestion, early and latent health effects, and economic costs. The MACCS code can be used for a variety of applications. These include (1) probabilistic risk assessment (PRA) of nuclear power plants and other nuclear facilities, (2) sensitivity studies to gain a better understanding of the parameters important to PRA, and (3) cost-benefit analysis. This report is composed of three volumes. Volume 1, the User's Guide, describes the input data requirements of the MACCS code and provides directions for its use as illustrated by three sample problems. Volume 2, the Model Description, describes the underlying models that are implemented in the code, and Volume 3, the Programmer's Reference Manual, describes the code's structure and database management.

NUREG/CR-4691 V02: MELCOR ACCIDENT CONSEQUENCE CODE SYSTEM (MACCS).Volume 2: Model Description. JOW,H-N.; SPRUNG,J.L.; ROLLSTIN,J.A.; et al. Sandia National Laboratories. February 1990. 186pp. 9003190289. SAND86-1562. 53005:059.

See NUREG/CR-4691,V01 abstract.

NUREG/CR-4691 V03: MELCOR ACCIDENT CONSEQUENCE CODE SYSTEM (MACCS).Volume 3: Programmer's Reference Manual. ROLLSTIN,J.A.; CHANIN,D.I.; JOW,H-N. Sandia National Laboratories. February 1990. 341pp. 9003190293. SAND86-1562. 53042:250.

See NUREG/CR-4691,V01 abstract.

NUREG/CR-4704 V03: RELATIVE BIOLOGICAL EFFECTIVENESS (RBE) OF FISSION NEUTRONS AND GAMMA RAYS AT OCCUPATIONAL EXPOSURE LEVELS.Studies On The Gross And Microscopic Pathology Observed At Death Of Mice Exposed To 60 Equal Once-Weekly Doses Of Fission.... GRAHN,D.; THOMSON,J.F.; CARNES,B.A. Argonne National Laboratory. January 1990. 112pp. 9004090170. ANL-86-33. 53314:010.

Dose-response analyses of the pathologic consequences of exposure to 60 equal once-weekly doses of fission neutrons or 60Co gamma rays were used to generate relative biological effectiveness (RBE) values for the major causes of death or for tumor occurrences. Cumulative probabilities of death or occurrence were generated for 15 categories of neoplastic disease in the interval 800-999 days since first exposure for each dose, sex, and radiation quality. Depending upon the pathologic endpoint, RBE values varied from about 10 to 50. Variation in RBE values was not random or nonsystematic. RBE values were lower for tumors involving the connective tissues (about 10 to 15) than for the epithelial tissue tumors (about 20 to 50). Females did have significantly higher risk coefficients for many endpoints. The results suggest that the RBE value of 20 for life shortening from all causes of death is a weighted average of the RBE values representing the individual neoplastic diseases (approximately 90% of the excess risk of mortality induced by radiation at low doses). These results in conjunction with the results of large-scale studies funded by the Department of Energy and using the same facilities, support the conclusion that the quality factor (Q) for fission neutrons should be raised from the present value of 10 to a value of 20.

NUREG/CR-4731 V02: RESIDUAL LIFE ASSESSMENT OF MAJOR LIGHT WATER REACTOR COMPONENTS - OVERVIEW. SHAH,V.N.; MACDONALD,P.E.; AMAR,A.S.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). November 1989. 467pp. 9002120329. EGG-2469. 52378:023.

This report presents an assessment of the aging (time-dependent degradation) of selected major light water reactor components and structures. The stressors, possible degradation sites and mechanisms, potential failure modes, and current in-service inspection requirements are discussed for eleven major light water reactor components: reactor coolant pumps, pressurized water reactor (PWR) pressurizers, PWR pressurizer surge and spray lines, PWR reactor coolant system charging and safety injection nozzles, PWR feedwater lines, PWR control rod drive mechanisms and reactor internals, boiling water reactor (BWR) containments, BWR feedwater and main steam lines, BWR control rod drive mechanisms and reactor internals, electrical cables and connections, and emergency diesel generators. Unresolved technical issues related to understanding and managing the aging of these major components are identified.

NUREG/CR-4735 V06: EVALUATION AND COMPILATION OF DOE WASTE PACKAGE TEST DATA. Biannual Report. August 1988 - January 1989. INTERRANTE,C.G. Office of Nuclear Material Safety & Safeguards. ESCALANTE,E.; FRAKER,A.C. National Institute of Standards & Technology (formerly National Bureau of Standa. November 1990. 115pp. 9011270224. 55856:001.

This report summarizes evaluations by the National Institute of Standards and Technology (NIST) of Department Energy (DOE) activities on waste packages designed for containment of radioactive high-level nuclear waste (HLW) for the six-month period August 1988 through January 1989. Included are reviews of related materials research and plans, activities for the DOE Materials Characterization Center, information on the Yucca Mountain Project, and other information regarding supporting research and special assistance. NIST comments are given on the Yucca Mountain Consultation Draft Site Characterization Plan (CDSCP) and on the Waste Compliance Plan for the West

NUREG/CR-4744 V03 N1: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,October 1987 - March 1988. CHOPRA,O.K.; CHUNG,H.M. Argonne National Laboratory. February 1990. 58pp. 9003070245. ANL-89/22. 52802:198.

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 October 1987 to March 1988. A mechanistic understanding of the activation energy of aging is described on the basis of the results of microstructural characterization of various heats of Grades CF-3, CF-8, and CF-8M stainless steel that were used in aging studies at different laboratories. The kinetics of the spinodal decomposition of ferrite (i.e., the primary mechanism of aging embrittlement) appear to be strongly influenced by a synergistic effect of G-phase nucleation and growth. When the activation energies (ranging from 18 to 50 kcal/mole) were plotted as a function of the volume fraction of G phase produced during accelerated aging, a good correlation was obtained regardless of variations in grade, bulk chemical composition, and fabrication process. Spinodal-like decomposition of austenite in heats containing a relatively high level of Ni has also been investigated. Charpy-impact data for thermally aged cast stainless steel were analyzed to determine the kinetics and extent of embrittlement. The ferrite morphology had a strong effect on the extent of embrittlement, whereas the material composition influenced the kinetics of embrittlement. Results obtained from the present study of mechanical properties, and data of other investigators were analyzed to develop the procedure and correlations for predicting the kinetics and extent of embrittlement, under reactor operating conditions, from the material parameters.

NUREG/CR-4744 V03 N2: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,April-September 1988. CHOPRA,O.K.; CHUNG,H.M. Argonne National Laboratory. August 1990. 50pp. 9009250047. ANL-90/5. 55244:021.

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 to September 1988. Characteristics of the primary mechanism of aging embrittlement (i.e., spinodal decomposition of ferrite) and synergistic effects of alloying and impurity elements that influence the kinetics of the primary mechanism are discussed. Several secondary metallurgical processes of embrittlement, strongly dependent on the C, N, Ni, Mo, and Si content of various heats, are identified. Information on kinetics and data on impact properties are analyzed and correlated with microstructural characteristics to provide a unified method of extrapolating accelerated-aging data to reactor operating conditions. Fracture toughness data are presented for several heats of cast stainless steel aged at temperatures between 320 and 450 degrees C for times up to 10,000 h. Mechanical-property data are analyzed to develop the procedure and correlations for predicting the kinetics and extent of embrittlement of reactor components from known material parameters. The method and examples of estimating the impact strength and fracture toughness of cast components during reactor service are described. The lower-bound values of impact strength and fracture toughness for cast stainless steels at LWR operating temperatures are defined.

NUREG/CR-4753 V03: CANADIAN SEISMIC AGREEMENT.Annual Report: 1987-1988. WETMILLER,R.J.; LYONS,J.A.; SHANNON,W.E.; et al. Canadian Commercial Corp. April 1990. 41pp. 9005210158. 53887:048.

During the period of this report, the contract resources were spent on operation and maintenance of the Eastern Canada Telemetred Network (ECTN), development of special purpose local network systems, servicing and maintenance of the strong-motion seismograph network in eastern Canada, operation and of the Ottawa data lab and earthquake monitoring and reporting. Of special note in this period was the final completion of the Sudbury (SLTN) and Charlevoix (CLTN) local networks and the integration of their data processing and analysis requirements in the regular analysis stream for ECTN data. These networks now acquire high quality digital data for detailed analysis of seismic activity and source properties from these two areas, thus effectively doubling the amount of seismic data being received by the Ottawa data lab.

NUREG/CR-4753 V04: CANADIAN SEISMIC AGREEMENT.Annual Report: 1988-1989. WETMILLER,R.J.; LYONS,J.A.; SHANNON,W.E.; et al. Canadian Commercial Corp. April 1990. 61pp. 9005210166. 53891:180.

In this time period eastern Canada experienced its largest earthquake in over 50 years when a magnitude (M) 6.0 event took place at 18:46 EST on Friday, November 25 near 48.12 degrees W just south of Chicoutimi, Quebec. This earthquake, which has been christened the Saguenay earthquake, has provided a wealth of new data pertinent to earthquake engineering studies in eastern North America and is the subject of many continuing studies. The bibliography gives a summary of the scientific reports on earthquake or related studies that have been published or submitted for publication by GD staff in this period.

NUREG/CR-4794: EXPERIMENTAL RESULTS OF CORE-CONCRETE INTERACTIONS USING MOLTEN STEEL WITH ZIRCONIUM. COPUS,E.R.; BLOSE,R.E.; BROCKMANN,J.E.; et al. Sandia National Laboratories. July 1990. 386pp. 9008070412. SAND86-2638. 54849:001.

Four experiments were performed in order to evaluate the additional effects of zirconium metal oxidation on core debris interactions with limestone concrete using molten stainless steel as the core debris simulant. The QT-D, QT-E, SURC-3 and SURC-3A experiments eroded between 10 and 33 cm of limestone concrete during sustained interactions which lasted 35 to 120 minutes. Melt pool temperatures during the tests ranged from 1900 K before zirconium addition to 2100 K during the zirconium-steel-concrete phase of the tests. Large increases in erosion rate, gas production and aerosol release were also measured shortly after Zr metal was added to the melt.

NUREG/CR-4816: 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. June 1990. 104pp. 9006290154. ORNL/TM-10328. 54363:220.

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 compilation of the Power Reactor Embrittlement Data Base (PR-EDB, version 1) contains results from surveillance capsule reports of 78 reactors with 381 data points from 110 different irradiated base materials (plates and forgings) and 161 data points from 79 different welds. Results from heat-affected-zone materials are also listed. Electric Power Research Institute (EPRI), reactor vendors, and utilities are in the process of providing back-up quality assurance checks of the PR-EDB and will be supplementing the data base with additional data and documentation. Periodic updates of data and software will be released to authorized users. Future updates will also include results from irradiations in materials test reactors.

[illegible] FOR THE EXTERNAL EVENT [illegible] CY ANALYSES FOR NUREG-1150. [illegible] AMBRIGHT,J.A. Sandia National Laboratories. [illegible] 990. 145pp. 9012280211. SAND88-3102. [illegible]

[illegible] presents procedures which can be used to assess external hazards at nuclear power plants. These methods were used to perform the external events risk assessments for the Surry and Peach Bottom nuclear power plants as part of the NRC-sponsored NUREG-1150 program. These methods apply to the full range of hazards such as earthquakes, fires, floods, etc. which are collectively known as external events and are based on making full utilization of the power plant systems logic models developed in the internal events analyses. Hallmarks of the methods described include the use of extensive computer-aided screening prior to the detailed analysis of each external event hazard to which the plant might conceivably be exposed. These screening procedures identify those external events which could contribute to the risk at the plant and significantly reduce the number of events for which subsequent detailed analysis is required. Taken together, these techniques provide a relatively straightforward and, in some cases, simplified set of techniques for the analysis of the full range of external events and provides for both scrutability and reproducibility of the final results.

NUREG/CR-4882: QUALIFICATION PROCESS FOR ULTRASONIC TESTING IN NUCLEAR INSERVICE INSPECTION APPLICATIONS. SPANNER,J.C.; DOCTOR,S.R.; TAYLOR,T.T.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1990. 150pp. 9004100344. PNL-6179. 53350:090.

The report documents one of the tasks conducted under a Pacific Northwest Laboratory (PNL) program entitled Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors (NDE Reliability Program). The objective of this task was to develop recommended requirements and processes for qualifying the UT/ISI systems (personnel, equipment, and procedures) for inservice inspection of nuclear power plant components. This report describes an overall qualification process intended to achieve statistically designed performance validations, as well as include the prerequisite training and other qualification recommendations. The document also contains recommendations for the test specimens, environment, and other conditions under which the qualification processes should be conducted. In general, the recommendations described in this document are more stringent than the current industry requirements; however, there are exceptions. The major areas where specific enhancements are recommended include more stringent criteria for Level III qualifications, explicit recommendations for requalification, greater emphasis on periodic (annual) training, and recommendations for coordinating and administering the qualification processes on a national (rather than local employer) basis.

NUREG/CR-4908: ULTRASONIC INSPECTION RELIABILITY FOR INTERGRANULAR STRESS CORROSION CRACKS.A Round Robin Study Of The Effects Of Personnel, Procedures, Equipment, And Crack Characteristics. HEASLER,P.G.; TAYLOR,T.T.; SPANNER,J.C.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1990. 153pp. 9008310179. PNL-6196. 55043:262.

A pipe inspection round robin entitled "Mini-Round Robin" was conducted at Pacific Northwest Laboratory from May 1985 through October 1985. The research was sponsored by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under a program entitled "Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors." The Mini-Round Robin (MRR) measured the IGSC crack detection and sizing capabilities of inservice inspection (ISI) inspectors that had passed the requirements of IEB 83-02 and the EPRI sizing training course. The MRR data base was compared with an earlier Pipe Inspection Round Robin (PIRR) that had measured effective detection prior to 1982. Comparison of the MRR and PIRR data bases indicated no difference in detection capability was measured for long and short cracks. In addition to the pipe inspection round robin, a human factors study was conducted in conjunction with the MRR. The most important result of the human factors study is that the Relative Operating Characteristics (ROC) curves provide a better methodology for describing inspector performance than only POD or single-point crack/no crack data.

NUREG/CR-4918 V04: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Progress Report On Field Experiments At A Humid Region Site,Beltsville,Maryland. SCHULTZ,R.K. California, Univ. of, Berkeley, CA. RIDKY,R.W. Maryland, Univ. of, College Park, MD. O'DONNELL,E. Waste Management Branch. December 1990. 28pp. 9101040285. 56271:057.

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-5001: EFFECTS OF MANUFACTURING VARIABLES ON PERFORMANCE OF HIGH-LEVEL WASTE LOW CARBON STEEL CONTAINERS. FROST,R.H. Colorado School of Mines, Golden, CO. MUTH,T.R.; LIBY,A.L. Manufacturing Sciences Corp. April 1990. 129pp. 9005040070. 53647:242.

Analytical and experimental research was performed to determine the effect of manufacturing variables on the performance of cast steel overpacks. The work examines the influence of casting and welding process variables on the long-term performance of low carbon steel overpacks in the repository environment. Centrifugal casting was indicated to be the most economical and technically favorable manufacturing approach for cast steel overpacks. A bottom would be welded into a hollow cylinder to make the container and final closure welding to secure the lid would be done at the repository. Effects of alloy chemistry, solidification processing, and solid state phase transformations on final microstructure of the cast and welded overpack has been examined in detail in this report. Codes and standards governing the manufacture of overpacks do not presently exist. An extension of the ASME Boiler and pressure vessel code supplemented by government standards could be adopted for the purpose. Standard, well-established methods of non-destructive evaluation are adequate for the purpose of identifying likely manufacturing defects. Experimental work focused on material and process combinations to be used in manufacture of overpacks. Practical process limits were explored and changes in microstructure due to repository thermal conditions were investigated.

NUREG/CR-5111: INTEGRATED RELIABILITY AND RISK ANALYSIS SYSTEM (IRRAS) VERSION 2.0 USER'S GUIDE. RUSSELL,K.D.; SATTISON,M.B. EG&G Idaho, Inc. (subs. of EG&G, Inc.). RASMUSON,D.M. Probabilistic Risk Analysis Branch. June 1990. 464pp. 9007120205. EGG-2535. 54471:203.

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. Also provided in the system is an integrated full-screen editor for use when interfacing with remote mainframe computer systems. 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 2.0 and is the subject of this user's guide. Version 2.0 of IRRAS provides all of 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-5117: STEAM GENERATOR TUBE INTEGRITY PROGRAM/STEAM GENERATOR GROUP PROJECT.Final Project Summary Report. KURTZ,R.J.; CLARK,R.A.; BRADLEY,E.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1990. 210pp. 9008070393. PNL-6446. 54846:060.

The Steam Generator Tube Integrity Program/Steam Generator Group Project was a three-phase program conducted for the U.S. Nuclear Regulatory Commission (NRC) by Pacific Northwest Laboratory. The main goal of the program was to provide the NRC with validated information on the reliability of nondestructive examination techniques to detect and size flaws in steam generator tubing and to determine the remaining integrity of service-degraded tubing. The program was performed in three phases. The first phase involved burst and collapse tests and single-frequency eddy-current (EC) examinations of typical steam generator tubing with precision machined flaws. The goal of Phase I was to develop empirical models of remaining tube integrity as a function of flaw type and size, and to determine the capability of EC inspection methods to detect and size tube degradation. In Phase II, a smaller number of specimens with the same flaw types were investigated, but tube specimens were degraded by chemical means rather than machining methods. This approach was used to better simulate the irregular geometry of service-induced degradation. In the final phase of the program, the retired-from-service Surry 2A Steam Generator was used as a test bed to investigate the reliability of inservice EC inspection equipment, personnel, and procedures, and as a source of service-degraded tubes for further validating the empirical equations of remaining tube integrity.

NUREG/CR-5169: MOBILIZATION AND TRANSPORT OF URANIUM AT URANIUM MILL TAILINGS DISPOSAL SITES.Application Of A Chemical Transport Model. ERIKSON,R.L.; HOSTETLER,C.J.; KEMNER,M.L. Battelle Memorial Institute, Pacific Northwest Laboratory. January 1990. 115pp. 9002120263. PNL-7154. 52596:210.

The geochemical processes of aqueous specification, precipitation, dissolution, and adsorption influence the transport of uranium at uranium mill tailings disposal sites. Traditional transport models involve the use of a single parameter, the retardation factor, to simulate the effects of these geochemical processes. Single parameter models are most applicable to field situations exhibiting no changes in major element chemistry along the flow path. Because of the changes in major element chemistry that occur when acidic leachate contacts a neutralizing soil, a single parameter transport model cannot accurately capture the details of uranium migration at a number of disposal sites. We have used a chemical transport model to qualitatively describe the effects of geochemical mechanisms on uranium transport. The result is a generalized conceptual model that can reproduce the features observed at a number of uranium mill tailings disposal sites.

NUREG/CR-5181: NUCLEAR PLANT AGING RESEARCH:THE 1E POWER SYSTEM. MEYER,L.C.; EDSON,J.L. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1990. 102pp. 9006080179. EGG-2545. 54075:216.

This report presents the results of a study of aging effects on the Class 1E power system in nuclear power plants. The 1E power system is the part of the plant auxiliary power system that supplies power to the safety systems. The purpose of this report is to evaluate the effects of aging caused by operation within a nuclear facility and the effectiveness of maintenance, testing, and monitoring on detecting and mitigating the effects of aging. The U.S. Nuclear Regulatory Commission's Nuclear Plant Aging Research guidelines were followed in performing the detailed study that identifies 1E power system components susceptible to aging, stressors, environmental factors, and failure modes. Testing, maintenance, codes and standards, and regulatory issues are discussed. Degradation mechanisms, failure modes and inspection, surveillance, and monitoring methods are summarized for major class 1E components. This report also presents the results of a review of the 1E power system operating experiences reported in Licensee Event Reports, the Nuclear Power Experience data base, the Nuclear Plant Reliability Data System, and plant maintenance records. Included in this report are the alternating current system (4160 to 120 V), the direct current system, and the vital 120 V ac instrument and control system.

NUREG/CR-5211: UNCERTAINTIES ASSOCIATED WITH PERFORMANCE ASSESSMENT OF HIGH-LEVEL RADIOACTIVE WASTE REPOSITORIES.A Summary Report. DAVIS,P.A.; BONANO,E.J. Sandia National Laboratories. WAHI,K.K.; et al. Gram, Inc. November 1990. 34pp. 9012100331. SAND88-2703. 55991:254.

This report culminates work performed by Sandia National Laboratories (SNL) for the U.S. Nuclear Regulatory Commission (NRC) on uncertainties associated with performance assessment of HLW repositories. Many different types of uncertainty can affect the performance of an HLW repository. In a performance assessment, these uncertainties should be identified and considered, and to the extent practicable, should be quantified and reduced. Conventionally, the different types of uncertainty are classified in three major categories: uncertainty in the future state of the disposal system; uncertainty in models needed to simulate the behavior of the disposal system; and uncertainty in data, parameters, and coefficients needed for the analysis of the system. All three major categories of uncertainty are covered in this report. The reader should not rely on this report for an in-depth treatise of these types of uncertainty. Only a short overview is presented with numerous references to SNL reports where different uncertainty topics are discussed in detail; as such, this report is not a stand-alone report. The report can be used by (1) managers to familiarize themselves with the issues regarding uncertainty in HLW repository performance, and (2) technical staff as a review of SNL's work for NRC in this area.

NUREG/CR-5213 V01: THE COGNITIVE ENVIRONMENT SIMULATION AS A TOOL FOR MODELING HUMAN PERFORMANCE AND RELIABILITY.Executive Summary. WOODS,D.D. Ohio State Univ., Columbus, OH. POPLE,H.E. Pittsburgh, Univ. of, Pittsburgh, PA. ROTH,E.M. Westinghouse Electric Corp. June 1990. 31pp. 9007120152. 54488:163.

The U.S. Nuclear Regulatory Commission is sponsoring a program to develop improved methods to model cognitive behavior of nuclear power plant (NPP) personnel. A tool called Cognitive Environment Simulation (CES) was developed for simulating how people form intentions to act in NPP emergencies. CES provides an analytic tool for exploring plausible human responses in emergency situations. In addition a methodology called Cognitive Reliability Assessment Technique (CREATE) was developed that describes how CES can be used to provide input to human reliability analyses (HRA) in probabilistic risk assessment (PRA) studies. This report describes the results of three activities that were performed to evaluate CES/CREATE: (1) A technical review was conducted by a panel of experts in cognitive modeling, PRA and HRA; (2) CES was exercised on steam generator tube rupture incidents for which data on operator performance exist; (3) A workshop with HRA practitioners was held to analyze a "worked example" of the CREATE methodology. The results of all three evaluations indicate that CES/CREATE is a promising approach for modeling intention formation. Volume 1 provides a summary of the results. Volume 2 provides details on the three evaluations, including the CES computer outputs for the tube rupture events.

NUREG/CR-5213 V02: THE COGNITIVE ENVIRONMENT SIMULATION AS A TOOL FOR MODELING HUMAN PERFORMANCE AND RELIABILITY.Main Report. WOODS,D.D. Ohio State Univ., Columbus, OH. POPLE,H.E. Pittsburgh, Univ. of, Pittsburgh, PA. ROTH,E.M. Westinghouse Electric Corp. June 1990. 69pp. 9007120162. 54486:091.

See NUREG/CR-5213,V01 abstract.

NUREG/CR-5229 V02: TMI-2 EPICOR-II RESIN/LINER INVESTIGATION: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1989.Annual Report. MCCONNELL,J.W.; ROGERS,R.D. EG&G Idaho, Inc. (subs. of EG&G, Inc.). DAVIS,E.C.; et al. Oak Ridge National Laboratory. February 1990. 52pp. 9003120764. EGG-2577. 52682:124.

The EPICOR-II Resin/Liner Investigation: Low-Level Waste Data Base Development Program, funded by the U.S. Nuclear Regulatory Commission (NRC), is studying the degradation effects in EPICOR-II organic ion exchange resins caused by radiation; examining the adequacy of test procedures recommended in the Branch Technical Position on Waste Forms to meet the requirements of 10 CFR 61 using solidified EPICOR-II resin; obtaining performance information on solidified EPICOR-II ion exchange resins in a disposal environment; and determining the condition of EPICOR-II liners. This report summarizes accomplishments of Fiscal Year 1989.

NUREG/CR-5229 V03: EPICOR-II RESIN/LINER INVESTIGATION: LOW-LEVEL WASTE DATABASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1990.Annual Report. MCCONNELL,J.W.; ROGERS,R.D.; JOHNSON,D.A.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). December 1990. 65pp. 9101160006. EGG-2619. 56430:340.

The EPICOR-II Resin/Liner Investigation: Low-Level Waste Database Development Program, funded by the U.S. Nuclear Regulatory Commission, is (a) studying the degradation effects in EPICOR-II organic ion exchange resins caused by radiation, (b) examining the adequacy of test procedures recommended in the Branch Technical Position on Waste Forms to meet the requirements of 10 CFR 61 using solidified EPICOR-II resins, (c) obtaining performance information on solidified EPICOR-II ion exchange resins in a disposal environment, and (d) determining the condition of EPICOR-II liners. Results of the third sampling analysis of ion exchange resins from prefilters PF-8 and PF-20 are compared with baseline data from tests performed on unirradiated resins and with results from the first and second samplings to determine if degradation has occurred because of the high internal radiation dose. Results of compression tests on seven-year old waste forms containing EPICOR-II resins solidified with both Portland Type I-II cement and DOW vinyl ester-styrene are presented and compared to earlier compression test data. Results of the fifth year of data acquisition from the field testing are also presented and discussed.

NUREG/CR-5253: PARTITION: A PROGRAM FOR DEFINING THE SOURCE TERM/ CONSEQUENCE ANALYSIS INTERFACE IN THE NUREG-1150 PROBABILISTIC RISK ASSESSMENTS.User's Guide. IMAN,R.L. Sandia National Laboratories. HELTON,J.C. Arizona State Univ., Tempe, AZ. JOHNSON,J.D. Science Applications International Corp. (formerly Science Applications, Inc.). May 1990. 52pp. 9006080142. SAND88-2940. 54075:318.

This document has been designed for users of the PARTITION computer program developed by the authors at Sandia National Laboratories for defining the interface between the source term analysis and the consequence analysis. The purpose of the PARTITION program is to form groups of source terms with similar properties. One set of MACCS calculations is performed for each of these groups. The following operations are performed in PARTITION: (1) an early fatality weight and a chronic fatality weight are defined for each source term, (2) the source terms are partitioned into groups of source terms with similar radiological potential on the basis of these weights and a single frequency-weighted source term is calculated for each source term group, (3) the source terms in each source term group are divided into subgroups on the basis of evacuation timing and a frequency-weighted source term is calculated for each subgroup, (4) various summary plots are produced to aid in checking the adequacy of the partitioning, and (5) an output file that serves as input to the consequence analysis is generated. The result of the partitioning process is a subdivision of the source terms on the basis of three dimensions: early fatality potential, chronic fatality potential, and evacuation timing.

NUREG/CR-5254: BIAS IN PEAK CLAD TEMPERATURE PREDICTIONS DUE TO UNCERTAINTIES IN MODELING OF ECC BYPASS AND DISSOLVED NON-CONDENSABLE GAS PHENOMENA. ROHATGI,U.S.; NEYMOTIN,L.Y.; JO,J.; et al. Brookhaven National Laboratory. September 1990. 171pp. 9010090047. BNL-NUREG-52168. 55322:253.

The U.S. Nuclear Regulatory Commission (USNRC), its contractors and consultants have developed a methodology for evaluating Code Scaling, Applicability and Uncertainty (CSAU). The CSAU method has been demonstrated by applying it to the TRAC-PF1/MOD1, Version 14.3 code and its analysis of a Large Break Loss of Coolant Accident (LBLOCA) for a Westinghouse four-loop plant. In applying the methodology, the accident course is divided into three different phases, namely: Blowdown, Refill and Reflood. There are two distinct peak clad temperatures (PCT), one in the Blowdown Phase and one in the Reflood Phase. The Reflood Phase PCT is affected by the phenomena related to Emergency Core Cooling System (ECCS) in the downcomer and lower plenum of the reactor vessel. This report describes a general method for estimating the biases in the Reflood Phase PCT from systematic errors (biases) associated with the modelling of the ECCS and dissolved nitrogen, and the application of this method. The bias in the Reflood Phase PCT due to the uncertainty in the existing code models for ECCS related phenomena is -19 degrees K (-34 degrees F). The bias in the PCT due to the lack of modelling of dissolved N(2) in the code is estimated to be 9.9 degrees K (17.8 degrees F). The code prediction for PCT is conservative if the bias is negative, and nonconservative if the bias is positive. The bias estimated here is based on full scale data from the Upper Plenum Test facility and is unaffected by the scale distortions.

NUREG/CR-5256: COMPONENTS OF AN OVERALL PERFORMANCE ASSESSMENT METHODOLOGY. DAVIS,P.A.; PRICE,L.L.; WAHI,K.K.; et al. Sandia National Laboratories. February 1990. 109pp. 9003190287. SAND88-3020. 53012:285.

Both the U.S. Environmental Protection Agency (EPA) and the U.S. Nuclear Regulatory Commission (NRC) have promulgated regulations regarding the performance of geologic repositories for the disposal of high-level nuclear waste. Specifically, the EPA has promulgated three quantitative, postclosure requirements that apply to the entire disposal system, while the

NRC's three quantitative, postclosure requirements apply only to particular subsystems of the repository. To assess compliance with all six of these quantitative requirements, the phenomena that can affect the performance of the repository, the processes by which these phenomena are produced, and the parameters associated with these processes will have to be identified and quantified. In addition, the analyses performed to assess compliance will have to be conducted in accordance with a performance assessment methodology to ensure that all regulatory criteria are addressed. A performance assessment methodology proposed by Sandia National Laboratories is composed of scenario development and screening, consequence analysis, uncertainty analysis, and sensitivity analysis. This methodology can be used to assess compliance with the EPA's and NRC's requirements.

NUREG/CR-5258 V02: GEORGIA/ALABAMA REGIONAL SEISMOGRAPHIC NETWORK.Annual Report,July 1986 - June 1987. LONG,L.T. Georgia Institute of Technology, Atlanta, GA. April 1990. 71pp. 9005210180. 53891:242.

Data from continued operation of the seismographic network were used for topical studies. A small swarm of earthquakes occurred in the Lake Sinclair, Georgia vicinity during the first quarter of 1987. The events occurred in three main shocks of magnitude 2 and their aftershocks. The third event had an unusually rapid decay in aftershock activity, suggesting complete release of stress. Attenuation and scattering were studied on the basis of theoretical evaluations of the effects of spherical inclusions in elastic or viscoelastic media and on the basis of P-wave coda. Resonance of the scatterer and viscosity of a solid material play an important role in determining scattering coefficients. Wavelets comprising the P-wave coda show systematic increases and decreases in spectral peaks that may be explained by shifts in the corner frequency caused by absorptive attenuation and scattering. In locating hypocenters, the method used at Georgia Tech computes origin time independent of location. Location and depth are then computed with a fixed origin time. When S and P times are available, including off-diagonal elements in the covariance matrix leads to locations that are more precise than those obtained with standard methods.

NUREG/CR-5258 V03: GEORGIA/ALABAMA REGIONAL SEISMOGRAPHIC NETWORK.Annual Report,July 1987 - June 1988. LONG,L.T. Georgia Institute of Technology, Atlanta, GA. September 1990. 103pp. 9010230086. 55472:078.

A review of the historical seismicity of southeastern Tennessee suggests that the period of 1970 through 1988 was anomalously more active than before 1970. If this is true, then it represents a unique stress release event that would satisfy the model for major intraplate earthquakes proposed by Long (1988). First motion data and SV/P amplitude ratios were used to determine 36 single events and two composite focal mechanism solutions for earthquakes in southeastern Tennessee. The dominant focal mechanism is strike slip with reverse or normal fault components. Linearizing the relation between measured apparent coda Q and assumed constant coda Q for discrete zones of crust suggests an anomalously low (Q=50) region northeast of station CBT. Fracture density was mapped in an area of induced seismicity near Clarks Hill Reservoir. It appears that areas of higher joint spacing correspond to zones of induced seismicity. In this and other reservoirs, earthquakes tend to occur in granite gneiss where jointing is sparse and the rock is strong. Seismic activity is not often observed in foliated schists and altered mafic rocks in which stress may be released through creep.

NUREG/CR-5258 V04: GEORGIA/ALABAMA REGIONAL SEISMOGRAPHIC NETWORK.Final Report, August 1985 - October 1990. LONG,L.T. Georgia Institute of Technology, Atlanta, GA. November 1990. 132pp. 9012280158. 56195:316.

Data from the Georgia/Alabama network have contributed to a better understanding of the seismicity in the Southeast. Based on these data, a new theory explaining intraplate earthquakes was developed. The theory predicts that a decrease in strength of the lower crust (e.g., through a change in the fluid regime) leads to weakening and deformation of the stress channel in the mid-crustal zone. The weakening and stress concentration may lead to major earthquakes. Earthquake focal mechanisms in southeastern Tennessee are consistent with such a model. Conclusions reached from this and other studies suggest that major earthquakes have happened or could happen in southeastern Tennessee. Earthquakes in the Piedmont are of a different type, being mostly shallow, consistent with failure along joints and predominantly associated with reservoir impoundment. This mechanism may leave an upper magnitude limit of 5.7. In Alabama, except near southeastern Tennessee, the seismicity is largely induced by coal mine collapses.

NUREG/CR-5262: PRAMIS: PROBABILITY RISK ASSESSMENT MODEL INTEGRATION SYSTEM.User's Guide. IMAN,R.L.; JOHNSON,J.D.; HELTON,J.C. Sandia National Laboratories. May 1990. 80pp. 9006080164. SAND88-3093. 54077:012.

This document has been designed for users of the Probabilistic Risk Assessment Model Integration System (PRAMIS) computer program developed by the authors at Sandia National Laboratories for easy assembly of the individual parts of the NUREG-1150 plant analyses into overall risk results. PRAMIS assembles the following files associated with the NUREG-1150 analyses in matrix format to obtain risk: the Latin hypercube sample, the results of the systems analysis, the results of the accident progression analysis, the results of the source term/partitioning analysis, and the results of the consequence analysis. In addition, various intermediate and conditional quantities are calculated when requested by user-specified input; the fractional contribution to risk of individual plant damage states, accident progression bins and source term groups are determined, and a file containing the original Latin hypercube sample and user-specified dependent variables is generated for use as input to the SAS statistical package. This report provides a tutorial that details how to use the PRAMIS program. The PRAMIS program is written in ANSI standard FORTRAN 77 to make the code as machine-independent (that is, portable) as possible.

NUREG/CR-5273 V04: SCDAP/RELAP5/MOD2 CODE MANUAL,VOLUME 4:MATPRO - A LIBRARY OF MATERIALS PROPERTIES FOR LIGHT-WATER-REACTOR ACCIDENT ANALYSIS. BUCCAFURNI,A.; CARLSON,E.R.; CHAMBERS,R.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). February 1990. 1,099pp. 9003190269. EGG-2555. 53002:040.

This report describes the materials properties correlations and computer subcodes (MATPRO) developed for use with various light water reactor (LWR) accident analysis computer programs. Formulation of the materials properties are generally semiempirical in nature. The materials properties subcodes contained in this document are for uranium, uranium dioxide, mixed uranium-plutonium dioxide fuel, zircaloy cladding, zirconium dioxide, stainless steel, stainless steel oxide, silver-indium-cadmium alloy, boron carbide, Inconel 718, zirconium-uranium-oxygen melts, and fill gas mixtures.

NUREG/CR-5280 V01: AGE-RELATED DEGRADATION OF WESTINGHOUSE 480-VOLT CIRCUIT BREAKERS.Aging Assessment And Recommendations For Improving Breaker Reliability. SUBUDHI,M.; SHIER,W.; MACDOUGALL,E. Brookhaven National Laboratory. July 1990. 90pp. 9008070370. BNL-NUREG-52178. 54845:160.

An aging assessment of Westinghouse DS-series low-voltage air circuit breakers was performed as part of the Nuclear Plant Aging Research (NPAR) program. The objectives of this study are to characterize age-related degradation within the breaker assembly and to identify maintenance practices to mitigate their effects. Since this study has been promulgated by the failures of the reactor trip breakers at the McGuire Nuclear Station in July 1987, results relating to the welds in the breaker pole lever welds are also discussed. The design and operation of DS-206 and DS-416 breakers were reviewed. Failure data from various national data bases were analyzed to identify the predominant failure modes, causes, and mechanisms. Additional operating experiences from one nuclear station and two industrial breaker-service companies were obtained to develop aging trends of various subcomponents. The responses of the utilities to the NRC Bulletin 88-01, which discusses the center pole lever welds, were analyzed to assess the final resolution of failures of welds in the reactor trips.

NUREG/CR-5280 V02: AGE-RELATED DEGRADATION OF WESTINGHOUSE 480-VOLT CIRCUIT BREAKERS.Mechanical Cycling Of A DS-416 Breaker.Test Results. SUBUDHI,M.; MACDOUGALL,E.; KOCHIS,S.; et al. Brookhaven National Laboratory. November 1990. 124pp. 9012280152. BNL-NUREG-52178. 56195:192.

A DS-416 medium voltage air circuit breaker manufactured by Westinghouse was mechanically cycled to identify age-related degradations. This accelerated aging test was conducted for over 36,000 cycles during 9 months. Three separate pole shafts, one with a 60 degree weld, one with a 120 degree, and one with a 180 degree were used to characterize the cracking in the pole lever welds. In addition, three different operating mechanisms and several other parts were replaced as they became inoperable. The testing yielded many useful results. The burning of the closing coils was found to be the effect of binding in the linkages that are connected to this device. Among the seven welds on the pole shaft, #1 and #3 were the critical ones which cracked first to cause misalignment of the pole levers, which in turn, had led to many problems with the operating mechanism including the burning of coils, excessive wear in certain parts, and overstressed linkages. Based on these findings, a maintenance program is suggested to alleviate the age-related degradations that occur due to mechanical cycling of a breaker.

NUREG/CR-5286: CLOSEOUT OF IE BULLETIN 79-17:PIPE CRACKS IN STAGNANT BORATED WATER SYSTEMS AT PWR PLANTS. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. February 1990. 35pp. 9004090172. PARAMETER IE177. 53314:237.

Documentation is provided in this report for the closeout of IE Bulletin 79-17 and its revision on the safety-related subject of pipe cracks in stagnant borated water systems at operating plants with pressurized water reactors (PWRs). Closeout is based on the implementation and verification of actions required by the bulletin. Evaluation of utility responses and NRC/Region inspection reports indicates that the bulletin is closed for all of the 41 operating PWRs to which it was issued for action. It is concluded that the concerns of the bulletin have been resolved. Background information is supplied in the Introduction and Appendix A.

NUREG/CR-5289: CLOSEOUT OF IE BULLETIN 79-23:POTENTIAL FAILURE OF EMERGENCY DIESEL GENERATOR FIELD EXCITER TRANSFORMER. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. March 1990. 27pp. 9004090279. PARAMETER IE180. 53330:298.

Documentation is provided in this report for the closeout of IE Bulletin 79-23 regarding the potential failure of emergency diesel generator field exciter transformers. Closeout is based upon the implementation and verification of three actions required by the bulletin. Evaluation of utility responses and NRC/Region inspection reports indicates that the bulletin is closed for all of the 119 nuclear power facilities with an operating license or a construction permit at the time the bulletin was issued, September 12, 1979. Deviations from bulletin testing requirements, along with licensee justifications, are listed. It is concluded that the problem of concern was not generic, since only two plants, Nine Mile Point 1 and Turkey Point, required modifications to correct connections which could cause high circulating currents. The problem at Turkey Point is described in the bulletin (see page A-1). Background information is supplied in the Introduction and Appendix A.

NUREG/CR-5295: CLOSEOUT OF IE COMPLIANCE BULLETIN 86-03: POTENTIAL FAILURE OF MULTIPLE ECCS PUMPS DUE TO SINGLE FAILURE OF AIR-OPERATED VALVE IN MINIMUM FLOW RECIRCULATION LINE. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. October 1990. 31pp. 9011050058. PARAMETER IE186. 55614:152.

Documentation is provided in this report for the closeout of IE Compliance Bulletin 86-03 regarding the potential failure of multiple Emergency Core Cooling System (ECCS) pumps due to a single failure of an air-operated valve in a minimum flow recirculation line. Closeout is based on the implementation and verification of four actions required by the bulletin. Evaluation of utility responses and NRC/Region inspection reports in accordance with specific criteria indicates that the bulletin is closed for 116 (98%) of the 118 nuclear power facilities in operation or under construction to which it was issued for action. Facilities which were shut down indefinitely or permanently, or which had construction halted indefinitely were not included in this review. A follow-up item is proposed for the two (2) facilities with open bulletin status. It is concluded that the bulletin concern has been resolved, pending closeout by the NRC of Zion 1,2. Background information is provided in the Introduction and Appendix A.

NUREG/CR-5297: CLOSEOUT OF IE BULLETIN 83-05:ASME NUCLEAR CODE PUMPS AND SPARE PARTS MANUFACTURED BY THE HAYWARD TYLER PUMP COMPANY. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. August 1990. 33pp. 9009180017. PARAMETER IE188. 55214:290.

Documentation is provided in this report to close IE Bulletin 83-05 regarding ASME nuclear code pumps and spare parts manufactured by the Hayward Tyler Pump Company (HTPC). The bulletin was issued (1) to alert holders of operating licenses and construction permits of nuclear power plants that HTPC failed to implement effectively their quality assurance (QA) program from 1977 to 1981 and (2) to require affected utilities to take action to resolve the potential for failure of the subject pumps and their spare parts. Evaluation of utility responses and NRC/Region inspection reports shows that reliability of the affected pumps was ensured by means of procedures and performance testing of the pumps as required by the bulletin. Based on the evaluation, in accordance with specific criteria, the bulletin is closed for 116 (98%) of the 118 facilities to which it was issued for action and which were not shut down indefinitely or permanently at the time of issuance of this report. A follow-up item is proposed for the two facilities with open bulletin status. Based on favorable results, a conclusion is presented to indicate that the bulletin concerns have been resolved. Background information is supplied in the Introduction and Appendix A.

NUREG/CR-5298: CLOSEOUT OF IE BULLETIN 85-01: STEAM BINDING OF AUXILIARY FEEDWATER PUMPS. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. January 1990. 37pp. 9003070241. PARAMETER IE189. 52803:268.

Documentation is provided in this report for the closeout of IE Bulletin 85-01 regarding steam binding of auxiliary feedwater pumps for certain pressurized water reactors (PWRs) in nuclear power facilities. Individual facility closeouts are based on the implementation and verification of the three actions required by the bulletin. Evaluation of utility responses and NRC/Region inspection reports indicates that the bulletin is closed for 44 (92%) of the 48 facilities to which it was issued for action. Followup items are proposed for the use of NRC regional inspectors in ensuring satisfactory completion of required actions for the four facilities with open bulletin status. Conclusions are summarized in accordance with Generic Letter 88-03, which announced the NRC Staff's resolution of Generic Issue 93 on the subject of this bulletin. Background information is supplied in the Introduction and Appendix A.

NUREG/CR-5302: CLOSEOUT OF IE BULLETIN 80-10: CONTAMINATION OF NONRADIOACTIVE SYSTEM AND RESULTING POTENTIAL FOR UNMONITORED, UNCONTROLLED RELEASE OF RADIOACTIVITY TO ENVIRONMENT. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. February 1990. 30pp. 9003120758. PARAMETER IE193. 52882:237.

Documentation is provided in this report for the closeout of IE Bulletin 80-10 regarding contamination of nonradioactive systems resulting in the potential for unmonitored, uncontrolled release of radioactivity to the environment. Closeout is based on the documentation and verification of four actions required by the bulletin for holders of an operating license for a nuclear power facility at the time the bulletin was issued (05-06-80). The bulletin was issued for information to holders of a construction permit for a nuclear power facility. Evaluation of utility responses and NRC/Region inspection reports in accordance with the close-out criterion indicates that the bulletin is closed for 65 (98%) of the 66 nuclear power facilities to which it was issued for action. A follow-up item is proposed for the facility with open status, for the use of NRC regional inspectors in ensuring successful completion of required actions. When the bulletin is closed as anticipated for the facility which requires follow-up (see page C-1), the concerns of the bulletin will have been resolved completely. Background information is supplied in the Introduction and Appendix A.

NUREG/CR-5306: CLOSEOUT OF IE BULLETIN 79-18: AUDIBILITY PROBLEMS ENCOUNTERED ON EVACUATION OF PERSONNEL FROM HIGH-NOISE AREAS. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. November 1990. 19pp. 9012280214. PARAMETER IE197. 56258:132.

Documentation is provided in this report for the closeout of IE Bulletin 79-18 regarding audibility problems encountered on evacuation of personnel from high-noise areas. Closeout is based on the implementation and verification of four (4) required actions. Evaluation of utility responses and NRC/Region inspection reports indicates that the bulletin is closed for all of the 64 operating nuclear power facilities to which it was issued for action. 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.

NUREG/CR-5307: CLOSEOUT OF IE BULLETIN 80-02: INADEQUATE QUALITY ASSURANCE FOR NUCLEAR SUPPLIED EQUIPMENT. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. December 1989. 15pp. 9002120237. PARAMETER IE198. 52611:204.

Documentation is provided in this report for the closeout of IE Bulletin 80-02. The subject is inadequate quality assurance for nuclear equipment supplied by the Marvin Engineering Company. The equipment of concern is supplied either directly or through other suppliers for use in General Electric boiling water reactors (BWRs). Closeout is based on the implementation and verification of three (3) required actions. Evaluation of utility responses and NRC/Region inspection reports indicates that the bulletin is closed for all of the 38 BWR facilities to which it was issued for action. It is concluded that the safety concerns reflected in the IE Bulletin 80-02 were adequately resolved by the actions taken by licensees and verified by NRC inspectors. Background information is presented in the Introduction and Appendix A.

NUREG/CR-5308: CLOSEOUT OF IE BULLETIN 80-15: POSSIBLE LOSS OF EMERGENCY NOTIFICATION SYSTEM (ENS) WITH LOSS OF OFFSITE POWER. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. December 1990. 38pp. 9101040257. PARAMETER IE199. 56270:237.

Documentation is provided in this report for the closeout of IE Bulletin 80-15 for nuclear power reactors. This bulletin pertained to a possible loss of the Emergency Notification System (ENS) upon loss of offsite power. Closeout is based on the implementation and verification of six (6) required actions by licensees of nuclear power reactors in operation or near to receiving an operating license when the bulletin was issued on June 18, 1980. Evaluation of utility responses and NRC/Region inspection reports indicates that the bulletin is closed for all of the 69 nuclear power reactors to which it was issued for action and which were not shut down indefinitely or permanently at the time of issuance of this report. Background information is supplied in the Introduction and Appendix A. Nuclear fuel facilities as well as nuclear power facilities were identified in the enclosures to the bulletin. However, per an NRC memorandum, the closeout of the bulletin for nuclear fuel facilities is not within the scope of this report.

NUREG/CR-5314 V03: LIFE ASSESSMENT PROCEDURES FOR MAJOR LWR COMPONENTS.Cast Stainless Steel Components. JASKE,C.E.; SHAH,V.N. EG&G Idaho, Inc. (subs. of EG&G, Inc.). October 1990. 64pp. 9011060093. EGG-2562. 55616:297.

This report presents a procedure for estimating the current condition and residual life of safety-related cast stainless steel components in light water reactors (LWRs). The procedure accounts for loss of fractures toughness caused by thermal embrittlement and includes the following: a review of design and fabrication records, inservice inspection records, and operating history; a fracture mechanics evaluation to determine the required toughness estimates; and criteria regarding continued service, repair, or replacement of the component being evaluated. The report discusses the available Charpy V-notch impact energy, fracture toughness, tensile strength, fatigue resistance, and fatigue-crack growth data, and presents two methods for assessing the degree of thermal embrittlement: metallurgical evaluation and analytical modeling of inservice degradation.

NUREG/CR-5316: MELT PROGRESSION,OXIDATION, AND NATURAL CONVECTION IN A SEVERELY DAMAGED REACTOR CORE. DOSANJH,S.S. Sandia National Laboratories. February 1990. 93pp. 9003070229. SAND88-3476. 53361:156.

In the revised Severe Accident Research Program plan, the U.S. Nuclear Regulatory Commission places a great emphasis on code documentation. This present report, which describes work that was conducted over a period of several years in the MELPROG code development project, is intended to support this goal. Of interest here is the late phase of core melt progression. A model that treats some of the important physical processes that can occur during this phase of the accident is described herein. A number of straightforward examples are given to illustrate the utility of the model and to identify the dominant physical processes.

NUREG/CR-5366: HTAS2: A THREE-DIMENSIONAL TRANSIENT SHIPPING CASK ANALYSIS TOOL. WENDEL,M.W.; GILES,G.E. Oak Ridge National Laboratory. May 1990. 81pp. 9006290147. ORNL/CSD/TM-267. 54324:312.

This report describes the HTAS2 computer program which can be used to assess the thermal behavior of shipping casks containing PWR-type fuel assemblies. HTAS2 has two parts: a global cask analysis and a single assembly analysis. The global cask analysis is a three-dimensional lumped parameter model for the entire shipping cask and its contents. The user has the option of simulating a prefire steady state, a fire transient, a postfire transient, or a final steady state in acceptable order. Details within the fuel assemblies are not resolved in this portion of the analysis. The single assembly analysis is two-dimensional and contains a detailed radiation and conduction model which can be used to represent PWR square pitched fuel assemblies. A simple convection model is available. The boundary conditions on the basket walls surrounding the assembly can be directly specified or may come from a previous global cask analysis. Good comparison was obtained with other computational results for both parts of the analysis. Although good agreement was found between the single assembly results and experimental data, the global cask analysis has not yet been validated against experiments; this task is recommended as a part of future work.

NUREG/CR-5368: REACTIVITY ACCIDENTS.A Reassessment Of The Design-Basis Events. DIAMOND,D.J.; HSU,C.J.; FITZPATRICK,R. Brookhaven National Laboratory. January 1990. 135pp. 9002120269. BNL-NUREG-52198. 52596:325.

This report documents a study of light water reactor event sequences which have been investigated for their potential to result in reactivity accidents with severe consequences. The study is an outgrowth of the concern which arose after the accident at Chernobyl and was recommended by the report of the U.S. Nuclear Regulatory Comission (NRC) on the implications of that accident (NUREG-1251). The work was done for the NRC to reconfirm or bring into question previous judgements on reactivity events which must be analyzed for licensing. Event sequences were defined and then a probabilistic assessment was completed to estimate the frequency of the reactivity event and/or a deterministic calculation was completed to estimate the consequences to the fuel. Using the results of this analysis done by others, and a set of screening criteria developed within this study, judgements were made for each sequence as to its importance, and recommendations were made as to whether the NRC ought to be considering the important sequences as part of the design basis or for further, more detailed, investigation.

NUREG/CR-5369: THE SEISMIC CATEGORY I STRUCTURES PROGRAM RESULTS FOR FY 1987. FARRAR,C.R.; BENNETT,J.G.; DUNWOODY,W.E.; et al. Los Alamos National Laboratory. October 1990. 83pp. 9011060099. LA-11607-MS. 55616:212.

The accomplishments of the Seismic Category I Structures Program for FY 1987 are summarized. These accomplishments include the quasi-static load cycle testing of large shear wall elements, an extensive analysis of previous data to determine if equivalent linear analytical models can predict the response of damaged shear wall structures, and code committee activities. In addition, previous testing and results that led to the FY 1987 program plan are discussed and all previous data relating to shear wall stiffness are summarized. Because separate reports have already summarized the experimental and analytical work in FY 1987, this report will briefly highlight this work and the appropriate reports will be referenced for a more detailed discussion.

NUREG/CR-5373: A DEMONSTRATION EXPERIMENT OF STEAM-DRIVEN, HIGH-PRESSURE MELT EJECTION.The HIPS-10S Test. ALLEN,M.D.; NICHOLS,R.T.; PILCH,M. Sandia National Laboratories. July 1990. 92pp. 9008310209. SAND89-1135. 55043:170.

A steam blowdown test was performed at the Surtsey Direct Heating Test Facility to test the steam supply system and burst diaphragm arrangement that will be used in subsequent Surtsey Direct Containment Heating (DCH) experiments. Following successful completion of the steam blowdown test, the HIPS-10S (High-Pressure Melt Streaming) experiment was conducted to demonstrate that the technology to perform steam-driven, high-pressure melt ejection (HPME) experiments had been successfully developed. In addition, the HIPS-10S experiment was used to assess techniques and instrumentation designed to create the proper timing of events in HPME experiments.

NUREG/CR-5374: SUMMARY OF INADEQUATE CORE COOLING INSTRUMENTATION FOR UNITED STATES NUCLEAR POWER PLANTS. ANDERSON,J.L.; HAGEN,E.W.; MORELOCK,T.C. Oak Ridge National Laboratory. July 1990. 197pp. 9008070334. ORNL/TM-11200. 54850:027.

This report summarizes a review of Inadequate Core Cooling Instrumentation installed in U.S. nuclear power plants in response to the requirements of NUREG-0737, Clarification of TMI Action Plan Requirements, and related orders. The review includes descriptions of generic systems developed by Westinghouse and Combustion Engineering, as well as plant specific reviews of each pressurized water reactor installation. Performance characteristics are discussed, including feedback from plant personnel concerning installation, operational experience, and operator acceptance. An evaluation of boiling water reactor systems is included.

NUREG/CR-5376: QUALITY ASSURANCE AND VERIFICATION OF THE MACCS CODE, Version 1.5. DOBBE,C.A.; MARWIL,E.S.; CARLSON,E.R.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). February 1990. 61pp. 9003120744. EGG-2566. 52882:176.

An independent quality assurance (QA) and verification of Version 1.5 of the MELCOR Accident Consequence Code System (MACCS) was performed. The QA and verification involved examination of the code and associated documentation for consistent and correct implementation of the models in an error-free FORTRAN computer code. The QA and verification was not intended to determine either the adequacy or appropriateness of the models that are used in MACCS 1.5. The reviews uncovered errors which were fixed by the SNL MACCS code development staff prior to the release of MACCS 1.5. Some difficulties related to documentation improvement and code restructuring are also presented. The QA and verification process concluded that Version 1.5 of the MACCS code, within the scope and limitations of the models implemented in the code, is essentially error free and ready for widespread use.

NUREG/CR-5381: ECONOMIC RISK OF CONTAMINATION CLEANUP COSTS RESULTING FROM LARGE NONREACTOR NUCLEAR MATERIAL LICENSEE OPERATIONS. PHILBIN,J.S. Sandia National Laboratories. SALOIO,J.H. ERC Environmental & Energy Services, Inc. ROLLSTIN,J. Gram, Inc. March 1990. 185pp. 9004100430. SAND89-1302. 53344:114.

Several potential incident scenarios involving the accidental release of radioactive material at five reference, nonreactor nuclear material licensees are analyzed in this report. The economic risk ($/licensee/yr) of decontamination is evaluated for each reference licensee. Although most releases and cleanup costs are minor, some less frequent incidents may result in very high cleanup costs that dominate the economic risk of decontamination of a particular licensee. The economic risk for the 5 plants ranged from a low of $14,000 per licensee per year to a high of $104,000 per licensee per year. This report is the second of two reports by Sandia National Laboratories on the economic risk of nonreactor nuclear material licensee operations. This report provides technical basis for a proposed financial responsibility rulemaking for nonreactor nuclear material licensees.

NUREG/CR-5385: INITIAL ASSESSMENT OF THE MECHANISMS AND SIGNIFICANCE OF LOW-TEMPERATURE EMBRITTLEMENT OF CAST STAINLESS STEELS IN LWR SYSTEMS. CHOPRA,O.K.; SATHER,A. Argonne National Laboratory. August 1990. 252pp. 9009040085. ANL-89/17. 55048:043.

This report summarizes work performed by Argonne National Laboratory on long-term embrittlement of cast duplex stainless steels in LWR systems. Metallurgical characterization and mechanical property data from Charpy-impact, tensile, and J-R curve tests are presented for several experimental and commercial heats, as well as for reactor-aged CF-3, CF-8, and CF-8M cast stainless steels. The effects of material variables on the embrittlement of cast stainless steels are evaluated. Chemical composition and ferrite morphology strongly affect the extent and kinetics of embrittlement. In general, the low-carbon CF-3 stainless steels are the most resistant and the molybdenum-containing high-carbon CF-8M stainless steels are most susceptible to embrittlement. The microstructural and mechanical-property data are analyzed to establish the mechanisms of embrittlement. The procedure and correlations for predicting the impact strength and fracture toughness of cast components during reactor service are described. The lower bound values of impact strength and fracture toughness for low-temperature-aged cast stainless steel are defined.

NUREG/CR-5386: BASIS FOR SNUBBER AGING RESEARCH: NUCLEAR PLANT AGING RESEARCH PROGRAM. BROWN,D.P. Lake Engineering, Inc. PALMER,G.R. Wyle Laboratories. WERRY,E.V.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. January 1990. 119pp. 9002120341. PNL-6911. 52601:008. This report describes a research plan to address the safety concerns of aging in snubbers used on piping and equipment in commercial nuclear power plants. The work is to be performed under Phase II of the Snubber Aging Study of the Nuclear Plant Aging Research Program of the U.S. Nuclear Regulatory Commission with the Pacific Northwest Laboratory (PNL) as the prime contractor. Research conducted by PNL under Phase I provided an initial assessment of snubber operating experience and was primarily based on a review of licensee event reports. The work proposed is an extension of Phase I and includes research at nuclear power plants and in test laboratories. Included is technical background on the design and use of snubbers in commercial nuclear power applications; the primary failure modes of both hydraulic and mechanical snubbers are discussed. The anticipated safety, technical, and regulatory benefits of the work, along with concerns of the NRC and the utilities, are also described.

NUREG/CR-5393: A REVIEW OF TECHNIQUES FOR PROPAGATING DATA AND PARAMETER UNCERTAINTIES IN HIGH-LEVEL RADIOACTIVE WASTE REPOSITORY PERFORMANCE ASSESSMENT MODELS. ZIMMERMAN,D.A.; WAHL,K.K.; GUTJAHR,A.L.; et al. Sandia National Laboratories. March 1990. 150pp. 9007120186. SAND89-1432. 54470:014.

Techniques for propagating data and parameter uncertainties in high-level waste (HLW) repository performance assessment models are discussed. Uncertainty analysis techniques ascribe quantitative measures of reliability to model predictions. Both 10 CFR Part 60 and 40 CFR Part 191 require consideration of uncertainties, including uncertainties in data and parameters, in the performance assessment of an HLW repository system. Four categories of uncertainty analysis methods are discussed: Monte Carlo simulation, replacement models (response surface techniques), differential techniques (direct, adjoint, and Green's function technique), and geostatistical techniques (stochastic modeling using Monte Carlo simulation and spectral analysis). Advantages, disadvantages and applications of each technique are presented. Propagation of uncertainties through multiple, linked models is also discussed. Application of these techniques to sensitivity analysis is also presented. Sensitivity analyses can be useful to uncertainty studies because the number of parameters included in the uncertainty analysis can be reduced by eliminating those parameters for which the uncertainty has a minimal effect on the performance variable(s).

NUREG/CR-5395 V02: MULTILOOP INTEGRAL SYSTEM TEST (MIST): FINAL REPORT.Test Group 30,Mapping Tests. GEISSLER,G.O. Babcock & Wilcox Co. December 1989. 1,058pp. 9003070265. EPRI/NP-6480. 52803:305.

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 and two other supporting facilities were specifically designed and constructed for this program, and an existing facility--the Once Through Integral System (OTIS)--was also used. Data from MIST and the other facilities will be 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. The program is summarized in Volume 1; Volumes 2 through 8 describes groups of tests by test type; Volume 9 presents inter-group comparisons; Volume 10 provides comparisons between the calculations of RELAP5/MOD2 and MIST observations, and Volume 11 presents the later Phase 4 tests. This Volume 2 pertains to MIST mapping tests performed to traverse the early post-SBLOCA events slowly. The tests investigated the effect of test-to-test variations in boundary system controls, and only the primary fluid mass varied during a specific test in this test group.

NUREG/CR-5395 V10: MULTILOOP INTEGRAL SYSTEM TEST (MIST): FINAL REPORT. RELAP5/MOD2 MIST Analysis Comparisons. KLINGENFUS,J.A.; PARECE,M.V. Babcock & Wilcox Co. December 1989. 300pp. 9002120143. EPRI/NP-6480. 52580:313.

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 and two other supporting facilities were specifically designed and constructed for this program, and an existing facility--the Once Through Integral System (OTIS)--was also used. Data from MIST and the other facilities will be 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. The program is summarized in Volume 1; Volumes 2 through 8 describes groups of tests by test type; Volume 9 presents inter-group comparisons; Volume 10 provides comparisons between the calculations of RELAP5/MOD2 and MIST observations, and Volume 11 presents the later Phase 4 tests. The comparisons of RELAP5/MOD2 against the MIST data and conclusions reached are the subject of this volume.

NUREG/CR-5395 V11: MULTILOOP INTEGRAL SYSTEM TEST (MIST):FINAL REPORT.MIST Phase IV Tests. GEISSLER,G.O. Babcock & Wilcox Co. August 1990. 528pp. 9009210290. EPRI/NP-6480. 55237:169.

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 and two other supporting facilities were specifically designed and constructed for this program, and an existing facility--the Once Through Integral System (OTIS)--was also used. Data from MIST and the other facilities will be 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. The program is summarized in Volume 1; Volumes 2 through 8 describe groups of tests by test type; Volume 9 presents inter-group comparisons; Volume 10 provides comparisons between the calculations of RELAP5/MOD2 and MIST observations, and Volume 11 presents the later Phase 4 tests. This Volume 11 pertains to MIST Phase IV tests performed to investigate risk dominant transients and non-LOCA events.

NUREG/CR-5395 V11 AD: MULTILOOP INTEGRAL SYSTEM TEST (MIST):FINAL REPORT.MIST Phase IV Tests. GEISSLER,G.O. Babcock & Wilcox Co. August 1990. 273pp. 9009210229. EPRI/NP-6480. 55233:202.

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 and two other supporting facilities were specifically designed and constructed for this program, and an existing facility--the Once Through Integral System (OTIS)--was also used. Data from MIST and the other facilities will be 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. The program is summarized in Volume 1; Volumes 2 through 8 describe groups of tests by test type; Volume 9 presents inter-group comparisons; Volume 10 provides comparisons between the calculations of RELAP5/MOD2 and MIST observations, and Volume 11 presents the later Phase 4 tests. This Volume 11 addendum pertains to MIST natural circulation tests.

NUREG/CR-5397: VALUE-IMPACT ANALYSIS OF REGULATORY OPTIONS FOR RESOLUTION OF GENERIC ISSUE C-8:MSIV LEAKAGE AND LCS FAILURE. JAMISON,J.D.; VO,T.V.; TABATABAI,A.S. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1990. 53pp. 9007120218. PNL-6931. 54467:273.

This report describes the analysis conducted to establish the basis for answering two remaining regulatory questions facing the NRC staff regarding the resolution of Generic Issue C-8, specifically: 1) What action should the NRC take concerning plants that currently have a leakage control system (LCS)? 2) What action should the NRC take concerning plants that do not have an LCS? Using individual MSIV leak test data, the performance of a system of eight such valves in a standard BWR configuration was modeled. The performance model was used along with estimates of core damage accident frequency and calculated dose consequences to determine the public risk associated with each of the alternatives. The occupational exposure implications of each alternative were calculated using estimates of labor hours in radiation zones that would be incurred or avoided. The costs to industry of implementing each alternative were estimated using standard cost formulae and NRC staff estimates. The costs to the NRC were estimated based on the effort incurred or avoided for reviews or other staff actions engendered by the selection of a particular alternative. The costs and risks thus calculated suggest that no regulatory action can be justified on the basis of risk reduction or cost savings.

NUREG/CR-5398: TECHNICAL BASIS FOR REVIEW OF HIGH-LEVEL WASTE REPOSITORY MODELING. PRICE,L.L.; WAHI,K.K.; GALLEGOS,D.P.; et al. Sandia National Laboratories. March 1990. 43pp. 9004100413. SAND89-1557. 53344:042.

Both the U.S. Environmental Protection Agency (EPA) and the U.S. Nuclear Regulatory Commission (NRC) have promulgated regulations regarding the performance of geologic repositories for the disposal of high-level nuclear waste. One of the responsibilities of the U.S. Department of Energy (DOE) is to demonstrate compliance with the appropriate regulations. The DOE will most likely use extensive numerical modeling to show compliance with the various quantitative requirements. Theses analyses will then be evaluated by the NRC. There are different levels of evaluation: peer review, conservative estimates, use of existing models/codes, and development of models/codes by the NRC. The intensity of the review will vary from analysis to analysis, depending on the importance of the analysis, the acceptability of the conceptual model behind the analysis and the solution technique used, and the potential for increasing confidence in the system description, should the NRC decide to develop its own models/codes. An appropriate level of review can be determined by applying these four criteria in a specific manner.

NUREG/CR-5399: SURVEY OF STATE AND TRIBAL EMERGENCY RESPONSE CAPABILITIES FOR RADIOLOGICAL TRANSPORTATION INCIDENTS. VILARDO,F.J.; MITTER,E.L.; PALMER,J.A.; et al. Indiana Univ., Bloomington, IN. May 1990. 230pp. 9006290123. 54322:294.

This publication is the final report of a project to survey the fifty states, the District of Columbia, Puerto Rico, and selected Indian tribal jurisdictions to ascertain their emergency-preparedness planning and capabilities for responding to transportation incidents involving radioactive materials. The survey was conducted to provide the Nuclear Regulatory Commission and other federal agencies with information concerning the current level of emergency-response preparedness of the states and selected tribes and an assessment of the changes that have occurred since 1980 (when a similar survey was performed [NUREG/CR-1620]). There have been no major changes in the states' emergency-response planning strategies and field tactics. The changes noted included an increased availability of dedicated emergency-response vehicles, wider availability of specialized radiation-detection instruments, and higher proportions of police and fire personnel with training in the handling of suspected radiation threats. Most Indian tribes have no capability to evaluate suspected radiation threats and have no formal relations with emergency-response personnel in adjacent states. For the nation as a whole, the incidence of suspected radiation threats declined substantially from 1980 to 1988.

NUREG/CR-5404 V01: AUXILIARY FEEDWATER SYSTEM AGING STUDY. CASADA,D.A. Oak Ridge National Laboratory. March 1990. 183pp. 9004090260. ORNL-6566. 53314:311.

This report documents the results of a study of the Auxiliary Feedwater (AFW) System that has been conducted for the U.S. Nuclear Regulatory Commission's Nuclear Plant Aging Research Progam. The study reviews historical failure data available from the Nuclear Plant Reliability Data System, Licensee Event Report Sequence Coding and Search System, and Nuclear Power Experience data bases. The failure histories of AFW System components are considered from the perspectives of how the failures were detected and the significance of the failure. Results of a detailed review of operating and monitoring practices at a plant owned by a cooperating utility are presented. General system configurations and pertinent data are provided for Westinghouse and Babcock and Wilcox units.

NUREG/CR-5405: ANALYSIS OF SHELL-RUPTURE FAILURE DUE TO HYPOTHETICAL ELEVATED-TEMPERATURE PRESSURIZATION OF THE SEQUOYAH UNIT 1 STEEL CONTAINMENT BUILDING. MILLER,J.D. Sandia National Laboratories. February 1990. 100pp. 9003070222. SAND89-1650. 52811:132.

Sandia National Laboratories, as part of the Containment Integrity Programs under the sponsorship of the Nuclear Regulatory Commission (NRC), has developed analytical techniques for predicting the performance of light water reactor steel containment buildings subject to loads beyond the design basis. The analytical techniques are based on experience with large-scale steel containment model tests that provided important insights and experimental validation of the analytical methods. As a means of demonstrating these analytical techniques, the NRC asked Sandia to conduct a structural evaluation of an actual steel containment building. The objective of the analysis was to determine the actual pressure capacity and the mode, location, and size of failure, where a functional definition of failure is used. The purpose of this report is to document the calculations performed to determine the pressure limits for the shell-rupture mode of failure. General failure of the containment shell is predicted by application of a failure criterion to the results from finite element structural analyses. The failure criterion relates the calculated values of strain in the containment plates, due to internal-pressurization loading, to the ultimate strain limit of the steel. Included in the failure criterion are adjustments for factors inherent in finite element analysis, such as level of detail and element size of finite element model and variations in material property data. Separate finite element models were used to evaluate the overall free-field behavior of the structure and the localized behavior at a specific penetration location. Three scenarios of static internal pressurization, based on gases building up slowly within the containment shell during a severe accident, were evaluated. The scenarios included pressure loading with temperatures uniformly increasing in the finite element model in correspondence to the properties of pressurized saturated steam, pressure loading with non-uniformly increasing temperatures, and pressure loading without a corresponding temperature increase, which was done for comparison with earlier published analyses. It is concluded that thermal effects do not change the overall response of the structure or the general shell failure mode, compared to the response due to pressurization at ambient temperature. The reduction in the predicted internal-pressure capacity of the containment building at temperature corresponds to the reduction in the ultimate strength of the A516 Grade 60 steel due to the temperature increase.

NUREG/CR-5409: NEUTRON EXPOSURE PARAMETERS FOR THE METALLURGICAL TEST SPECIMENS IN THE SIXTH HEAVY-SECTION STEEL IRRADIATION SERIES. MILLER,L.F.; BALDWIN,C.A.; STALLMANN,F.W.; et al. Oak Ridge National Laboratory. May 1990. 49pp. 9005210206. ORNL/TM-11267. 53891:314.

The goal of the Heavy-Section Steel Irradiation (HSSI) Program Sixth Irradiation Series is to determine the effect of irradiation on the shape and shift of the crack-arrest toughness versus temperature curve. Two capsules which contained crack-arrest and Charpy V-notch test specimens have been irradiated at the Oak Ridge Research Reactor located at the Oak Ridge National Laboratory. These capsules have been disassembled, internal dosimeters have been analyzed, and exposure parameters are presented for each irradiation test specimen. This report describes the computational methodology for the least-squares adjustment of the dosimetry data with neutronics calculations, and presents exposure parameters at each test specimen location for the fluence rate greater than 1.0 MeV, fluence rate greater than 0.1 MeV, and displacements per atom. The specific activity of each dosimeter at the end of irradiation is listed in the Appendix.

NUREG/CR-5411: ELICITATION & USE OF EXPERT JUDGMENT IN PERFORMANCE ASSESSMENT FOR HIGH-LEVEL RADIOACTIVE WASTE REPOSITORIES. BONANO,E.J. Sandia National Laboratories. HORA,S.C. Hawaii, Univ. of, Hilo, HI. KEENEY,R.L.; et al. Southern California, Univ. of, Los Angeles, CA. May 1990. 94pp. 9006080176. SAND89-1821. 54075:122.

This report presents the concept of formalizing the elicitation and use of expert judgment in the performance assessment of high-level radioactive waste (HLW) repositories in deep geologic formations. The report begins with a discussion of (1) characteristics (advantages or disadvantages) of formalizing expert judgment, (2) examples of previous uses of expert judgment in radioactive waste programs, (3) criteria that can assist in deciding when to formalize expert judgment, and (4) the relationship of formal use of expert judgment to data collection and modeling. The current state of the art with respect to the elicitation, use, and communication of formal expert judgment is presented. The report concludes with a discussion on potential applications of formal expert judgment in performance assessment of HLW repositories.

NUREG/CR-5419: AGING ASSESSMENT OF INSTRUMENT AIR SYSTEMS IN NUCLEAR POWER PLANTS. VILLARAN,M.; FULLWOOD,R.; SUBUDHI,M. Brookhaven National Laboratory. January 1990. 139pp. 9003070250. BNL-NUREG-52212. 52803:129.

NRC Generic Issue 43, "Contamination of Instrument Air Lines", has been unresolved since 1980. The potential seriousness of this issue was reinforced in a 1987 study by the Office for Analysis and Evaluation of Operational Data. Aging of components within compressed air systems, leading to degraded function of the system, is the subject of this study. This work was performed under the auspices of the NRC's Office of Nuclear Regulatory Research as part of the Nuclear Plant Aging Research (NPAR) Program. The objective of this study was to identify all the aging modes and their causes, which should be mitigated to achieve a reliable operation of all safety-related air equipment. Also included is an interim review of typical maintenance activities for air systems in the nuclear power industry. The Phase 2 effort of this study will make recommendations for developing an effective maintenance program industry-wide to counter the effects of aging. The analysis of operating experience data revealed that aging degradation occurs in the compressed air system, and becomes a factor as the system ages. Normal wear of the system and contamination of the air dominate the problems of system failure. Existing maintenance programs within the industry lack uniformity, and quality assurance is not rigorous because the system is classified as non-safety.

NUREG/CR-5421: LAPUR USER'S GUIDE. OTADUY,P.J.; MARCH-LEUBA,J. Oak Ridge National Laboratory. January 1990. 77pp. 9003070237. ORNL-TM/11285. 52801:001.

LAPUR, a computer program in FORTRAN-IV, is a mathematical description of the core of a boiling water reactor. Its two linked modules, LAPURX and LAPURW, respectively solve the steady state governing equations for the coolant and fuel and the dynamic equations for the coolant, fuel, and neutron field in the frequency domain. General implementation descriptions are followed by a detailed description of input and output parameters of LAPURX and LAPURW. Sample inputs are included and stability benchmarks are noted.

NUREG/CR-5424: ELICITING AND ANALYZING EXPERT JUDGEMENT:A Practical Guide. MEYER,M.A.; BOOKER,J.M. Los Alamos National Laboratory. January 1990. 424pp. 9002120292. LA-11667-MS. 52574:303.

In this book we describe how to elicit and analyze expert judgment. Expert judgment is defined here to include both the experts' answers to technical questions and their mental processes in reaching an answer. It refers specifically to data that are obtained in a deliberate, structured manner that makes use of the body of research on human cognition and communication. Our aim is to provide a guide for lay persons in expert judgment. These persons may be from physical and engineering sciences, mathematics and statistics, business, or the military. We provide background on the uses of expert judgment and on the processes by which humans solve problems, including those that lead to bias. Detailed guidance is offered on how to elicit expert judgment ranging from selecting the questions to be posed of the experts to selecting and motivating the experts to setting up for and conducting the elicitation. Analysis procedures are introduced and guidance is given on how to understand the data base structure, detect bias and correlation, form models, and aggregate the expert judgments.

NUREG/CR-5435: ENVIRONMENTAL EFFECTS ON CORROSION IN THE TUFF REPOSITORY. BEAVERS,J.A.; THOMPSON,N.G. Cortest Columbus, Inc. February 1990. 52pp. 9003190298. 53012:133.

Cortest Columbus is investigating the long-term performance of container materials used for high-level waste packages as part of the information needed by the Nuclear Regulatory Commission to assess the Department of Energy's application to construct a geologic repository for high-level radioactive waste. The scope of work consists of employing short-term techniques, such as electrochemical and slow strain rate mechanical test techniques, to examine a wide range of possible failure modes. Long-term tests are being used to verify and further examine specific failure modes identified as important by the short-term studies. This report summarizes the results of a literature survey performed under Task 1 of the program. The survey focuses on the influence of environmental variables on the corrosion behavior of candidate container materials for the Tuff repository. Environmental variables considered included: radiation, thermal and microbial effects.

NUREG/CR-5436 V01: THE DEVELOPMENT AND EVALUATION OF PROGRAMMATIC PERFORMANCE INDICATORS ASSOCIATED WITH MAINTENANCE AT NUCLEAR POWER PLANTS.Main Report. WREATHALL,J.; FRAGOLA,J.; APPIGNANI,P.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). May 1990. 133pp. 9006080157. SAIC-90/1130. 54076:239.

This report summarizes the development and evaluation of programmatic performance indicators of maintenance. These indicators were selected by: (1) creating a formal framework of plant processes; (2) identifying features of plant behavior considered important to safety; (3) evaluating existing indicators against these features; and (4) performing statistical analyses for the selected indicators. The report recommends additional testing.

NUREG/CR-5436 V02: THE DEVELOPMENT AND EVALUATION OF PROGRAMMATIC PERFORMANCE INDICATORS ASSOCIATED WITH MAINTENANCE AT NUCLEAR POWER PLANTS.Appendices. WREATHALL,J.; FRAGOLA,J.; APPIGNANI,P.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). May 1990. 225pp. 9006080152. SAIC-90/1130. 54076:014.

See NUREG/CR-5436,V01 abstract.

NUREG/CR-5437: ORGANIZATION AND SAFETY IN NUCLEAR POWER PLANTS. MARCUS,A.; NICHOLS,M.; BROMILEY,P.; et al. Minnesota, Univ. of, Minneapolis, MN. May 1990. 214pp. 9006080166. 54077:098.

Perspectives from industry, academe, and the NRC are brought together to develop a logical framework that links organization factors and safety in nuclear power plant performance. The framework focuses on intermediate outcomes which can be predicted by organizational factors, and which are subsequently linked to safety. The intermediate outcomes are efficiency, compliance, quality, and innovation. The organization factors can be classified in terms of environment, context, organizational governance, organizational design, and emergent process. Initial empirical analyses were conducted on a limited set of hypotheses derived from the framework. One set of hypotheses concerned the relationships between efficiency, measured by critical hours and outage rate, and safety, measured by 5 NRC indicators. Results of the analysis suggest that critical hours and outage rates and safety, as measured in this study, are not related to each other. Hypotheses were tested concerning the effects on safety and efficiency of utility financial resources and the lagged recognition and correction of problems that accompanies the reporting of major violations and licensee event reports. Results suggest that both financial resources and organizational problem solving/learning have significant effects on the outcome variables when time is properly taken into account.

NUREG/CR-5438: BASIC CONSIDERATIONS IN PREDICTING ERROR PROBABILITIES IN HUMAN TASK PERFORMANCE. FLEISHMAN,E.A.; BUFFARDI,L.C.; ALLEN,J.A.; et al. George Mason Univ., Fairfax, VA. April 1990. 49pp. 9005040081. CBCS RPP #90-1. 53622:022.

It is well established that human error plays a major role in the malfunctioning of complex systems. This report takes a broad look at the study of human error and addresses the conceptual, methodological, and measurement issues involved in defining and describing errors in complex systems. In addition, a review of existing sources of human reliability data and approaches to human performance data base development is presented. Alternative task taxonomies, which are promising for establishing the comparability on nuclear and non-nuclear tasks, are also identified. Based on such taxonomic schemes, various data base prototypes for generalizing human error rates across settings are proposed.

NUREG/CR-5439: HUMAN FACTORS ISSUES ASSOCIATED WITH ADVANCED INSTRUMENTATION AND CONTROLS TECHNOLOGIES IN NUCLEAR PLANTS. CARTER,R.J.; UHRIG,R.E. Oak Ridge National Laboratory. June 1990. 202pp. 9007120224. ORNL/TM-11319. 54467:071.

A survey of advanced instrumentation and controls (I&C) technologies and associated human factors issues in the U.S. and Canadian nuclear industries was carried out. The purpose of the survey was to provide background for the development of regulatory policy, criteria, and guides for review of advanced I&C systems as well as human engineering guidelines for evaluating these systems. The survey found those components of the U.S. nuclear industry surveyed to be quite interested in advanced I&C, but very cautious in implementing such systems in nuclear facilities and power plants. The trend in the facilities surveyed is to experiment cautiously when there is an intuitive advantage or short-term payoff. The most advanced I&C systems were found in the Canadian CANDU plants, where the newest plant has digital systems in almost 100% of its control systems and in over 70% of its plant protection system. The hypothesis that properly "introducing digital systems increases safety" is supported by the Canadian experience. A number of safety- related human factors issues were derived from the results of this survey. They include: is an advanced I&C guideline equivalent to NUREG-0700 needed? What changes will there be in the role of the control room operator? The potential problem of information overload needs to be addressed. How should existing training technology be made applicable for advanced I&C? How will operator acceptance and trust be accomplished?

NUREG/CR-5447: DEPRESSURIZATION AS AN ACCIDENT MANAGEMENT STRATEGY TO MINIMIZE THE CONSEQUENCES OF DIRECT CONTAINMENT HEATING. HANSON,D.J.; GOLDEN,D.W.; CHAMBERS,R.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). October 1990. 124pp. 9011090143. EGG-2574. 55660:252.

Probabilistic Risk Assessments (PRAs) have identified severe accidents for nuclear power plants that have the potential to cause failure of the containment through direct containment heating (DCH). Prevention of DCH or mitigation of its effects may be possible using accident management strategies that intentionally depressurize the reactor coolant system (RCS). The effectiveness of intentional depressurization during a station blackout TMLB' sequence was evaluated considering the phenomenological behavior, hardware performance, and operational performance. Phenomenological behavior was calculated using the SCDAP/RELAP5 severe accident analysis code. Two strategies to mitigate DCH by depressurization of the RCS were considered. One strategy, called early depressurization, assumed that the reactor head vent and pressurizer power-operated relief valves (PORVs) were latched open at steam generator dryout. The second strategy, called late depressurization, assumed that the head vent and PORVs were latched open at a core exit temperature of about 922 K (1200F). Depressurization of the RCS to a low value that may mitigate DCH was predicted prior to reactor pressure vessel breach for both early and late depressurization. The strategy of late depressurization is preferred over early depressurization because there are greater opportunities to recover plant functions prior to core damage and because failure uncertainties are lessened.

NUREG/CR-5448: AGING EVALUATION OF CLASS 1E BATTERIES: SEISMIC TESTING. EDSON,J.L. EG&G Idaho, Inc. (subs. of EG&G, Inc.). August 1990. 233pp. 9008310221. EGG-2576. 55042:136.

This report presents the results of a seismic testing program on naturally aged class 1E batteries obtained from a nuclear plant. The testing program is a Phase II activity resulting from a Phase I aging evaluation of class 1E batteries in safety systems of nuclear power plants, performed previously as a part of the U.S. Nuclear Regulatory Commission's Nuclear Plant Aging Research Program and reported in NUREG/CR-4457. The primary purpose of the program was to evaluate the seismic ruggedness of naturally aged batteries to determine if aged batteries could have adequate electrical capacity, as determined by tests recommended by IEEE Standards, and yet have inadequate seismic ruggedness to provide needed electrical power during and after a safe shutdown earthquake (SSE) event. A secondary purpose of the program was to evaluate selected advanced surveillance methods to determine if they were likely to be more sensitive to the aging degradation that reduces seismic ruggedness. The program used twelve batteries naturally aged to about 14 years of age in a nuclear facility and tested them at four different seismic levels representative of the levels of possible earthquakes specified for nuclear plants in the United States. Seismic testing of the batteries did not cause any loss of electrical capacity.

NUREG/CR-5449: DETERMINATION OF THE NEUTRON AND GAMMA FLUX DISTRIBUTION IN THE PRESSURE VESSEL AND CAVITY OF A BOILILNG WATER REACTOR. ASGARI,M.; WILLIAMS,M.L.; KAM,F.B.K. Oak Ridge National Laboratory. June 1990. 115pp. 9006290142. ORNL/TM-11350. 54325:036.

The Grand Gulf Boiling Water Reactor (BWR/6), owned and operated by Mississippi Power & Light Company has been analyzed to determine the neutron and gamma energy spectrum and flux levels in regions from the reactor vessel throughout the concrete shield wall. Several two-dimensional and one-dimensional transport calculations were performed for the Grand Gulf reactor configuration. The results from these calculations were synthesized to obtain the three-dimensional neutron flux spectra and dosimeter activities. The results from the transport calculations indicate the flux above 1 MeV peaks near the axial midplane and azimuthal angle between 40 degrees and 45 degrees, depending on the radial locations. The peak flux above 1 MeV incident on the vessel and at midcavity is about 1.82 x 10(9) and 1.07 x 10(8) n x cm(-2) x s(-1), respectively. The vessel fluence accumulated during Cycle 2 and after 32 effective full power years is about 4.41 x 10(16) and 1.84 x 10(18) n x cm(-2) x s(-1), respectively. The peak flux above 1 MeV at the front of the concrete shield wall, 15.24 cm (6 in.) into the concrete wall, and 30.48 cm (1 ft) into the concrete wall is about 7.91 x 10(7), 7.24 x 10(6), and 6.44 x 10(5) n x cm(-2) x s(-1), respectively. The results obtained from the gamma calculations show that the peak gamma heating at the O-T location of the reactor pressure vessel has a value of 2.54 x 10(-3) W/g of stainless steel (SS 304). The peak gamma absorbed dose rate at the midcavity is about 7.31 x 10(3) rad/h at full power operation.

NUREG/CR-5450: HIGH-TEMPERATURE CRACK-ARREST TESTS USING 152-MM-THICK SEN WIDE PLATES OF LOW-UPPER-SHELF BASE MATERIAL:TESTS WP-2.2 AND WP-2.6. NAUS,D.J.; KEENEY-WALKER; BASS,B.R.; et al. Oak Ridge National Laboratory. February 1990. 143pp. 9003070238. ORNL/TM-11352. 52802:055.

Two 152-mm-thick wide-plate crack-arrest tests (WP-2 series) are discussed in this report. Each test used a 1 x 1 x 0.15 m thick single-edge-notch specimen (a/w = 0.2), fabricated from a low-upper-shelf base material, that was subjected to a linear thermal gradient along the plane of crack propagation. The tests were conducted at the National Institute of Standards and Technology and were designed to provide fracture-toughness measurements at temperatures approaching or above the onset of the Charpy upper-shelf regime in a rising toughness region and with an increasing driving force. Results obtained from these tests have produced crack-arrest toughness values well above the limit recognized by the current ASME guidelines (220 MPa $\cdot\sqrt{m}$) with arrests occurring at 44 to 102 degrees C above the material DW(NDT) (60 degrees C). The fracture data support (1) the use of fracture mechanics concepts to analyze cleavage run-arrest events, (2) the treatment of cleavage run-arrest and ductile fracture modes as separate events, and (3) the fact that cleavage arrest occurs above the ASME limit.

NUREG/CR-5451: CRACK-ARREST BEHAVIOR IN SEN WIDE PLATES OF LOW-UPPER-SHELF BASE METAL TESTED UNDER NONISOTHERMAL CONDITIONS: WP-2 SERIES. NAUS,D.J.; KEENEY-WALKER; BASS,B.R.; et al. Oak Ridge National Laboratory. August 1990. 299pp. 9009170076. ORNL-6584. 55201:163.

Six wide-plate crack-arrest tests (WP-2 Series) are discussed in this report. Each test utilized either a 1 x 1 x 0.1-m or a 1 x 1 x 0.15-m thick single-edge notch specimen (a/w = 0.2), fabricated from a low-upper-shelf base material, that was subjected to a linear thermal gradient along the plane of crack propagation. The tests were conducted at the National Institute of Standards and Technology and were designed to provide fracture-toughness measurements at temperatures approaching or above the onset of the Charpy upper-shelf regime, in a rising toughness region, and with an increasing driving force. Results obtained from these tests have produced crack-arrest toughness values well above the limit recognized by the current ASME guidelines (220 MPa $\cdot\sqrt{m}$) with arrests occurring at up to 102 degrees C above the material DW(NDT)(60 degrees C). The fracture data support: (1) use of fracture mechanics concepts to analyze cleavage run-arrest events, (2) treatment of cleavage and ductile fracture modes as separate events, and (3) fact that cleavage arrest occurs above the ASME limit.

NUREG/CR-5453 V05: BACKGROUND INFORMATION FOR THE DEVELOPMENT OF A LOW-LEVEL WASTE PERFORMANCE ASSESSMENT METHODOLOGY.Computer Code Implementation And Assessment. KOZAK,M.W.; CHU,M.S.Y.; MATTINGLY,P.A.; et al. Sandia National Laboratories. August 1990. 105pp. 9008310184. SAND89-2509. 55044:055.

This report documents the implementation and assessment of computer codes for a low-level waste performance assessment methodology. Computer codes and analytical solutions are implemented for ground-water flow and transport analyses, source-term analyses, surface-water transport analyses, air-transport analyses, food-chain analyses, and dosimetry analyses. The capability has been retained to perform either simple or more complicated analyses of the source term and ground-water transport aspects of the performance assessment. The simple approaches consist of analytical and simple numerical analyses that are appropriate for relatively simple conceptual models. For fully multi-dimensional or transient problems, more complicated numerical solutions are recommended. Details are given of the recommended analytical methods, together with sensitivity analyses that demonstrate important aspects of the solutions. The implementation processes for the more complicated computer codes and those problems that arose during implementation are discussed. Finally, a comparison is given between the simple and complicated ground-water transport analyses for a simple conceptual model.

NUREG/CR-5460: A CAUSE-DEFENSE APPROACH TO THE UNDERSTANDING AND ANALYSIS OF COMMON CAUSE FAILURES. PAULA,H.; CAMPBELL,D.; et al. JBF Associates, Inc. PARRY,G. NUS Corp. March 1990. 137pp. 9005020340. SAND89-2368. [illegible]90:038.

This report presents the results of research to develop a new methodology for common cause failure analysis and prevention. Common cause failures are defined as those caused by single events that can make redundant components unavailable, thus compromising the reliability of redundant trains of equipment. Probabilistic risk assessments (PRAs) and nuclear power plant (NPP) operations have demonstrated that CCF events are often major contributors to the potential risk posed at such plants. Over the years, qualitative and quantitative analysis methods have been developed to improve CCF analysis. However, these methodologies have not explicitly accounted for the impact of plant-specific defenses, such as design features and operational and maintenance policies, in reducing the likelihood of failures at NPPs. The research documented in this report describes a cause-defense methodology for CCF analysis and prevention to help correct this deficiency. The report discusses the development of (1) procedures for identifying the potential for CCF events at individual NPPs and (2) cause-defense matrices for analyzing CCF events. New concepts and more precise definitions are introduced to enhance CCF terminology and interpretation of historical event data.

NUREG/CR-5461: AGING OF CABLES, CONNECTIONS, AND ELECTRICAL PENETRATION ASSEMBLIES USED IN NUCLEAR POWER PLANTS. JACOBUS,M.J. Sandia National Laboratories. July 1990. 81pp. 9008080195. SAND89-2369. 54871:328.

This report examines effects of aging on cables, connections, and containment electrical penetration assemblies (EPAs). Aging is defined as the cumulative effects that occur to a component with the passage of time. If unchecked, these effects can lead to a loss of function and a potential impairment of plant safety. This study includes a review of component usage in nuclear power plants; a review of some commonly used components and their materials of construction; a review of the stressors that the components might be exposed to in both normal and accident environments; a compilation and evaluation of industry failure data; a discussion of component failure modes and causes; and a brief description of current industry testing and maintenance practices.

NUREG/CR-5463: EFFECTS OF MINERALOGY ON SORPTION OF STRONTIUM AND CESIUM ONTO CALICO HILLS TUFF. MEYER,R.E.; ARNOLD,W.D.; CASE,F.I.; et al. Oak Ridge National Laboratory. April 1990. 42pp. 9005040072. ORNL-6589. 53622:071.

Sorption and desorption measurements were made of strontium and cesium onto clinoptilolite and Calico Hills Tuff. The object was to see whether there was a correlation between sorption of strontium and cesium onto Calico Hills Tuff and the sorption of strontium and cesium onto clinoptilolite based on the content of clinoptilolite in the Calico Hills Tuff. If sorption onto Calico Hills Tuff is solely due to the presence of clinoptilolite, then the ratios of the sorption ratios on tuff to those on clinoptilolite at similar conditions should be the weight fraction of the clinoptilolite on the tuff. Since the tuff contained about 50% clinoptilolite, the ratios would be expected to be about 0.5 if sorption was due solely to clinoptilolite. The experimental evidence showed that the ratios were generally near 0.5 for both cesium and strontium sorption and that ion-exchange processes were operative for both the clinoptilolite and the tuff. However, the ratios differed to a small extent for the different conditions, and there were indications that other sorption processes were involved.

NUREG/CR-5468: QADS: A MULTIDIMENSIONAL POINT KERNEL ANALYSIS MODULE. BROADHEAD,B.L. Oak Ridge National Laboratory. May 1990. 66pp. 9006080320. ORNL/CSD/TM-270. 54075:054.

QADS is a multidimensional point kernel computer code that utilizes the simplified free-form input of the SCALE system as well as compatibility with ORIGEN-S produced sources, SCALE cross section libraries, and standard composition data sets. QADS consists of a preprocessor that takes the free-form input and prepares input for the widely available QAD-CGGP code, which is then automatically executed by a driver module. This report describes the point kernel theory briefly, followed by numerous tips on successfully applying the theory to various types of shielding problems. The remainder of the document is devoted to input and output descriptions of the QADS code with several illustrative sample problems.

NUREG/CR-5472: A RISK-BASED REVIEW OF INSTRUMENT AIR SYSTEMS AT NUCLEAR POWER PLANTS. DEMOSS,G.; LOFGREN,E.; ROTHLEDER,B.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). January 1990. 165pp. 9003070261. BNL-NUREG-52220. 52802:324.

The broad objective of this analysis was to provide risk-based information to help focus regulatory actions related to Instrument Air (IA) systems at operating nuclear power plants. We first created an extensive data base of summarized and characterized IA-related events that gave a qualitative indication of the nature and severity of these events. Additionally, this data base was used to calculate the frequencies of certain events, which were used in the risk analysis. The risk analysis consisted of reviewing published PRAs and NRC Accident Sequence Precursor reports for IA-initiated accident sequences, IA interactions with frontline systems, and IA-related risk significant events. Sensitivity calculations were performed when possible. Generically, IA was found to contribute less to total risk than many safety systems; however, specific design weaknesses in safety systems, non-safety systems, and the IA system were found to be significant in risk.

NUREG/CR-5473: INCLUSION OF UNSTABLE DUCTILE TEARING AND EXTRAPOLATED CRACK-ARREST TOUGHNESS DATA IN PWR VESSEL INTEGRITY ASSESSMENT. DICKSON,T.L.; CHEVERTON,R.D.; SHUM,D.K. Oak Ridge National Laboratory. May 1990. 35pp. 9007120177. ORNL/TM-11450. 54470:164.

Over the past several years, the Heavy-Section Steel Technology Program at Oak Ridge National Laboratory has performed a series of large-scale fracture-mechanics experiments. These experiments have demonstrated that prototypical nuclear reactor vessel steels can exhibit crack-arrest toughness values considerably above 220 MPa $\bullet\sqrt{m}$ although arrest can be followed immediately by unstable ductile tearing. This report evaluates the influence of the crack arrest toughness above 220 MPa $\bullet\sqrt{m}$ on the integrity assessments of nuclear reactor pressure vessels for pressurized-thermal shock (PTS) loading conditions, taking into account the potential for unstable ductile tearing following arrest. The influence of the high crack-arrest toughness data and unstable ductile tearing on pressurized water reactor vessel integrity assessment is PTS transient dependent. It appears that the potential benefit from crack-arrest events corresponding to toughness values above 240 MPa $\bullet\sqrt{m}$ for low upper-shelf weld (LUSW) material and above 370 MPa $\bullet\sqrt{m}$ for those vessels not containing LUSW material will usually be negated by unstable ductile tearing.

NUREG/CR-5474: ASSESSMENT OF CANDIDATE ACCIDENT MANAGEMENT STRATEGIES. LUCKAS,W.; VANDENKIEBOOM; LEHNER,J.R. Brookhaven National Laboratory. March 1990. 46pp. 9004100337. BNL-NUREG-52221. 53343:029.

A set of selected candidate accident management strategies, whose purpose is to prevent or mitigate in-vessel core damage, were developed from various NRC and industry reports. These strategies have been grouped in this report by the challenges they are intended to meet, and assessed to provide information which may be useful to individual licensees for consideration when they perform their individual Plant Examinations. Each assessment focused on describing and explaining the strategy, considering its relationship to existing requirements and practices as well as identifying possible associated adverse effects.

NUREG/CR-5475: MODEL FEASIBILITY STUDY OF RADIOACTIVE PATHWAYS FROM ATMOSPHERE TO SURFACE WATER. SMITH,R.E.; SUMMER,R.M.; FERREIRA,V.A. Agriculture, Dept. of. March 1990. 43pp. 9005020324. 53588:328.

A feasibility study of the ain ... to surface water radionuclide pathways was performe ... small catchments using a physically-based hydro-ecosystem model, Opus. Detail ... e-intensity (breakpoint) precipitation records from Arizo... d Georgia were used as input to drive the model. Tests of ... sel sensitivity to distribution coefficients, Kd, for Cs-137, Cs-134, and Sr-90 illustrated different vegetation-soil-erosion-runoff pathways in response to agricultural management practices. Results reflected the fact that low Kd values allow a radionuclide to infiltrate into the soil profile and isolate it from subsequent runoff and erosion. Of the radionuclides and physical settings studied only the Sr-90 with low Kd values is sufficiently mobile and long-lived to be removed from the system via percolation below the root zone. Conversely, highly-adsorbed radionuclides were subject to removal by adsorption to sediment particles and subsequent runoff. Comparison of different effective half-lives of I-131 demonstrated the importance of the timing of an erosion-runoff storm event during or immediately after a fallout event. Seasonal timing of a fallout event and crop management also affect the fate of this short-lived radionuclide. Removal by solution to surface water runoff was negligible for all nuclides studied. Model simulation results for up to 10 half-lives are corroborated by results from long-term field studies. These results show the feasibility of modeling pathways in small catchments using Opus.

NUREG/CR-5476: POSTTEST ANALYSIS OF A 1:6-SCALE REINFORCED CONCRETE REACTOR CONTAINMENT BUILDING. WEATHERBY,J.R. Sandia National Laboratories. February 1990. 99pp. 9004030052. SAND89-2603. 53226:001.

In an experiment conducted at Sandia National Laboratories, a 1:6-scale model of a reinforced concrete light water reactor containment building was pressurized with nitrogen gas to more than three times its design pressure. The pressurization produced one large tear and several smaller tears in the steel liner plate that functioned as the primary pneumatic seal for the structure. This report describes posttest finite element analyses of the 1:6-scale model test and compares pretest predictions of the structural response to the experimental results. Strains and displacements calculated in axisymmetric finite element analyses of the 1:6-scale model are compared to strains and displacements measured in the experiment. Detailed analyses of the liner plate are also described in the report. The results from these analyses indicate that the primary mechanisms that initiated the tear can be captured in a two-dimensional finite element model. Furthermore, the analyses show that studs which were used to anchor the liner to the concrete wall, played an important role in initiating the liner tear.

NUREG/CR-5477: AN EVALUATION OF THE RELIABILITY AND USEFULNESS OF EXTERNAL-INITIATOR PRA METHODOLOGIES. BUDNITZ,R.J.; LAMBERT,H.E. Future Resources Associates, Inc. January 1990. 111pp. 9002120348. 52577:129.

This report, prepared to assist policy-level decision-makers, evaluates the extent to which each category of external-initiators PRA methodology produces reliable and useful results and insights, at its current state-of-the-art level. This report addresses this need in the following five categories of external initiators: (1) earthquakes; (2) internal fires; (3) external floods; (4) extreme winds; and (5) transportation accidents. Each initiator is examined separately. The thrust is to identify and describe the principal aspects of the current state-of-the-art PRA methodology, what aspects are less robust and therefore provide less reliable insights, and why.

NUREG/CR-5478: IMPROVED EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING. Progress Report For January 1985 To December 1987. DODD,C.V.; DEEDS,W.E.; MCCLUNG,R.W. Oak Ridge National Laboratory. January 1990. 36pp. 9002120284. ORNL/TM-11389. 52597:255.

A major limitation of eddy-current inspection of steam generator tubing is that small flaw signals can be masked by the effects of benign variables, such as tube supports. To identify the critical flaw properties accurately and reliably in the presence of signals caused by these other property variations, we must have enough information to distinguish the flaw signals from the extraneous ones. Therefore, we developed instrumentation to measure both the amplitude and the phase of the eddy-current signal at several different frequencies, as well as computer equipment to process the data quickly and reliably. This need to detect small flaws in the presence perturbing property variations has also required the development of more sensitive and more complicated probes, such as pancake and reflection probes. These smaller coils can detect much smaller flaws and are less sensitive to artifacts outside the tube, such as tube supports, magnetite, or copper. By being pressed against the tube wall, they also avoid liftoff effects. To increase the inspection speed an array of these small coils has been constructed and tested. Finally, new and more complicated tube standards were constructed to include the range of property variations.

NUREG/CR-5479: CURRENT APPLICATIONS OF VIBRATION MONITORING AND NEUTRON NOISE ANALYSIS.Detection And Analysis Of Structural Degradation Of Reactor Vessel Internals From Operational Aging. DAMIANO,B.; KRYTER,R.C. Oak Ridge National Laboratory. February 1990. 43pp. 9004090190. ORNL/TM-11398. 53314:270.

This report, which was prepared under the Nuclear Plant Aging Research Program sponsored by the United States Nuclear Regulatory Commission, discusses the application of vibration monitoring and neutron noise analysis for monitoring light water reactor (LWR) vessel internals. The report begins by describing the effects of loss of structural integrity on internals vibration and how sensible parameters can be used to detect and track the progress of degradation. This is followed by a description and comparison of vibration monitoring and neutron noise analysis, two methods for monitoring the mechanical integrity of reactor vessel internal components. The major section of the report describes the status of reactor vessel internals condition monitoring programs in the United States, Federal Republic of Germany, and France, three countries having substantial commitments to nuclear power. The last section presents guidelines for U.S. utilities wishing to establish reactor internals condition monitoring programs.

NUREG/CR-5480: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-3. OSBORNE,M.F.; LORENZ,R.A.; COLLINS,J.L.; et al. Oak Ridge National Laboratory. June 1990. 70pp. 9006290135. ORNL/TM-11399. 54325:151.

Test VI-3, the third in a series of high-temperature fission product release tests in the vertical test apparatus, was conducted in flowing steam. 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 42 MWd/kg. Using an induction furnace, it was heated under simulated LWR accident conditions to two test temperatures, 20 min at 2000 K and then 20 min at 2700 K. The cladding was completely oxidized during the test, and very little melting or fuel-cladding interaction had occurred. 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 Kr-85, 5% for Ru-106, 99% for Sb-125, and 99% for both Cs-134 and Cs-137. Small release fractions for many other fission products were detected. In addition, very small amounts of fuel material - uranium and plutonium - were released. The total mass released from the furnace to the collection system was 3.17 g, 78% of which was collected on the filters. The results from this test were compared with previous tests in this series and with a commonly used model for fission product release.

NUREG/CR-5482: LABORATORY ANALYSIS OF FLUID FLOW AND SOLUTE TRANSPORT THROUGH A VARIABLY SATURATED FRACTURE EMBEDDED IN POROUS TUFF. CHUANG,Y.; HALDEMAN,W.R.; RASMUSSEN,T.C.; et al. Arizona, Univ. of, Tucson, AZ. February 1990. 332pp. 9003070258. 52601:083.

Laboratory techniques are developed that allow concurrent measurement of unsaturated matrix hydraulic conductivity and fracture transmissivity of fractured rock blocks. Two Apache Leap tuff blocks with natural fractures were removed from near Superior, Arizona, shaped into rectangular prisms, and instrumented in the laboratory. Porous ceramic plates provided solution to block tops at regulated pressures. Infiltration tests were performed on both test blocks. Steady flow testing of the saturated first block provided estimates of matrix hydraulic conductivity and fracture transmissivity. Fifteen centimeters of suction applied to the second block top showed that fracture flow was minimal and matrix hydraulic conductivity was an order of magnitude less than the first block saturated matrix conductivity. Coated-wire ion-selective electrodes monitored aqueous chlorided breakthrough concentrations. Minute samples of tracer solution were collected with filter paper. The techniques worked well for studying transport behavior at near-saturated flow conditions and also appear to be promising for unsaturated conditions. Breakthrough curves in the fracture and matrix, and a concentration map of chloride concentrations within the fracture, suggest preferential flow paths in the fracture and substantial diffusion into the matrix. Average travel velocity, dispersion coefficient and longitudinal dispersivity in the fracture are obtained.

NUREG/CR-5484: PH SENSORS BASED ON IRIDIUM OXIDE. TARLOV,M.J.; KREIDER,K.G.; SEMANCIK,S.; et al. National Institute of Standards & Technology (formerly National Bureau of Standa. March 1990. 21pp. 9004100360. 53344:092.

Results are presented on the pH-potential response of d.c. magnetron reactively sputtered iridium oxide films. The films exhibit a nearly Nernstian response to pH, no hysteresis effects, and minimal response to ionic interferences. Sensitivity to certain redox species is observed, however. In addition, methods are discussed for preparing model iridium oxide sensor surfaces for ultrahigh vacuum surface analytical studies. Stoichiometric IrO(2)-like surfaces are shown to be relatively inert to gas phase water. However, hydroxylation of IrO(2)-like surfaces can be induced by rf water plasma treatment.

NUREG/CR-5487: STATIC LOAD CYCLE TESTING OF A LOW-ASPECT-RATIO FOUR-INCH WALL,TRG-TYPE STRUCTURE TRG-5-4 (1.0,0.56). FARRAR,C.R.; BENNETT,J.G.; DUNWOODY,W.E.; et al. Los Alamos National Laboratory. November 1990. 158pp. 9011260032. LA-11739-MS. 55797:348.

This report is the second in a series of test reports that details the quasi-static cyclic testing of low height-to-length aspect ratio reinforced concrete structures. The test structures were designed according to the recommendations of a technical review group for the U.S. Nuclear Regulatory Commission sponsored Seismic Category I Structures Program. The structure tested and reported here had 4-inch-thick shear and end walls, and the elastic deformation was dominated by shear. The background of the program and previous results are given for completeness. Details of the geometry, material property tests, construction history, ultrasonic testing, and modal testing to find the undamaged dynamic characteristics of the structures are given. Next, the static test procedure and results in terms of stiffness and load deformation behavior are given. Finally, results are shown relative to other known results, and conclusions are presented.

NUREG/CR-5489: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS. EIDSON,A.F. Inhalation Toxicology Research Institute. June 1990. 82pp. 9007120243. LMF-124. 54468:105.

Protection of uranium mill workers from occupational exposure to uranium through routine bioassay programs and the assessment of accidental worker exposures are addressed. Comparisons of chemical properties and the biological behavior of refine uranium ore (yellowcake) are made to identify important properties that influence uranium distribution among organs. These studies will facilitate calculations of organ doses after exposures and associated health risk estimates and will identify important bioassay procedures to improve evaluations of human exposures. Samples of airborne uranium from operating mills and deposition models were used to predict appreciable deposition in the upper respiratory tract of workers, if respiratory protection were not used. Laboratory analyses of commercial yellowcake, and inhalation studies in rats, showed that inhalation of yellowcake aerosols might be considered to be inhalation of variable mixtures of ammonium diuranate and U(3)O(8). Studies of yellowcake clearance from rats after wound contamination showed that uranium behavior in vivo could not be quantitatively related to chemical composition. A biokinetic model of yellowcake inhaled by Beagle dogs was developed. Comparison with available data from human exposures showed that organ burdens in an exposed worker can be estimated from urinary bioassay results and in vivo counting, if the chemical composition, or soluble fraction, of the inhaled yellowcake is known.

NUREG/CR-5490 V01: REGULATORY INSTRUMENT REVIEW: MANAGEMENT OF AGING OF LWR MAJOR SAFETY-RELATED COMPONENTS. WERRY,E.V. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1990. 140pp. 9011150043. PNL-7190. 53737:185.

This report comprises Part I of a review of U.S. nuclear plant regulatory instruments to determine the amount and kind of information they contain on managing the aging of safety-related components in U.S. nuclear power plants. The review was conducted for the U.S. Nuclear Regulatory Commission (NRC) by the Pacific Northwest Laboratory under the NRC Nuclear Plant Aging Research (NPAR) Program. Eight selected regulatory instruments, e.g., NRC Regulatory Guides and the Code of Federal Regulations, were reviewed for safety-related information on five selected components: reactor pressure vessels, steam generators, primary piping, pressurizers, and emergency diesel generators. The focus of the review was on 26 NPAR- defined safety-related aging issues, including examination, inspection, and maintenance and repair; excessive/harsh testing; and irradiation embrittlement. The major conclusion of the review is that safety-related regulatory instruments do provide implicit guidance for aging management, but include little explicit guidance. A major recommendation is that the instruments be revised or augmented to explicitly address the management of aging.

NUREG/CR-5491: SHIPPINGPORT STATION AGING EVALUATION. ALLEN,R.P.; JOHNSON,A.B. Battelle Memorial Institute, Pacific Northwest Laboratory. January 1990. 143pp. 9002120344. PNL-7191. 52577:240.

The Shippingport Atomic Power Station, the first U.S. large-scale, central-station nuclear plant, now in the final stages of decommissioning, has been a major source of naturally aged equipment for the Nuclear Plant Aging Research (NPAR) and other U.S. Nuclear Regulatory Commission (NRC) programs. The evaluation of naturally aged components is an integral part of the NPAR program strategy. Because naturally aged components and materials experience the actual service-related external stressors, corrosion and wear, testing procedures, and maintenance practices, their evaluation is valuable in verifying degradation models, validating aging projections based on the extrapolation of accelerated test data, and detecting unexpected aging mechanisms (surprises) that could significantly impact component or system safety performance. As part of the Shippingport Station aging evaluation work, more than 200 items, ranging in size from small instruments and materials samples to one of the main coolant pumps, have been removed and shipped to designated NRC contractors. Although detailed evaluations of the components and material from the Shippingport Station are just beginning, the preliminary results from the studies conducted to date are indicative of the value of the aging information that ultimately may be obtained.

NUREG/CR-5492: INVESTIGATIONS OF IRRADIATION-ANNEAL-REIRRADIATION (IAR) PROPERTIES TRENDS OF RPV WELDS.Phase 2 Final Report. HAWTHORNE,J.R.; HISER,A.L. Materials Engineering Associates, Inc. January 1990. 302pp. 9002120277. MEA-2088. 52574:001.

Notch ductility, fracture toughness and tensile property trends of two high copper content, submerged arc welds with 288 degrees C irradiation (I), 399 degrees C postirradiation annealing (IA) and 288 degrees C reirradiation (IAR) were investigated. Primary objectives included the determination of weld metal reembrittlement rate with fluence following an anneal and the influence of first cycle fluence level (high or low) on reembrittlement susceptibility. The welds were commercially made using a single lot of filler wire and two welding fluxes (Linde 80 and Linde 0091). A relatively-rapid reembrittlement of both weld types was observed with initial reirradiation; however, the reembrittlement rate decreased markedly after a reirradiation fluence of about 0.3 x 10(19) n/cm(2), E greater than 1 MeV. In turn, the benefit of IA procedures toward reducing total properties change with fluence was retained. The reembrittlement trend was independent of first cycle fluence level for the range investigated. Residual embrittlement after 399 degrees C - 168 hour annealing, indexed by 41-J temperature, appears independent of the pre-anneal condition (I or IAR); however, wide variability among welds in percent recovery is indicated. Trends in IAR behavior from Charpy-V specimen tests were found to be reinforced by tensile specimen and fracture toughness (0.5T-CT specimen) test findings.

NUREG/CR-5493: INFLUENCE OF FLUENCE RATE ON RADIATION-INDUCED MECHANICAL PROPERTY CHANGES IN REACTOR PRESSURE VESSEL STEELS.Final Report On Exploratory Experiments. HAWTHORNE,J.R.; HISER,A.L. Materials Engineering Associates, Inc. March 1990. 347pp. 9004100351. MEA-2376. 53344:299.

This report describes a set of experiments undertaken using a 2 MW test reactor, the UBR, to qualify the significance of fluence rate to the extent of embrittlement produced in reactor pressure vessel steels at their service temperature. The test materials included two reference plates (A 302-B, A 533-B steel) and two submerged arc weld deposits (Linde 80, Linde 0091 welding fluxes). Charpy-V (C(v)), tension and 0.5T-CT compact specimens were employed for notch ductility, strength and fracture toughness (J-R curve) determinations, respectively. Target fluence rates were 8 x 10(10), 6 x 10(11) and 9 x 10(12) n/cm(2)-s(-1). Specimen fluences ranged from 0.5 to 3.8 x 10(19) n/cm(2), E greater than 1 MeV. The data describe a fluence-rate effect which may extend to power reactor surveillance as well as test reactor facilities now in use. The dependence of embrittlement sensitivity on fluence rate appears to differ for plate and weld deposit materials. Relatively good agreement in fluence-rate effects definition was observed among the three test methods.

NUREG/CR-5494: CORRELATION OF IRRADIATION-INDUCED TRANSITION TEMPERATURE INCREASES FROM C(V) AND K(JC)/K(IC) DATA.Final Report. HISER,A. Materials Engineering Associates, Inc. March 1990. 224pp. 9005020331. MEA-2377. 53590:175.

Reactor pressure vessel (RPV) surveillance capsules contain Charpy-V (C(v)) specimens, but many do not contain fracture toughness specimens; accordingly, the radiation-induced shift (increase) in the brittle-to-ductile transition region (ΔT) is based upon the ΔT determined from notch ductility (C(v)) tests. Since the ASME K(Ic) and K(Ir) reference fracture toughness curves are shifted by the ΔT from C(v), assurance that this ΔT does not underestimate ΔT associated with the actual irradiated fracture toughness is required to provide confidence that safety margins do not fall below assumed levels. To assess this behavior, comparisons of ΔT's defined by elastic-plastic fracture toughness and C(v) tests have been made using data from RPV base and weld metals in which irradiations were made under test reactor conditions.

NUREG/CR-5506: PRELIMINARY STRUCTURAL EVALUATION OF TROJAN RCL SUBJECT TO POSTULATED RPV SUPPORT FAILURE. LU,S.C. Lawrence Livermore National Laboratory. January 1990. 35pp. 9002120234. UCID-21831. 52611:224.

This report describes a preliminary structural evaluation made to determine whether the reactor coolant loop (RCL) piping of the Trojan nuclear power plant is capable of transferring the loads normally carried by the reactor pressure vessel (RPV) supports to other component supports in the RCL system if the RPV supports should fail, say from radiation damage. For the evaluation, we use the computer model of the RCL system of Unit 1 of the Zion nuclear power plant because it is readily available; the RCL systems of these two plants closely resemble each other. As a bounding case in the evaluation we postulate that all four RPV supports have failed. Two load combinations are evaluated: (1) the combination of dead weight, operating pressure, and the safe-shutdown earthquake, and (2) the combination of dead weight, operating pressure, and a loss-of-coolant accident. Both load combinations are classified as Level D Service Limits in accordance with the ASME Boiler and Pressure Vessel Code. Static and dynamic linear elastic analyses are conducted to comply with rules specified by Subsection NB in conjunction with Appendix F, Division 1, Section III of the ASME Code. Results of this preliminary evaluation indicate that ASME Code Appendix F requirements are satisfied by each of the load combinations considered in the analysis, leading to the conclusion that the Trojan RCL piping is capable of transferring the RPV support loads to the steam generator and reactor coolant pump supports.

NUREG/CR-5507: RESULTS FROM THE NUCLEAR PLANT AGING RESEARCH PROGRAM:THEIR USE IN INSPECTION ACTIVITIES. GUNTHER,W.; TAYLOR,J. Brookhaven National Laboratory. September 1990. 80pp. 9010240401. BNL-NUREG-52222. 55495:329.

The US NRC's Nuclear Plant Aging Research (NPAR) Program has determined the susceptibility to aging of components and systems and the potential for aging to impact plant safety and availability. The NPAR Program also identified methods for detecting and mitigating aging in components. This report describes the NPAR results which can enhance NRC inspection activities. Recommendations are provided for communicating pertinent information to NRC inspectors. These recommendations are based on a detailed assessment of the NRC's Inspection Program, and feedback from resident and regional inspectors as described within. Included are NPAR report summaries and aging inspection guides for components and systems which have been evaluated at BNL.

NUREG/CR-5510: EVALUATIONS OF CORE MELT FREQUENCY EFFECTS DUE TO COMPONENT AGING AND MAINTENANCE. VESELY,W.E.; KURTH,R.E.; SCALZO,S.M. Science Applications International Corp. (formerly Science Applications, Inc.). June 1990. 236pp. 9007120191. SAIC-89/1744. 54486:001.

A methodology is developed to incorporate aging effects into Probabilistic Risk Analyses (PRAs). The methodology separates the PRA analyses from the aging analyses, allowing available PRAs to be efficiently used in evaluating risk effects of aging. The methodology was applied to two NUREG-1150 PRAs using aging rate data that was developed for active components. Various surveillance and maintenance programs were evaluated to determine their effects in controlling aging. Both point evaluations and uncertainty evaluations were carried out. The results of the applications showed the sensitivity of aging effects on core melt frequency to the efficiency of the maintenance and surveillance program in managing aging effects. Th detailed contributors to the aging effects showed re' 'vely fe components contributing, implying that prioritized aging mar agement programs would be most effective in controlling risk.

NUREG/CR-5511: IRRADIATION EFFECTS ON STRENG AND TOUGHNESS OF THREE-WIRE SERIES-ARC STAINLESS STEEL WELD OVERLAY CLADDING. HAGGAG,F.M.; CORWIN,W.R.; NANSTAD,R.K. Oak Ridge National Laboratory. February 1990. 150pp. 9003190328. ORNL/TM-11439. 53044:223.

The potential for stainless steel cladding to improve the fracture behavior of an operating nuclear reactor pressure vessel, particularly during certain overcooling transients, may depend greatly on the properties of the irradiated cladding. Therefore, three-wire stainless steel cladding irradiated at temperatures and to fluences relevant to power reactor operation was examined. Postirradiation tensile testing results show that, from 125 to 288 degrees C, the yield strength increased by 8 to %, ductility increased insignificantly, with almost no change in ultimate tensile strength. All cladding exhibited ductile-to-brittle transition behavior during Charpy impact testing, because of the dominance of delta ferrite failures at low temperatures. On the upper shelf, energy was reduced 15 and 20%, and lateral expansion 13 and 41%, owing to irradiation exposure of 2 and 5 x 10(19) neutrons /CM(2) (greater than 1 MeV), respectively. In addition, radiation damage resulted in 13 and 28 degrees C shifts of the Charpy impact transition temperature for the low and high fluences, respectively. Irradiation exposure of 12.5-mm-thick compact specimens (0.5TCS), to a fluence of 2.41 x 10(19) neutrons/cm(2) (greater than 1 MeV), resulted in decreases in the initiation fracture toughness, J(Ic), and the tearing modulus.

NUREG/CR-5512 DRF FC: RESIDUAL RADIOACTIVE CONTAMINATION FROM DECOMMISSIONING. Technical Basis For Translating Contamination Levels To Annual Dose.Draft Report For Comment. KENNEDY,W.E.; PELOQUIN,R.A. Battelle Memorial Institute, Pacific Northwest Laboratory. January 1990. 317pp. 9003070227. PNL-7212. 52811:232.

This document describes the generic modeling of the total effective dose equivalent (TEDE) to an individual in a population from a unit concentration of residual radioactive contamination. Radioactive contamination inside buildings and soil contamination are considered. Unit concentration TEDE factors by radionuclide, exposure pathway, and exposure scenario are calculated. Reference radiation exposure scenarios are used to derive unit concentration TEDE factors for about 200 individual radionuclides and parent- daughter mixtures. For buildings, these unit concentration factors list the annual TEDE for volume and surface contamination situations. For soil, annual TEDE factors are presented for unit concentrations of radionuclides in soil during residential use of contaminated land and the TEDE per unit total inventory for potential use of drinking water from a ground-water source. Because of the generic treatment of potentially complex ground-water systems, the annual TEDE factors for drinking water for a given inventory may only indicate when additional site data or modeling sophistication are warranted. Descriptions are provided of the models, exposure pathways, exposure scenarios, parameter values, and assumptions used. An analysis of the potential annual TEDE resulting from reference mixtures of residual radionuclides is provided to demonstrate application of the TEDE factors.

NUREG/CR-5513 V01: ACCIDENT MANAGEMENT INFORMATION NEEDS.Volume 1 - Methodology Development And Application To A Pressurized Water Reactor (PWR) With A Large, Dry Containment. HANSON,D.J.; WARD,L.W.; NELSON,W.R.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). April 1990. 69pp. 9005070008. EGG-2592. 53656:277.

In support of the U.S. Nuclear Regulatory Commission (NRC) Accident Management Research Program, a methodology has been developed for identifying the plant information needs necessary for personnel involved in the management of an accident to diagnose that an accident is in progress, select and implement strategies to prevent or mitigate the accident, and monitor the effectiveness of these strategies. This report describes the methodology and presents an application of this methodology to a pressurized water reactor (PWR) with a large dry containment. A risk-important severe accident sequence for a PWR is used to examine the capability of the existing measurements to supply the necessary information. The method includes an assessment of the effects of the sequence on the measurement availability including the effects of environmental conditions. The information needs and capabilities identified using this approach are also intended to form the basis for more comprehensive information needs assessment performed during the analyses and development of specific strategies for use in accident management prevention and mitigation.

NUREG/CR-5513 V02: ACCIDENT MANAGEMENT INFORMATION NEEDS.Volume 2 - Appendices. HANSON,D.J.; WARD,L.W.; NELSON,W.R.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). April 1990. 304pp. 9005040140. EGG-2592. 53622:246.

See NUREG/CR-5513,V01 abstract.

NUREG/CR-5514: MODELING AND PERFORMANCE OF THE MHTGR REACTOR CAVITY COOLING SYSTEM. CONKLIN,J.C. Oak Ridge National Laboratory. April 1990. 36pp. 9005210212. ORNL/TM-11451. 53863:327.

The Reactor Cavity Cooling System (RCCS) of the Modular High Temperature Gas-Cooled Reactor (MHTGR) proposed by the U.S. Department of Energy is designed to remove the nuclear afterheat passively in the event that neither the heat transport system nor the shutdown cooling circulator subsystem is available. A computer dynamic simulation for the physical and mathematical modeling of an RCCS is described here. Two conclusions can be made from computations performed under the assumption of a uniform reactor vessel temperature. First, the heat transferred across the annulus from the reactor vessel and then to ambient conditions is very dependent on the surface emissivities of the reactor vessel and the RCCS panels. These emissivities should be periodically checked to ensure the safety function of the RCCS. Second, the heat transfer from the reactor vessel is reduced by a maximum of 10% by the presence of steam at 1 atm in the reactor cavity annulus for an assumed constant reactor vessel temperature of 500 K. Thus, the presence of a medium that participates in the transmission of radiant energy across the annulus can be expected to result in an increase in the reactor vessel temperature for the MHTGR. Further investigation of participating radiation media, including small particles, in the reactor cavity annulus is warranted.

NUREG/CR-5515: LIGHT WATER REACTOR PRESSURE ISOLATION VALVE PERFORMANCE TESTING. NEELY,H.H.; JEANMOUGIN,N.M.; CORUGEDO,J.J. Energy Technology Engineering Center. July 1990. 111pp. 9008070338. ETEC 88-01. 54860:014.

The Light Water Reactor Valve Performance Testing Program was initiated by the NRC to evaluate leakage as an indication of valve condition, provide input to Section XI of the ASME Code, evaluate motor signature testing to measure valve operability, evaluate acoustic emission monitoring for condition and degradation and in-service inspection techniques. Six typical check and gate valves were purchased for testing at typical plant conditions (550 F at 2250 psig) for an assumed number of cycles for a 40-year plant lifetime. Tests revealed that there were variances between the test results and the present statement of the Code; however, the testing was not conclusive. The lifecycle tests showed that high tech acoustic emission can be utilized to trend small leaks, that specific motor signature measurement on gate valves can trend and indicate potential failure, and that in-service inspection techniques for check valves was shown to be both feasible and an excellent preventive maintenance indicator. Lifecycle testing performance here did not cause large valve leakage typical of some plant operation. Other testing is required to fully understand the implication of these results and the required program to fully implement them.

NUREG/CR-5516: CAUSES OF FAILING THE DRAFT ANSI STANDARD N13.30 RADIOBIOASSAY PERFORMANCE CRITERION FOR MINIMUM DETECTABLE AMOUNT. MACLELLAN,J.A. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1990. 43pp. 9003070190. PNL-7217. 52813:026.

The test methods used for PNL bioassay performance tests were evaluated by comparing the MDA based on performance tests results with the MDA calculated by PNL using the bioassay laboratory's own quality control (QC) data. Two in vitro laboratories and two in vivo laboratories were studied and a correlation between the perfomance test MDA estimates and QC data was demonstrated. However, it was often necessary to examine the QC data to identify important characteristics of the blank distribution that affect the MDA calculation. Since the MDA equation must be based on the specific analysis and calculational methods of the procedure evaluated. Even when the correct MDA equation is applied, the MDA calculated will have a relatively large confidence interval when only a few replicates are used to estimate the standard deviation. For this reason, a relatively precise estimate of the MDA is generally only available when Poisson statistics may be applied. It was concluded that performance testing alone cannot provide all the information necessary to make an accurate estimate of the measurement process MDA. Review of the laboratory's QC data and the entire measurement procedure will be necessary. Specific recommendations for changes to draft ANSI N13.30 "Performance Criteria for Radiobioassay" are given.

NUREG/CR-5517: IMPACTS-BRC,VERSION 2.0.Program User's Manual. O'NEAL,B.L.; LEE,C.E. Sandia National Laboratories. April 1990. 244pp. 9005040021. SAND89-3060. 53649:308.

This manual describes the procedures for implementing IMPACTS-BRC Version 2.0. The IMPACTS-BRC computer code was designed for use by the Nuclear Regulatory Commission and industry to evaluate petitions to classify specific waste streams as below regulatory concern (BRC). The code provides a capability for calculating radiation doses to a maximal individual, critical group, and the general population as a result of transportation, treatment, disposal, and post-disposal activities involving low level radioactive waste. Impacts are calculated for multiple nuclides and pathways depending on the treatment/disposal options specified by the code user. The treatment/disposal options include onsite incineration, offsite incineration at municipal and hazardous waste facilities, and offsite disposal at municipal and hazardous waste landfills. Included within the disposal options is the ability to calculate impacts from the sorting and/or recycling of metal containers and metal and glass materials. Default environmental and facility parameters are developed from reference treatment/disposal sites, but the user has the option to replace default parameters with site-specific parameters. To facilitate use of the code, data input files are created and/or edited using a data preprocessor with pull-down menus and contact sensitive help screens. The code is written in FORTRAN and runs on 640K IBM-PC and compatible computers.

NUREG/CR-5519 V01: AGING OF CONTROL AND SERVICE AIR COMPRESSORS AND DRYERS USED IN NUCLEAR POWER PLANTS. MOYERS,J.C. Oak Ridge National Laboratory. July 1990. 129pp. 9008070332. ORNL-6607. 54850:224.

This report was produced under the Detection of Defects and Degradation Monitoring of Nuclear Plant Safety Equipment element of the Nuclear Plant Aging Research Program. This element includes the identification of practical and cost-effective methods for detecting, monitoring, and assessing the severity of time-dependent degradation (aging) of control and service air compressors and dryers in nuclear power plants. These methods are to provide capabilities for establishing degradation trends prior to failure and developing guidance for effective maintenance. The topics of this Phase I assessment report are failure modes and causes resulting from aging, manufacturer-recommended maintenance and surveillance practices, and measurable parameters (including functional indicators) for use in assessing operational readiness, establishing degradation trends, and detecting incipient failure. The results presented are based on information derived from operating experience records, manufacturer-supplied information, and input from plant operators. For each failure mode, failure causes are listed by sub-component, and parameters potentially useful for detecting degradation that could lead to failure are identified.

NUREG/CR-5521: USE OF PERFORMANCE ASSESSMENT IN ASSESSING COMPLIANCE WITH THE CONTAINMENT REQUIREMENTS IN 40 CFR PART 191. BONANO,E.J. Sandia National Laboratories. WAHI,K.K. Gram, Inc. September 1990. 53pp. 9010230042. SAND90-0127. 55473:167.

This report summarizes the role of performance assessment in assessing compliance with the containment requirements in 40 CFR Part 191, the Environmental Protection Agency's Standard for the disposal of spent nuclear fuel, high-level and transuranic radioactive wastes. In 1986, Hunter et al. prepared a similar report (NUREG/CR-4510, SAND86-0121) which provided an overview of the approach to assess compliance with this standard. The present report builds on its predecessor in that it incorporates advances in performance assessment subsequent to Hunter et al.'s report. The main purpose of this report is to serve as a mechanism for transferring to the Nuclear Regulatory Commission (NRC) and its contractors the performance assessment methodologies (PAMs) developed by Sandia National Laboratories (SNL) for high-level radioactive waste repositories. The report starts with a discussion of the requirements in 40 CFR Part 191 and focuses on the containment requirements (Section 191.13). It follows with a discussion of the role of performance assessment and its use in regulatory compliance. The report concludes with a discussion of sources of uncertainty, treatment of uncertainties, and the construction of the complementary cumulative distribution function of summed normalized total releases to the accessible environment for one or more scenarios. Examples are presented of the demonstration of performance assessment methodologies for high-level waste disposal at two hypothetical sites. Consistent with the technology transfer objective, numerous references are made throughout this report to publications related to the SNL PAMs. As such, this is not a stand-alone report and the reader is encouraged to consult those references.

NUREG/CR-5523: DEVELOPMENT OF AN INFILTRATION EVALUATION METHODOLOGY FOR LOW-LEVEL WASTE SHALLOW LAND BURIAL SITES. SMYTH,J.D.; BRESLER,E.; GEE,G.W.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1990. 147pp. 9006290119. PNL-7356. 54323:164.

An infiltration evaluation methodology (IEM) has been developed to provide a consistent, well-formulated approach for evaluating field-scale infiltration and drainage at low-level waste sites. The IEM is designed to simulate significant factors and hydrologic conditions that determine or influence moisture infiltration, redistribution, and drainage in engineered cover and barrier systems. The IEM recognizes the sources of uncertainty in estimating moisture movement through engineered covers and quantifies their influences on infiltration and drainage estimates. The IEM is developed on the basis of regulatory requirements given in 10 CFR 61. Engineered cover design and data availability will be largely site specific. Because of the site-specific nature of the design and data availability, the IEM is a flexible framework that accepts various numerical models, closed-form analytical models, and uncertainty approaches. Two applications of the IEM (a closed-form analytical model and a series of integrated numerical models) are demonstrated for a hypothetical waste site.

NUREG/CR-5524 V01: TMI-2 VESSEL INVESTIGATION PROJECT (VIP) METALLURGICAL PROGRAM.Progress Report,January-September 1989. DIERCKS,D.R. Argonne National Laboratory. March 1990. 43pp. 9008090018. ANL-90/2. 54672:048.

This report summarizes the work performed by Argonne National Laboratory on the TMI-2 Vessel Investigation Project (VIP) Metallurgical Program during the nine months from the initiation of the program in January 1989 through September 1989. During the reporting period, archive material for the program was obtained from the lower head of the cancelled Midland nuclear reactor in Midland, MI, in the form of four plates. Chemical analyses and hardness measurements were performed on samples from the four plates, the as-received microstructure was characterized, and a tentative determination of rolling direction was made. Initial results from heat treatment experiments on the archive material indicate that those regions of the TMI-2 material where the maximum temperature exceeded 727 degrees C should be readily identifiable on the basis of microstructural observations. A series of round-robin mechanical tests and microstructural studies on the as-received archive material was developed, and specimens and specimen blanks for tensile and stress-rupture tests were distributed to the participating OECD laboratories. Two trial specimens cut from a plate of A36 plain-carbon structural steel by PCI Energy Systems using metal disintegration machining (MDM) were examined metallographically.

NUREG/CR-5524 V02: TMI-2 VESSEL INVESTIGATION PROJECT (VIP). METALLURGICAL PROGRAM.Progress Report,October 1989 - June 1990. DIERCKS,D.R. Argonne National Laboratory. November 1990. 79pp. 9011290026. ANL-90/34. 55880:106.

During the period from October 1989 through June 1990, a series of heat treatment experiments on archive material from the lower head of the Midland nuclear reactor was completed, the resulting microstructures were examined, and hardness values were determined. Round-robin microstructural characterizations and mechanical tests on the archive material were also completed by the participating Organisation for Economic Cooperation and Development (OECD) partner laboratories. Good agreement was generally obtained in these evaluations and tests. Decontamination of samples from the TMI-2 lower head is underway at ANL, and detailed microstructural and scanning electron microscope examinations of Specimen E-6 were carried out. Metallographic examination revealed that the surface cracks present extended through the stainless steel cladding, but continued for only about 3mm into the underlying ferritic steel base metal. Extensive secondary cracking of the cladding, not connected to the surface, was observed. The evaluation indicated that the base metal in the vicinity of the crack attained a maximum temperature of about 1000 to 1100 degrees C during the accident. The molten fuel apparently did not penetrate into the cracks and interact with the base metal.

NUREG/CR-5527: RISK SENSITIVITY TO HUMAN ERROR IN THE LASALLE PRA. WONG,S.; HIGGINS,J.; O'HARA,J.; et al. Brookhaven National Laboratory. March 1990. 153pp. 9004030071. BNL-NUREG-52228. 53226:169.

A sensitivity evaluation was conducted to assess the impact of human errors on the internal event risk parameters in the LaSalle plant. The results provide the variation in the risk parameters, namely, core melt frequency and accident sequence frequencies, due to hypothetical changes in human error probabilities. Also provided are insights derived from the results, which highlight important areas for concentration of risk limitation efforts associated with human performance.

NUREG/CR-5528: AN ASSESSMENT OF BWR MARK II CONTAINMENT CHALLENGES,FAILURE MODES, AND POTENTIAL IMPROVEMENTS IN PERFORMANCE. KELLY,D.L.; JONES,K.R.; DALLMAN,R.J.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). July 1990. 317pp. 9009040046. EGG-2593. 55049:270.

This report assesses challenges to BWR Mark II containment integrity that could potentially arise from severe accidents. Also assessed are some potential improvements that could prevent core damage or containment failure, or could mitigate the consequences of such failure by reducing the release of fission products to the environment. These challenges and improvements are analyzed via a limited quantitative risk/benefit analysis of a generic BWR/4 reactor with a Mark II containment. Point estimate frequencies of the dominant core damage sequences are obtained and simple containment event trees are constructed to evaluate the response of the containment to these severe accident sequences. The resulting containment release modes are then binned into source term release categories, which provide inputs to the consequence analysis. The output of the consequence analysis is used to construct an overall base case risk profile. Potential improvements and sensitivities are evaluated by modifying the event tree split fractions, thus generating a revised risk profile. Several important sensitivity cases are examined in order to evaluate the impact of phenomenological uncertainties on the final results.

NUREG/CR-5530: ANALYSIS OF H.B. ROBINSON PWR VESSEL FLUENCE FOR CYCLE 10 UTILIZING PARTIAL LENGTH SHIELD ASSEMBLIES. CHILDS,R.L. Oak Ridge National Laboratory. WILLIAMS,M.L.; ASGARI,M. Louisiana State Univ., Baton Rouge, LA. September 1990. 95pp. 9010230106. ORNL/TM-11476. 55470:177.

Neutron transport calculations have been performed to determine the pressure vessel fluence and cavity dosimeter responses for cycle 10 of the H.B. Robinson pressurized water reactor. This cycle was the first to utilize "partial length shield assemblies" within the core to reduce the fluence rate at the critical weld location in the vessel. This work is part of the ongoing surveillance of the Robinson plant to insure that the projected fluence rates are reliable. The flux calculations utilize a "two-channel" synthesis approximation and recently processed iron cross sections based on a new evaluation for the inelastic data above 3 MeV. The methodology used to calculate this highly asymmetrical configuration is discussed in detail, and a comparison of the calculated and measured cavity-dosimetry results is presented. Discrepancies are observed in the computed and measured results for the (237)Np dosimeter, and possible explanations are discussed. Calculated absolute neutron flux spectra, as well as radial, azimuthal, and axial variations in the fast flux and dpa within the pressure vessel, are given. The effect of a least-squares consolidation of the measured and calculated results is studied.

NUREG/CR-5532: A PERFORMANCE ASSESSMENT METHODOLOGY FOR LOW-LEVEL WASTE FACILITIES. KOZAK,M.W.; CHU,M.S.Y.; MATTINGLY,P.A. Sandia National Laboratories. July 1990. 85pp. 9008080187. SAND90-0375. 54871:243.

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 a summary of background reports on the development of the methodology and an overview of the models and codes selected for the methodology. The overview includes discussions of the philosophy and structure of the methodology, and a sequential procedure for applying the methodology. Discussions are provided of models and associated assumptions that are appropriate for each phase of the methodology, the goals of each phase, data required to implement the models, significant sources of uncertainty associated with each phase, and the computer codes used to implement the appropriate models. In addition, a sample demonstration of the methodology is presented for a simple conceptual model.

NUREG/CR-5533: STATIC LOAD CYCLE TESTING OF A VERY LOW-ASPECT-RATIO SIX-INCH WALL TRG-TYPE STRUCTURE TRG-6-6 (0.27, 0.50). FARRAR,C.R.; BENNETT,J.G. Los Alamos National Laboratory. BAKER,W.E.; et al. New Mexico, Univ. of, Albuquerque, NM. November 1990. 64pp. 9012100345. LA-11796-MS. 55990:311.

This test report is the third for a series of tests carried out by the Los Alamos National Laboratory under the sponsorship of the United States Nuclear Regulatory Commission's Division of Engineering. This research program has a Technical Review Group that recommended test geometries and sizes for the tests. The quasi- static load cycle testing of a totally shear-dominated structure (bending deformation negligible) made of 6-inch-thick reinforced concrete walls is reported herein. The background of the program and the results that led to this series of experiments is first reviewed for continuity. Next, the geometry of the test structure, the design parameters, and the construction of the structure, including the material property tests, are reported. Both modal analysis and modal testing were done to verify the undamaged dynamic properties of the structure. Finally, the results of the quasi-static cycle testing are reported in detail. Results are compared with other investigations and with the American Concrete Institute (ACL) 349-85 code predictions.

NUREG/CR-5540: PERFORMANCE TESTING OF EXTREMITY DOSIMETERS,STUDY 2. HARTY,R.; REECE,W.D.; HOOKER,C.D. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1990. 89pp. 9005040084. PNL-7276. 53646:319.

The second of two performance tests of extremity dosimeters was conducted using a draft of a proposed standard for extremity dosimeter performance testing. The draft standard was written by the Health Physics Society Standards Committee (HPSSC) Working Group on the Performance Testing of Extremity Dosimeters. The initial performance test study (reported in NUREG/CR-4959) indicated that approximately 60% of the time the processors met the performance criterion specified in the draft standard. Because of these results, an investigation conducted to determine the sources of error during the first performance test indicated that errors occurred as a result of poor procedures, equipment malfunctions, and because processors were not prepared for the tests. A second performance test resulted in a passing rate of approximately 70% for ring dosimeters and 81% for wrist dosimeters. Although this is an overall improvement, the results indicated that most processors were unable to meet the performance criterion consistently for all irradiation categories. Variations in the results were also observed within specific categories as a result of the irradiation source. Recommendations for changes to the draft standard include dividing the beta particle and mixture categories, eliminating the neutron category, and changing the tolerance level for the performance criterion.

NUREG/CR-5542: MODELS FOR ESTIMATION OF SERVICE LIFE OF CONCRETE BARRIERS IN LOW-LEVEL RADIOACTIVE WASTE DISPOSAL. WALTON,J.C.; PLANSKY,L.E.; SMITH,R.W. EG&G Idaho, Inc. (subs. of EG&G, Inc.). September 1990. 52pp. 9010090060. EGG-2597. 55319:043.

Concrete barriers will be used as intimate parts of systems for isolation of low-level radioactive wastes subsequent to disposal. This work reviews mathematical models for estimating degradation rate of concrete in typical service environments. The models considered cover sulfate attack, reinforcement corrosion, calcium hydroxide leaching, carbonation, freeze/thaw, and cracking. Additionally, fluid flow, mass transport, and geochemical properties of concrete are briefly reviewed. Example calculations included illustrate the types of predictions expected of the models.

NUREG/CR-5545: VICTORIA: A MECHANISTIC MODEL OF RADIONUCLIDE BEHAVIOR IN THE REACTOR COOLANT SYSTEM UNDER SEVERE ACCIDENT CONDITIONS. HEAMES,T.J.; WILLIAMS,D.A.; BIXLER,N.E.; et al. Sandia National Laboratories. October 1990. 215pp. 9011150045. SAND90-0756. 55737:325.

This document provides a description of a model of the radionuclide behavior in the reactor coolant system (RCS) of a light water reactor during a severe accident. This document serves as the user's manual for the computer code called VICTORIA, based upon the model. The VICTORIA code predicts fission product release from the fuel, chemical reactions between fission products and structural materials, vapor and aerosol behavior, and fission product decay heating. This document provides a detailed description of each part of the implementation of the model into VICTORIA, the numerical algorithms used, and the correlations and thermochemical data necessary for determining a solution. A description of the code structure, input and output, and a sample problem are provided. The VICTORIA code was developed upon a CRAY-XMP at Sandia National Laboratories in the U.S.A. and a CRAY-2 and various SUN workstations at the Winfrith Technology Centre in England.

NUREG/CR-5547: APPLICATION OF SURFACE COMPLEXATION MODELS FOR RADIONUCLIDE ADSORPTION.Sensitivity Analysis Of Model Input Parameters. HAYES,K.F.; REDDEN,G.; ELA,W.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1990. 88pp. 9005210220. PNL-7239. 53904:280.

This report discusses activity in two areas: 1) an evaluation of methodologies currently used for the physical and chemical characterization of metal oxide and hydroxide adsorbents and 2) the sensitivity of the various surface complexation models' adsorbent input parameters for describing adsorption. The report describes the relative merits of three surface complexation models (SCM), procedures to estimate values of the model parameters from titration data and what is required of experimental titration data sets. The ultimate goal is to determine how and whether SCMs can successfully describe adsorption of contaminants from disposed nuclear wastes in natural soils and sediment. This study's results help clarify the applicability of SCMs, particularly with respect to their sensitivity to input parameters. A method is presented by which unique, best fit values for these parameters may be obtained. An appendix is provided that reviews methods for determining surface area, site density, particle size distribution and pore structure.

NUREG/CR-5548: REVIEW OF GEOCHEMICAL PROCESSES AND CODES FOR ASSESSMENT OF RADIONUCLIDE MIGRATION POTENTIAL AT COMMERCIAL LLW SITES. SERNE,T.J.; ARTHUR,R.C.; KRUPKA,K.M. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1990. 136pp. 9005210229. PNL-7285. 53888:074.

Information on geochemical processes that control contaminant solution concentrations and migration at existing LLW waste sites is summarized. The review identifies the current status and future information needs required for the development of effective performance assessment models for use in site license applications. Except for some reports on LLW disposal sites at Sheffield, IL, and West Valley, NY, few references were identified that contained adequate geochemical data necessary to model geochemical processes that affect migration. Tritium appears to be the most mobile radionuclide migrating from burial trenches at commercial LLW sites. The review identified microbial-degradation induced anoxia, subsequent iron oxide precipitation during oxidation, alkalinity controlled pH changes, and organic complexation reactions as key controls of radionuclide migration. The quantity of experimental and field data against which to test geochemical codes is very limited. All experimental work on radionuclide adsorption at commercial LLW sites relies upon the K(d(60)) concept. Studies of the effects of organics on radionuclide mobility suggest that a Co-EDTA chelate formed in the original waste may persist indefinitely and lead to enhanced migration. Other organic-radionuclide complexes are less stable and do not significantly enhance mobility under conditions anticipated at commercial LLW sites.

NUREG/CR-[illegible]52: AN OVERVIEW OF THE LOW UPPER SHELF TOUGHNESS SAFETY MARGIN ISSUE. MERKLE,J.G. Oak Ridge National Laboratory. August 1990. 62pp. 9009040031. ORNL/TM-11314. 55052:2[illegible]6.

The low upper shelf toughness issue has a long history, beginning with the choice of materials for the submerged arc welding process, but also potentially involving the use of A302-B plate. Criteria for vessels containing low upper shelf materials have usually been expressed in terms of the Charpy upper shelf impact energy. Although these criteria have had several different bases, the range of limiting values for wall thicknesses approaching nine inches has remained between 40 and 50 ft.lbs. Values for vessels with thinner walls and/or only circumferential low upper shelf welds could be less. A decision on criteria to be incorporated into the ASME Code is approaching. Choices to be made concern the method for estimating the decrease in upper shelf impact energy, flaw geometry for circumferential welds, statistical significance of toughness values, the choice between J(D) and J(M), reference pressure, safety factors, and the inclusion of instability pressure calculations by means of R curve extrapolation. This report presents a comprehensive overview of the issue, including history and recommendations for expediting its resolution.

NUREG/CR-5553: COMPUTER PROGRAMS FOR EDDY-CURRENT DEFECT STUDIES. PATE,J.R.; DODD,C.V. Oak Ridge National Laboratory. June 1990. 256pp. 9007120211. ORNL/TM-11505. 54472:307.

Several computer programs to aid in the design of eddy-current tests and probes have been written. The programs, written in Fortran, deal in various ways with the response to defects exhibited by four types of probes: the pancake probe, the reflection probe, the circumferential boreside probe, and the circumferential encircling probe. Programs are included which calculate the impedance or voltage change in a coil due to a defect, which calculate and plot the defect sensitivity factor of a coil, and which invert calculated or experimental readings to obtain the size of a defect. The theory upon which the programs are based is the Burrows point defect theory, and thus the calculations of the programs will be more accurate for small defects.

NUREG/CR-5554: RECOMMENDATIONS FOR THE SHALLOW-CRACK FRACTURE TOUGHNESS TESTING TASK WITHIN THE HSST PROGRAM. THEISS,T.J. Oak Ridge National Laboratory. September 1990. 55pp. 9010090068. ORNL/TM-11509. 55318:346.

Recommendations for the Heavy-Section Steel Technology Program's investigation into the influence of crack depth on the fracture toughness of a steel under conditions prototypic of those in a reactor pressure vessel are included in this report. The primary goal of the shallow- crack project is to investigate the influence of crack depth on fracture toughness under conditions prototypic of a reactor vessel. A limited data base of fracture-toughness values will be assembled using a beam specimen with a depth of 100 mm (4 in.) using prototypic reactor vessel material. Results of the investigation are expected to improve the understanding of shallow-flaw behavior in pressure vessels, thereby providing more realistic information for application to the pressurized- thermal-shock issues.

NUREG/CR-5556: REVIEW OF CURRENT LITERATURE RELATED TO GENERIC SAFETY ISSUE 15. LIPINSKI,R.E.; GARNER,R.W. EG&G Idaho, Inc. (subs. of EG&G, Inc.). July 1990. 28pp. 9008080197. EGG-2598. 54881:320.

Recent evaluations of surveillance samples in the High Flux Isotope Reactor at the Oak Ridge National Laboratory led to the conclusion that the embrittlement rates of several reactor pressure vessel (RPV) steels may be greater than originally anticipated. In June 1987, the Advisory Committee on Reactor Safeguards requested that the U.S. Nuclear Regulatory Commission investigate the consequences of embrittlement of RPV supports. This report summarizes the current literature related to these studies, evaluates their contribution toward resolving Generic Safety Issue 15 concerning material embrittlement, and recommends any further action considered appropriate. This review also contains a short discussion of the uses of structural mechanics and fracture mechanics to analyze embrittlement.

NUREG/CR-5557: RELAP5 THERMAL-HYDRAULIC ANALYSIS OF THE SNUPPS PRESSURIZED WATER REACTOR. KULLBERG,C.M. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1990. 89pp. 9006080313. EGG-2599. 54074:283.

Thermal-hydraulic analyses of five hypothetical accident scenarios were performed with the RELAP5 computer code for the Westinghouse Standardized Nuclear Unit Power Plant System pressurized water reactor. This work was sponsored by the U.S. Nuclear Regulatory Commission and is being done in conjunction with future analysis work at the U.S. Nuclear Regulatory Commission Technical Training Center in Chattanooga, TN. These accident scenarios were chosen to assess and benchmark the thermal hydraulic capabilities of the Technical Training Center Standardized Nuclear Unit Power Plant System simulator to model abnormal transient conditions.

NUREG/CR-5560: AGING OF NUCLEAR PLANT RESISTANCE TEMPERATURE DETECTORS. HASHEMIAN,H.M.; BEVERLY,D.D.; MITCHELL,D.W.; et al. Analysis & Measurement Services Corp. June 1990. 247pp. 9007120246. 54469:127.

An experimental research project was completed to identify the effects of normal aging on performance of nuclear safety-related RTDs. The limit for initial accuracy of these RTDs was established and the range of their response time was determined. Representative nuclear grade RTDs were tested for calibration drift at simulated reactor conditions and for shelf-life drift. This included a number of naturally aged RTDs received from nuclear power plants. The results of this work have shown that periodic calibration and response time testing performed once every fuel cycle is a reasonable approach for management of aging of nuclear grade RTDs.

NUREG/CR-5566: EVALUATION OF HEALTH EFFECTS IN SEQUOYAH FUELS CORPORATION WORKERS FROM ACCIDENTAL EXPOSURE TO URANIUM HEXAFLUORIDE. FISHER,D.R.; SWINT,M.J.; KATHREN,R.L. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1990. 60pp. 9006290116. PNL-7328. 54323:311.

Uranium urinalyses and medical laboratory results were studied to determine whether there were any health effects from uranium intake among a group of 31 workers exposed to uranium hexafluoride (UF_6) and hydrolysis products following the accidental rupture of a 14-ton shipping cylinder in early 1986 at the Sequoyah Fuels Corporation uranium conversion facility in Gore, Oklahoma. Physiological indicators studied to detect kidney tissue damage included tests for urinary protein, casts and cells, blood, specific gravity, and urine pH, blood urea nitrogen, and blood creatinine. We concluded after reviewing two years of follow-up medical data that none of the 31 workers sustained any observable health effects from exposure to uranium. The urinary excretion data were used to develop an improved systemic recycling model for inhaled soluble uranium. We estimated initial intakes, clearance rates, kidney burdens, and resulting radiation doses to lungs, kidneys, and bone surfaces. Radiation dose limits and limits on intake, as recommended by the ICRP, were not exceeded. However, the NRC derived limit of 9.6 mg was exceeded by eight of the 31 workers. Maximum kidney concentrations in exposed workers ranged from 0.05 to 2.5 μg U/g kidney tissue. We found no toxicological effects on the kidneys of workers at these concentrations.

NUREG/CR-5567: PWR DRY CONTAINMENT ISSUE CHARACTERIZATION. YANG,J.W. Brookhaven National Laboratory. August 1990. 204pp. 9009040026. BNL-NUREG-52234. 55046:311.

Severe accident issues have been characterized for pressurized water reactors with large dry containments. A description of PWR dry containment performance under severe accident conditions is provided. Reviews and discussions of early containment failure due to direct containment heating (DCH), in-vessel steam explosions, hydrogen burns and steam spikes, late containment failure due to gradual overpressurization and basemat melt-through, and containment bypass (interfacing systems LOCA) events are included. An assessment of potential improvements such as RCS depressurization, reactor cavity reflooding, hydrogen control, containment venting and accident management strategy is presented. The review and discussion are largely based on existing information obtained from the nuclear industry and the NRC's severe accident research programs. Additional analyses related to operator actions were performed and are presented in the appendices.

NUREG/CR-5568 V01: INDUSTRY BASED PERFORMANCE INDICATORS FOR NUCLEAR POWER PLANTS.Phase 1 Report: June 1989 - February 1990. CONNELLY,E.M.; VAN HEMEL,S.B. Communications Technology Applications, Inc. HAAS,P.M. Concord Associates, Inc. July 1990. 98pp. 9008090024. CTA 900215-025. 54872:094.

This report presents the results of the first phase of a two-phase study, performed with the goal of developing indirect (leading) indicators of nuclear power plant safety, using other industries as a model. It was hypothesized that other industries with similar public safety concerns could serve as analogs to the nuclear power industry. Many process industries have many more years of operating experience, and many more plants than the nuclear power industry, and thus should have accumulated much useful safety data. In Phase 1, the investigators screened a variety of potential industry analogs and chose a chemical/petrochemical manufacturing industry as the primary analog for further study. Information was gathered on safety programs and indicators in the chemical industry, as well as in the nuclear power industry. Frameworks were selected for the development of indicators which could be transferred from the chemical to the nuclear power environment, and candidate sets of direct and indirect safety indicators were developed. Estimates were made of the availability and quality of data in the chemical industry, and plans were developed for further investigating and testing these candidate indicators against safety data in both the chemical and nuclear power industries in Phase 2.

NUREG/CR-5572: AN EVALUATION OF THE EFFECTS OF LOCAL CONTROL STATION DESIGN CONFIGURATIONS ON HUMAN PERFORMANCE AND NUCLEAR POWER PLANT RISK. O'HARA,J.; RUGER,C.; HIGGINS,J.; et al. Brookhaven National Laboratory. September 1990. 71pp. 9009250057. BNL-NUREG-52236. 55243:312.

A human factors analysis was performed to assess how identified upgrades to local control stations (LCSs) in nuclear power plants affect both human performance and plant risk. Upgrades in the design of individual control panels and overall improvement of functional centralization were considered. The analysis methodology was accomplished in four stages. First, a list of LCS human engineering design deficiencies was developed using data collected from a variety of sources including visits to nuclear power plants. From these data, a set of potential upgrades were defined to correct the deficiencies. Second, the effects of the upgrades on human error probabilities (HEPs) were determined using a computer-based methodology for soliciting expert judgement. Third, the HEPs were propagated through a plant probabilistic risk assessm[illegible] (PRA), and new core melt frequencies were establi[illegible] preliminary, scoping value-impact assessment was [illegible] to evaluate the regulatory need for further review of possible action to improve the human factors engineering aspects of local control stations. The results indicated that implementation of both types of upgrades would improve human performance and lower risk, but that only the panel design improvements would be cost beneficial.

NUREG/CR-5573: BORON FLUSHING DURING A BWR ANTICIPATED TRANSIENT WITHOUT SCRAM. MIRKOVIC,D.; DIAMOND,D.J. Brookhaven National Laboratory. June 1990. 42pp. 9007120190. BNL-NUREG-52237. 54468:246.

This report documents a study of an accident sequence in a boiling water reactor (BWR) in which there is a large reactivity insertion due to the flushing of borated water from the core. This has the potential to occur during an anticipated transient without scram (ATWS) after the injection of borated water from the standby liquid control system. The boron shuts down the power, but if there is a rapid depressurization of the vessel (e.g., due to the inadvertent actuation of the automatic depressurization system), large amounts of low pressure, relatively cold, unborated water enters the vessel causing a rapid dilution and cooling. This study was carried out to determine if the reactivity addition caused by this flushing could lead to a power excursion sufficient to cause catastrophic fuel damage. Calculations were carried out using the RELAP5/MOD2 computer code under different assumptions regarding timing and availability of low pressure pumps and with different reactivity coefficients. The results showed that the fuel enthalpy rise was insufficient to cause catastrophic fuel damage although less severe fuel damage might still be possible due to the overheating of the fuel cladding.

NUREG/CR-5574: DETERMINATION OF THE CHEMICAL FORM OF TRITIUM IN SELF-LUMINOUS SIGNS. BOWERMAN,B.S.; CZAJKOWSKI,C.J. Brookhaven National Laboratory. May 1990. 43pp. 9006080314. BNL-NUREG-52238. 54075:009.

Building exit signs containing tritium self-luminous light sources were dismantled, and the light sources were tested to determine the chemical form of tritium in this study. The objective was to quantify the amounts of tritiated water (T(2)O or TOH) present in the light sources. The light sources consist of sealed glass tubes coated internally with a phosphor (zinc sulfide) and filled with tritium gas (T(2)). Light source tubes from four exit signs were tested. Two were new signs, one was six years old, and one was thirteen years old. In one of the new signs, the total tritium inventory included two percent tritiated water. Two signs had higher amounts of tritiated water: 4.5% for the other new sign, and 14.5% for the six-year-old sign. In the oldest sign, an accurate inventory of the tritium content was not available, but tritiated water accounted for 12.2% of the total tritium collected for counting.

NUREG/CR-5575: QUANTITATIVE ANALYSIS OF POTENTIAL PERFORMANCE IMPROVEMENTS FOR THE DRY PWR CONTAINMENT. KELLY,D.L.; PAFFORD,D.J.; SCHROEDER,J.A.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). August 1990. 163pp. 9009200008. EGG-2602. 55232:035.

This report calculates the risk benefit associated with potential performance improvements for the large dry pressurized water reactor (PWR) containment. The analysis is based on the June 1989 draft NUREG-1150 results for the Zion commercial nuclear reactor. Simplified containment event trees and the large accident progression event trees from draft NUREG-1150 are used to evaluate the effects of potential improvements on the response of the Zion containment to dominant severe accident sequences. Source terms are generated parametrically using the ZISOR code and offsite consequences are calculated with the MELCOR Accident Consequence Code System (MACCS). These results give point estimates of the risk reduction associated with each containment improvement identified by Brookhaven National Laboratory in their draft Issues Characterization Report.

NUREG/CR-5576: SURVEY OF BORIC ACID CORROSION OF CARBON STEEL COMPONENTS IN NUCLEAR PLANTS. CZAJKOWSKI,C.J. Brookhaven National Laboratory. June 1990. 45pp. 9007120198. BNL-NUREG-52239. 54468:203.

A review of licensee responses to Generic Letter No. 88-05 was performed by the U.S. Nuclear Regulatory Commission (USNRC). This review encompassed 50 satisfactory responses from the affected licensees. A series of ten (10) joint Brookhaven National Laboratory (BNL) and USNRC audits were performed on a selection of utilities. All of the licensees audited had program implementations which met the intent of the Generic Letter. A review of the available literature and the plant audits has led to the conclusion that the requirements of the Generic Letter have essentially been met for the 10 plants audited and it is recommended that resident inspectors verify that a documented and implemented program is in effect at their own plants.

NUREG/CR-5579: VALUE/IMPACT ASSESSMENT OF JET IMPINGEMENT LOADS AND PIPE-TO-PIPE IMPACT DAMAGE.Revised Methods And Criteria. BROWN,J.B.; BAMPTON,M.C.C.; ALZHEIMER,J.M. Battelle Memorial Institute, Pacific Northwest Laboratory. June 1990. 72pp. 9007120168. PNL-7339. 54470:267.

To account for effects that might result from a loss-of-coolant accident (LOCA), nuclear power plant designers have been required to analyze the effects of double-ended guillotine breaks (DEGB) in high-energy piping. The U.S. Nuclear Regulatory Commission (NRC), through its Standard Review Plan (SRP), requires that plant designers follow certain prescribed methods and criteria in the estimation of dynamic effects associated with the postulated rupture of piping. The work reported in this NUREG is intended to provide the basis for NRC decisions on adopting revisions to parts of the SRP 3.6.2 entitled "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping." The revisions considered in this work evaluated updated prescriptions for calculating jet impingement forces on critical systems and the requirement to consider pipe-whip damage to a new population of pipes. In accordance with the procedures documented in NUREG/CR-3568 entitled "A Handbook for Value-Impact Assessment", this report found indication that substantial costs and occupational radiation exposure would result from the proposed action without substantially reducing the risks to public health and safety.

NUREG/CR-5583: PREDICTION OF CHECK VALVE PERFORMANCE AND DEGRADATION IN NUCLEAR POWER PLANT SYSTEMS - WEAR AND IMPACT TESTS.Final Report,September 1988 - April 1990. KALSI,M.S.; HORST,C.L.; WANG,J.K.; et al. Kalsi Engineering, Inc. August 1990. 118pp. 9009070007. KEI-1656. 55070:010.

Check valve failures in nuclear power plants have led to safety concerns as well as extensive damage and loss of plant availability in recent years. Swing check valve internals may experience premature degradation if the disc is not firmly held open against its stop and significant flow disturbances are present upstream within 10 pipe diameters. The objective of the current Phase II research was to develop and experimentally verify a quantitative methodology for predicting swing check valve performance and the degradation of internals caused by hinge pin wear or disc stud impact. Phase I research had focussed on investigating the stability of the swing check valve disc at different flow velocities for a wide variety of upstream flow disturbances located within 10 pipe diameters of the check valve. Valve performance predictions based on methodology developed as a result of Phase I and II research correlate well with actual valve operating history at plants. The conservative guidelines provided by this methodology, tempered and refined by actual performance history and integrated with preventive maintenance activities, have the potential for significantly improving the overall reliability of check valves in nuclear power plants.

NUREG/CR-5584: RESULTS OF CRACK-ARREST TESTS ON TWO IRRADIATED HIGH-COPPER WELDS. ISKANDER,S.K.; CORWIN,W.R.; NANSTAD,R.K. Oak Ridge National Laboratory. December 1990. 197pp. 9101040295. ORNL/TM-11575. 56271:085.

The objective of this study was to determine the effect of neutron irradiation on the shift and shape of the lower-bound curve to crack-arrest data. Two submerged-arc welds with copper contents of 0.23 and 0.31 wt % were commercially fabricated in 220-mm-thick plate. Crack-arrest specimens fabricated from these welds were irradiated at a nominal temperature of 288 degrees C to an average fluence of 1.9 x 10(19) neutrons/cm(2) (greater than 1 MeV). Evaluation of the results shows that the neutron-irradiation-induced crack- arrest toughness temperature shift is about the same as the Charpy V-notch impact temperature shift at the 41-J energy level. The shape of the lower- bound curves (for the range of test temperatures covered) did not seem to have been altered by irradiation compared to those of the ASME K(Ia) curve.

NUREG/CR-5586: MITIGATION OF DIRECT CONTAINMENT HEATING AND HYDROGEN COMBUSTION EVENTS IN ICE CONDENSER PLANTS.Analyses With The CONTAIN Code And NUREG-1150 PRA Methodology. WILLIAMS,D.C.; GREGORY,J.J. Sandia National Laboratories. October 1990. 262pp. 9012110307. SAND90-1102. 56044:046.

Using Sequoyah as a representative plant, calculations have been performed with a developmental version of the CONTAIN computer code to assess the effectiveness of various possible improvements to ice condenser containments in mitigating severe accident scenarios involving direct containment heating (DCH) and/or hydrogen combustion. Mitigation strategies considered included backup power for igniters and/or air return fans, augmented igniter systems, containment venting, containment inerting, subatmospheric containment operation, reduced ice condenser bypass, and primary system depressurization. Various combinations of these improvements were also considered. Only inerting the containment or primary system depressurization combined with backup power supplies for the igniter systems resulted in large decreases in the peak pressures calculated to result from DCH events. Potential hydrogen detonation threats were also assessed; providing backup power for both the igniter systems and the air return fans would significantly reduce the potential for detonations but might not totally eliminate it. Sensitivity studies using the NUREG-1150 PRA methodology indicated that primary system depressurization combined with backup power for both igniters and fans could reduce the contribution to the mean risk potential of the class of events considered by about a factor of three.

NUREG/CR-5588 V01: CARES (COMPUTER ANALYSIS FOR RAPID EVALUATION OF STRUCTURES) VERSION 1.0.Seismic Module.Theoretical Manual. XU,J.; PHILIPPACOPOULOS; MILLER,C.A.; et al. Brookhaven National Laboratory. July 1990. 80pp. 9008070356. BNL-NUREG-52241. 54844:316.

During FY's 1988 and 1989, Brookhaven National Laboratory (BNL) developed the CARES system (Computer Analysis for Rapid Evaluation of Structures) for the U.S. Nuclear Regulatory Commission (NRC). CARES is a PC software system which has been designed to perform structural response computations similar to those encountered in licensing reviews of nuclear power plant structures. The documentation of the Seismic Module of CARES consists of three volumes. This report represents Volume 1 of the three volume documentation of the Seismic Module of CARES. It concentrates on the theoretical basis of the system and presents modeling assumptions and limitations as well as solution schemes and algorithms of CARES. The User's Manual is published as Volume 2 while solutions and results from a set of sample problems are published as Volume 3 of the CARES documentation.

NUREG/CR-5588 V02: CARES (COMPUTER ANALYSIS FOR RAPID EVALUATION OF STRUCTURES) VERSION 1.0.Seismic Module.User's Manual. XU,J.; PHILIPPACOPOULO; MILLER,C.A.; et al. Brookhaven National Laboratory. July 1990. 110pp. 9008070352. BNL-NUREG-52241. 54848:105.

During FY's 1988 and 1989, Brookhaven National Laboratory (BNL) developed the CARES system (Computer Analysis for Rapid Evaluation of Structures) for the U.S. Nuclear Regulatory Commission (NRC). CARES is a PC software system which has been designed to perform structural response computations similar to those encountered in licensing reviews of nuclear power plant structures. The documentation of the Seismic Module of CARES consists of three volumes. This report is Volume 2 of the three volume documentation of the Seismic Module of CARES and represents the User's Manual. Volume 1 concentrates on the theoretical basis of the system and presents modeling assumptions and limitations as well as solution schemes and algorithms of CARES. Solutions and results from a set of sample problems are published as Volume 3 of the CARES documentation.

NUREG/CR-5588 V03: CARES (COMPUTER ANALYSIS FOR RAPID EVALUATION OF STRUCTURES) VERSION 1.0.Seismic Module.Sample Problems. XU,J.; PHILIPPACOPOULO; MILLER,C.A.; et al. Brookhaven National Laboratory. July 1990. 146pp. 9008070359. BNL-NUREG-52241. 54848:215.

During FY's 1988 and 1989, Brookhaven National Laboratory (BNL) developed the CARES system (Computer Analysis for Rapid Evaluation of Structures) for the U.S. Nuclear Regulatory Commission (NRC). CARES is a PC software system which has been designed to perform structural response computations similar to those encountered in licensing reviews of nuclear power plant structures. The documentation of the Seismic Module of CARES consists of three volumes. This report represents Volume 3 of the three volume documentation of the Seismic Module of CARES. It presents three sample problems typically encountered in the Soil-Structure Interaction analyses. The theoretical bases, modeling assumptions and limitations as well as solution schemes and algorithms of the Seismic Module of CARES are given in Volume 1. The User's Manual is published as Volume 2.

NUREG/CR-5589: ASSESSMENT OF ICE-CONDENSER CONTAINMENT PERFORMANCE ISSUES. NOURBAKHSH,H.P. Brookhaven National Laboratory. July 1990. 57pp. 9008070396. BNL-NUREG-52242. 54844:259.

Vulnerabilities of an ice-condenser containment to challenges that could arise from severe accidents have been assessed. The phenomenological issues associated with containment challenges have been evaluated. A number of containment improvements which have the potential to mitigate severe accident challenges have been evaluated. This report is intended to provide a comprehensive statement of the relevant issues that can be used in the NRC staff's evaluation process and by the utilities during their individual plant examinations (IPEs).

NUREG/CR-5590: ASSESSMENT OF THE COMBUSTION MODEL IN THE HECTR CODE. PONG,L.T. Science Applications International Corp. (formerly Science Applications, Inc.). * Sandia National Laboratories. November 1990. 59pp. 9011260023. SAND90-7080. 55806:178.

HECTR (Hydrogen Event: Containment Transient Response) is a lumped-parameter containment analysis code developed to model the containment atmosphere during a nuclear reactor accident involving the release, transport, and combustion of hydrogen. A new set of flame speed and combustion completeness correlations has been included in HECTR. The combustion model in HECTR was assessed against a simple two-compartment problem, as well as the NTS (Nevada Test Site) and VGES (Variable Geometry Experimental System) experiments. The example using the two-compartment problem demonstrates that the combustion model in the modified HECTR code is capable of locating the flame position in a compartment with a hydrogen burn, and convecting material with the proper composition instead of a mixture gas through flow junctions. HECTR predictions compare reasonably well with the measured peak pressure ratios for 12 NTS premixed hydrogen experiments. It is concluded that the capability of the new correlations for flame speed and combustion completeness is sufficient for the NTS experiments. Based on the analyses of the NTS and VGES experiments, using a single room as a node is recommended for the nodalization in the HECTR combustion calculation.

NUREG/CR-5591 V01 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For October 1989 - March 1990. CORWIN,W.R. Oak Ridge National Laboratory. August 1990. 44pp. 9008310217. ORNL/TM-11568. 55043:011.

The primary goal of the Heavy-Section Steel Irradiation Program is to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior (particularly the fracture toughness properties) of typical pressure-vessel steels as they relate to light- water-reactor pressure-vessel integrity. The program includes direct continuation of irradiation studies previously conducted by the Heavy- Section Steel Technology Program augmented by enhanced examinations of the accompanying microstructural changes. Effects of specimen size; material chemistry; product form and microstructure; irradiation fluence, flux, temperature, and spectrum; and postirradiation annealing are examined on a wide range of fracture properties. Detailed statistical analyses of the fracture data on K(Ic) shift of high-copper welds were performed. Analysis of the first phase of irradiated crack-arrest testing on high-copper welds was completed. Final analysis and publication of the results of the second phase of the irradiation studies on stainless steel weld-overlay cladding were completed. Determinations were made of the variations in chemistry and unirradiated RT(NDT) of low upper-shelf weld metal from the Midland reactor. Final analyses were performed on the Charpy impact and tensile data from the Second and Third Irradiation series on low upper-shelf welds, and the report on the series was drafted. A detailed survey of existing data on microstructural models and data bases of irradiation damage was performed, and initial development of a reaction-rate-based model was completed.

NUREG/CR-5594: RADIATION DEGRADATION IN EPICOR-II ION EXCHANGE RESINS. MCCONNELL,J.W.; JOHNSON,D.A.; SANDERS,R.D. EG&G Idaho, Inc. (subs. of EG&G, Inc.). September 1990. 46pp. 9010230109. EGG-2603. 55470:125.

The Low-Level Waste Data Base Development-EPICOR-II Resin/Liner Investigation Program funded by the U.S. Nuclear Regulatory Commission is investigating chemical and physical conditions for organic ion exchange resins contained in several EPICOR-II prefilters. Those prefilters were used during cleanup of contaminated water from the Three Mile Island Nuclear Power Station after the March 1979 accident. The work was performed by EG&G Idaho, Inc. at the Idaho Engineering Laboratory. This is the final report of this task and summarizes results and analyses of three samplings of ion exchange resins from prefilters PF-8 and -20. Results are compared with baseline data from tests performed on unirradiated resins supplied by Epicor, Inc. to determine the extent of degradation due to the high internal radiation dose received by the organic resins. Results also are compared with those of other researchers.

NUREG/CR-5596: UNSATURATED FRACTURED ROCK CHARACTERIZATION METHODS AND DATA SETS AT THE APACHE LEAP TUFF SITE. RASMUSSEN,T.C.; EVANS,D.D.; SHEETS,P.J.; et al. Arizona, Univ. of, Tucson, AZ. August 1990. 146pp. 9009110122. 55123:350.

Performance assessment of high-level nuclear waste containment feasibility requires representative values of parameters as input, including parameter moments, distributional characteristics, and covariance structures between parameters. To meet this need, characterization methods and data sets for interstitial, hydraulic, pneumatic and thermal parameters for a slightly welded fractured tuff at the Apache Leap Tuff Site situated in central Arizona are reported in this document. The data sets include the influence of matric suction on measured parameters. Spatial variability is investigated by sampling along nine boreholes at regular distances. Laboratory parameter estimates for 105 core segments are provided, as well as field estimates centered on the intervals where the core segments were collected. Measurement uncertainty is estimated by repetitively testing control samples.

NUREG/CR-5597: IN-VESSEL ZIRCALOY-OXIDATION/HYDROGEN-GENERATION BEHAVIOR DURING SEVERE ACCIDENTS. CRONENBERG,A.W. Engineering Science & Analysis. September 1990. 57pp. 9010230038. 55473:107.

In-vessel zircaloy oxidation and hydrogen generation data from various U.S. Nuclear Regulatory Commission severe-fuel damage test programs are presented and compared, where the effects of zircaloy melting, bundle reconfiguration, and bundle quenching by reflooding are assessed for common findings. The experiments evaluated include fuel bundles incorporating fresh and previously irradiated fuel rods, as well as control rods. Findings indicate that the extent of bundle oxidation is largely controlled by steam supply conditions and that high rates of hydrogen generation continued after melt formation and relocation. Likewise, no retardation of hydrogen generation was noted for experiments which incorporated control rods. Metallographic findings indicate extensive oxidation of once-molten zircaloy bearing test debris. Such test results indicate no apparent limitations to zircaloy oxidation for fuel bundles subjected to severe-accident coolant-boiloff conditions.

NUREG/CR-5599: DYNAMIC TESTING OF A CIRCULAR FOUNDATION AND ANALYSES OF SOIL/STRUCTURE INTERACTION. SRINIVASAN,M.G.; KOT,C.A.; HSIEH,B.J. Argonne National Laboratory. October 1990. 196pp. 9011050063. ANL-90/26. 55643:158.

The concrete basemat of a one-quarter scale model of a nuclear power plant containment building was subjected to steady-state forced-vibration testing. The measured response was used to estimate the impedance functions for all rigid-body degrees of freedom and natural frequencies and damping for the fundamental modes. Blind predictions of the soil/structure interaction (SSI) during the vibration tests were made independently by other investigators. The tests, extraction of SSI parameters (including impedance functions from measured responses), results of the blind predictions, and various comparisons of test-determined parameters with their analytical counterparts are described. Analytical impedance functions based on a model of a rigid plate on an elastic half-space were found to have a similar trend and/or values as the test-determined functions in the frequency range containing the rigid-body modes but not outside that range. The discrepancies between analysis and experiment are traced to possible errors in test measurements and geophysical data, as well as to the assumptions of rigid-body motion and soil homogeneity.

NUREG/CR-5602: SIMPLIFIED CONTAINMENT EVENT TREE ANALYSIS FOR THE SEQUOYAH ICE CONDENSER CONTAINMENT. GALYEAN,W.J.; SCHROEDER,J.A.; PAFFORD,D.J. EG&G Idaho, Inc. (subs. of EG&G, Inc.). December 1990. 130pp. 9101140284. EGG-2606. 56391:001.

An evaluation of a Pressurized Water Reactor (PWR) ice condenser containment was performed. In this evaluation, simplified containment event trees (SCETs) were developed that utilized the vast storehouse of information generated by the NRC's draft NUREG-1150 effort. Specifically, the computer programs and data files produced by the NUREG-1150 analysis of Sequoyah were used to electronically generate SCETs, as opposed to the NUREG-1150 accident progression event trees (APETs). This simplification was performed to allow graphic depiction of the SCETs in typical event tree format, which facilitates their understanding and use. SCETs were developed for five of the seven plant damage state groups (PDSGs) identified by the NUREG-1150 analyses, which includes: both short- and long-term station blackout sequences (SBOs), transients, loss-of-coolant accidents (LOCAs), and anticipated transient without scram (ATWS). Steam generator tube rupture (SGTR) and event-VPDSGs were not analyzed because of their containment bypass nature. After being benchmarked with the APETs, in terms of containment failure mode and risk, the SCETs were used to evaluate a number of potential containment modifications. The modifications were examined for their potential to mitigate or prevent containment failure from hydrogen burns or direct impingement on the containment by the core (both factors identified as significant contributors to risk in the NUREG-1150 Sequoyah analysis). However, because of the relatively low baseline risk postulated for Sequoyah (i.e., 12 person-rems per reactor year), none of the potential modifications appear to be cost effective.

NUREG/CR-5603: PRESSURE-DEPENDENT FRAGILITIES FOR PIPING COMPONENTS.Pilot Study On Davis-Besse Nuclear Power Station. WESLEY,D.A.; KIPP,T.R.; NAKAKI,D.K.; et al. ABB Impell Corp. (formerly Impell Corp.). October 1990. 113pp. 9011090189. EGG-2607. 55661:016.

The capacities of four, low-pressure fluid systems to withstand pressures and temperatures above the design levels were established for the Davis-Besse Nuclear Power Station. The results will be used in evaluating the probability of plant damage from interfacing System Loss of Coolant Accidents (ISLOCA) as part of the probabilistic risk assessment of the Davis-Besse nuclear power station undertaken by EG&G Idaho, Inc. Included in this evaluation are the tanks, heat exchangers, filters, pumps, valves, and flanged connections for each system. The probabilities of failure, as a function of internal pressure, are evaluated as well as the variabilities associated with them. Leak rates or leak areas are estimated for the controlling modes of failure. The pressure capacities for the pipes and vessels are evaluated using limit-state analyses for the various failure modes considered. The capacities are dependent on several factors, including the material properties, modeling assumptions, and the postulated failure criteria. The failure modes for gasketed-flange connections, valves, and pumps do not lend themselves to evaluation by conventional structural mechanics techniques and evaluation must rely primarily on the results from ongoing gasket research test programs and available vendor information and test data.

NUREG/CR-5605: LAPUR BENCHMARK AGAINST IN-PHASE AND OUT-OF-PHASE STABILITY TESTS. MARCH-LEUBA,J. Oak Ridge National Laboratory. October 1990. 34pp. 9011050068. ORNL/TM-11621. 55614:184.

This paper documents a benchmark of the LAPUR code versus experimental stability data collected during startup testing at the Oskarshamn-3 reactor. The data consists of decay ratios and natural frequencies of oscillation measured under several reactor operating conditions. Satisfactory agreement was found between the measured decay ratios and the ones calculated by LAPUR for both the in-phase and out- of-phase instability modes. The largest error in the calculated decay ratio was 0.11, and it corresponded to test point 3 that is a very stable condition and, thus, hard to determine the decay ratio from the experimental measurements. The fact that LAPUR predicts that in test 7, 8, and 9 the out-of-phase mode should have been unstable (decay ratio 1.03) when indeed a limit cycle was observed implies that the theoretical approach used in LAPUR to estimate the stability of the out-of-phase mode is not only physically sound but also able to produce numerically correct results.

NUREG/CR-5607: FLOW AND TRANSPORT AT THE LAS CRUCES TRENCH SITE.Experiments 1 And 2. WIERENGA,P.J.; HUDSON,D.B. Arizona, Univ. of, Tucson, AZ. HILLS,R.G.; et al. New Mexico State Univ., Las Cruces, NM. August 1990. 426pp. 9009040036. 55051:200.

Two water flow and solute transport experiments were performed as part of a comprehensive field trench study near Las Cruces, New Mexico. These experiments were designed to provide data to test deterministic and stochastic models of vadose zone flow and transport. In Experiment 1, a 4 m by 9 m area was irrigated for 10 days with water containing tritium. Thereafter, water was applied without tritium for an additional 76 days. Simple one-dimensional uniform and layered soil deterministic models for infiltration adequately predicted the overall movement of the wetting front during infiltration, but poorly predicted point values for water content due to spatial variability. Use of the layered soil model, rather than the uniform soil model, did not consistently improve prediction accuracy for this particular field application. In Experiment 2, a 1.22 m by 12 m area was irrigated for 11.5 days with water containing tritium and bromide. Thereafter, water was applied without tracers for an additional 64 days. Water and bromide moved fairly uniformly during infiltration, whereas high concentrations of tritium developed on one side of the irrigated area. During redistribution, tritium moved little, whereas bromide displayed significant movement both downward and to one side. A two-dimensional deterministic model for water flow showed qualitative, but not quantitative, agreement with observations. A two-dimensional deterministic model for solute transport poorly described tritium and bromide movement during redistribution.

NUREG/CR-5613: PALEOLIQUEFACTION FEATURES ALONG THE ATLANTIC SEABOARD. AMICK,D.; GELINAS,R.; MAURATH,G.; et al. Ebasco Services, Inc. (subs. of Enserch Corp.). October 1990. 167pp. 9011200055. 55791:078.

This study is a phased investigation to determine in a systematic fashion, whether or not seismically induced paleoliquefaction features such as those observed in the Charleston, S.C., area are present elsewhere in young sediments of the Atlantic Coastal Plain. The discovery of similar liquefaction features in other areas could indicate that large, potentially damaging, earthquakes have not been restricted to the Charleston area in the recent geologic past. Conversely, if no evidence of similar liquefaction features is found, the uniqueness of the Charleston area in the context of eastern United States seismicity would tend to be confirmed. Phase 1 of this study centered on documenting the ages and characteristics of control liquefaction sites and features located in the Charleston area, and identifying the criteria by which similar sites and features, which may be located elsewhere, could be identified. Phase 2 investigations built on the results of these control studies and centered on the search for seismically induced paleoliquefaction features outside the epicentral area of the 1886 Charleston earthquake.

NUREG/CR-5615: LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY CLOSURE.Part I: Long-Term Environmental Conditions Affecting Low-Level Waste Disposal Site Performance.Part II: Performance Monitoring To Support Regulatory Decisions. WHITE,G.J.; FERNS,T.W.; OTIS,M.D.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). November 1990. 140pp. 9011260040. EGG-2604. 55797:208.

Part I of this report describes and evaluates potential impacts associated with changes in environmental conditions on a low-level radioactive waste disposal site over a long period of time. Part II of this report contains guidance on the design and implementation of a performance monitoring program for low-level radioactive waste disposal facilities.

NUREG/CR-5616: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE DIABLO CANYON UNIT 1 NUCLEAR POWER PLANT. GORE,B.F.; VO,T.V.; HARRISON,D.G. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1990. 32pp. 9010230060. PNL-7351. 55472:318.

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. Diablo Canyon-1 was selected as the first 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 Diablo Canyon-1 plant.

NUREG/CR-5617: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE J.M. FARLEY NUCLEAR POWER PLANT. VO,T.V.; PUGH,R.; GORE,B.F.; et al. Battelle Memorial Insitute, Pacific Northwest Laboratory. October 1990. 34pp. 9011090201. PNL-7349. 55661:129.

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 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. J.M. Farley was selected as the second 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 J.M. Farley plant.

NUREG/CR-5622: ANALYSIS OF REACTOR TRIPS ORIGINATING IN BALANCE OF PLANT SYSTEMS. STETSON,F.T.; GALLAGHER,D.W.; LE,P.T.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). September 1990. 215pp. 9010090026. SAIC-89/1148. 55323:064.

This report documents the results of an analysis of balance-of- plant (BOP) related reactor trips at commercial U.S. nuclear power plants over a 5-year period, from January 1, 1984, through December 31, 1988. The study was performed for the Plant Systems Branch, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. The objectives of the study were: 1) to improve the level of understanding of BOP-related challenges to safety systems by identifying and categorizing such events; 2) to prepare a computerized data base of BOP-related reactor trip events and use the data base to identify trends and patterns in the population of these events; 3) to investigate the risk implications of BOP events that challenge safety systems; and 4) to provide recommendations on how to address BOP-related concerns in a regulatory context.

NUREG/CR-5627: ALTERNATE MODAL COMBINATION METHODS IN RESPONSE SPECTRUM ANALYSIS. BEZLER,P.; CURRERI,J.R.; WANG,Y.K.; et al. Brookhaven National Laboratory. October 1990. 216pp. 9011090235. BNL-NUREG-52257. 55632:080.

This technical report was prepared to document the results of an investigation of alternate methods to combine between modal response components when using the response spectrum method of analysis. These methods are mathematically based to properly account for the combination between rigid and flexible modal responses as well as closely spaced modes. The methods are those advanced by Gupta, Hadjian and Lindley-Yow to address rigid response modes and the Double Sum Combination (DSC) method and the Complete Quadratic Combination (CQC) method to account for closely spaced modes. A direct comparison between these methods as well as the SRSS procedure is made by using them to predict the response of six piping systems. The results provided by each method are compared to the corresponding time history estimates of results as well as to each other. The degree of conservatism associated with each method is characterized.

NUREG/CR-5629: MICROEARTHQUAKES IN KANSAS AND NEBRASKA 1977-1989.Final Report. STEEPLES,D.W.; BENNETT,B.C.; PARK,C.; et al. Kansas, Univ. of, Lawrence, KS. October 1990. 39pp. 9011060118. 55616:170.

The Kansas Geological Survey operated a microearthquake network from August 1977 to August 1989 with stations located in eastern Kansas and Nebraska. Locatable microearthquakes with duration magnitudes less than 3.2 occur at the rate of roughly 20 per year in the two-state area, with most of the magnitudes ranging from 1.4 to 2.5. The microearthquake pattern observed during the 12 years of recording is consistent with the pattern of historical earthquakes reported since 1867. Much of the activity occurs along the Precambrian Nemaha Ridge, which has been the site of several earthquakes of MM Intensity VII over the past 125 years. Some seismicity is observed along the northwest flank of the Midcontinent Geophysical Anomaly in Kansas, but little is observed in the Nebraska or Iowa portions of this Precambrian feature. The Central Kansas Uplift, a buried anticline similar in age to the Nemaha Ridge, has been the site of several felt earthquakes since 1982. Another trend of earthquakes extends northeastward across central Nebraska and is not associated with any prominent geologic structure. All the seismicity in central and eastern Kansas can be roughly correlated to known geologic structures.

NUREG/CR-5631: CONTRIBUTION OF MATERNAL RADIONUCLIDE BURDENS TO PRENATAL RADIATION DOSES.Interim Recommendations.For Comment. SIKOV,M.R.; TRAUB,R.J.; MEZNARICH,H.K. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1990. 109pp. 9011160230. PNL-7445. 55754:283.

This report describes approaches to calculating and expressing radiation doses to the embryo/fetus from internal radionuclides. Information was obtained for selected, occupationally significant radioelements to provide a spectrum of metabolic and dosimetric characteristics. Fractional placental transfer and ratios of concentration in the embryo/fetus to that in the women were calculated for these materials, and were integrated with data from biokinetic transfer models to estimate radioactivity levels in the embryo/fetus as a function of stage of pregnancy and time after entry. The MIRD methodologies were extended to formalize and describe details for calculating radiation absorbed doses to the embryo/fetus. Calculations were performed for representative situations; introduction of 1 μCi into a woman's blood at successive months of pregnancy was assumed to accommodate the stage dependence of geometric relationships and biological behaviors. Summary tables of results, correlations, and dosimetric relations, and of tentative generalized categorizations are provided in the report. These approaches yield radiation absorbed doses, and multiplication by quality factor (Q) converts them to dose equivalent. This is the most common quantity for stating prenatal dose limits and is appropriate for the unique effects of prenatal exposure. Our knowledge is currently insufficient to warrant the use of radiation protection limits for prenatal radionuclide exposures that are based on lifetime detriments.

NUREG/CR-5635: CELLULAR AND MOLECULAR RESEARCH TO REDUCE UNCERTAINTIES IN ESTIMATES OF HEALTH EFFECTS FROM LOW-LEVEL RADIATION.A Feasibility Study. ELKIND,M.M.; BEDFORD,J.; et al. Colorado State Univ., Fort Collins, CO. GOTCHY,R.L. Science Applications International Corp. (formerly Science Applications, Inc.). October 1990. 309pp. 9011150052. 55738:180.

A study was undertaken by a cytogeneticist, a cell biologist, a health physicist, a mammalian biologist, and a molecular biologist to examine the feasibility of reducing the uncertainties in the estimation of risk due to protracted low doses of ionizing radiation. In addressing the question of feasibility, a review was made of current cellular, molecular, and mammalian radiation data of the way in which altered oncogenic properties could be involved in the loss of growth control that culminates in tumorigenesis; and of the significant progress that had been made in the oncogenic characterizations of several human and animal neoplasms. On the basis of this analysis, the study group concluded that, at the present time, it is feasible to mount a program of radiation research directed at the mechanism(s) of radiation-induced cancer with special reference to risk of neoplasia due to protracted, low doses of sparsely ionizing radiation. To implement a program of research, a review was made by the study group of the methods, techniques, and instruments that would be needed. This review was followed by an initial survey of major laboratories and institutions where scientific personnel and facilities sufficient to participate in a significant part of the program are known to be available. A research agenda of the principal and broad objectives of the program is also discussed.

NUREG/CR-5637: GENERIC RISK INSIGHTS FOR WESTINGHOUSE AND COMBUSTION ENGINEERING PRESSURIZED WATER REACTORS. TRAVIS,R.; TAYLOR,J.; FRESCO,A.; et al. Brookhaven National Laboratory. November 1990. 90pp. 9012100341. BNL-NUREG-52260. 55991:016.

A methodology has been developed to extract generic risk-based information from probabilistic risk assessments (PRAs) of Westinghouse and Combustion Engineering (CE) pressurized water reactors (PWRs) and apply the insights gained to Westinghouse and CE plants that have not been subjected to a PRA. The available PRAs (five Westinghouse plants and one CE plant) 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. Eleven 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-5638: TECHNICAL CONSIDERATIONS FOR EVALUATING SUBSTANTIALLY COMPLETE CONTAINMENT OF HIGH-LEVEL WASTE WITHIN THE WASTE PACKAGE. MANAKTALA,H.K. Center for Nuclear Waste Regulatory Analyses. INTERRANTE,C.G. Division of High-Level Waste Management (Post 870413). December 1990. 116pp. 9101100445. 56321:147.

This report deals with technical information that is considered essential for demonstrating the ability of the high-level radioactive waste package to provide "substantially complete containment" of its contents (vitrified waste form or spent light-water reactor fuel) for a period of 300 to 1000 years in a geological repository environment. The discussion is centered around technical considerations of the repository environment, materials and fabrication processes for the waste-package components, various degradation modes of the materials of construction of the waste packages, and inspection and monitoring of the waste package during the preclosure and retrievability period, which could begin up to 50 years after initiation of waste emplacement. The emphasis in this report is on metallic materials. However, brief references have been made to other materials such as ceramics, graphite, bonded ceramic-metal systems, and other types of composites. The content of this report was presented to an external peer review panel of nine members at a workshop held at the Center for Nuclear Waste Regulatory Analyses (CNWRA), Southwest Research Institute, San Antonio, Texas, April 2-4, 1990. The recommendations of the peer review panel have been incorporated into this report. There are two companion reports; the second report in the series provides the state-of-the-art techniques for uncertainty evaluations. The methods provided in that report can be used to quantify various types of uncertainties. The third companion report, on the basis of the information provided in the first two reports, develops recommendations for the resolution of the issue of "substantially complete containment" of high-level radioactive waste within the waste package, as addressed in 10 CFR Part 60.

NUREG/CR-5640: OVERVIEW AND COMPARISON OF U.S. COMMERCIAL NUCLEAR POWER PLANTS.Nuclear Power Plant System Sourcebook. LOBNER,P.; DONAHOE,C.; CAVALLIN,C. Science Applications International Corp. (formerly Science Applications, Inc.). September 1990. 601pp. 9010090058. SAIC-89/1541. 55320:212.

This report is the introductory volume to the Nuclear Power Plant Sourcebook Series and is intended as a source of current summary and comparative information on U.S. commercial light water reactors (LWRs). The summary and comparative information is organized into the following four parts: (a) general U.S. LWRs, (b) pressurized water reactors (PWRs), (c) boiling water reactors (BWRs), and (d) bibliographies of general PWR and BWR references, plant-specific references, system- specific references, and component-specific references. This report is supplemented by a set of Sourcebooks that provides more detailed information on specific U.S. LWR plants.

NUREG/CR-5644: CONSEQUENCE EVALUATION OF RADIATION EMBRITTLEMENT OF TROJAN REACTOR PRESSURE VESSEL SUPPORTS. LU,S.C.; SOMMER,S.C.; JOHNSON,G.L.; et al. Lawrence Livermore National Laboratory. October 1990. 119pp. 9011160222. UCRL-ID-104845. 55806:244.

This report describes a consequence evaluation to address safety concerns raised by the radiation embrittlement of the reactor pressure vessel (RPV) supports for the Trojan nuclear power plant. The study comprises a structural evaluation and an effects evaluation and assumes that all four reactor vessel supports have completely lost the load carrying capability. The structural evaluation indicates that the Trojan reactor coolant loop (RCL) piping is capable of transferring loads to the steam generator (SG) supports and the reactor coolant pump (RCP) supports. A subsequent analysis further demonstrates that the SG supports and the RCP supports have sufficient design margins to accommodate additional loads transferred to them through the RCL piping. The effects evaluation, employing a systems analysis approach, investigates initiating events and the reliability of the engineered safeguard systems as the RPV is subject to movements caused by the RPV support failure. The study concludes that a hypothetical failure of the Trojan RPV supports due to radiation embrittlement will not result in consequences of significant safety concerns.

NUREG/CR-5649 V01: COMMIX-1C: A THREE-DIMENSIONAL TRANSIENT SINGLE-PHASE COMPUTER PROGRAM FOR THERMAL-HYDRAULIC ANALYSIS OF SINGLE-COMPONENT AND MULTICOMPONENT ENGINEERING SYSTEMS.Equations And Numerics. DOMANUS,H.M.; CHA,Y.S.; CHIEN,T.H.; et al. Argonne National Laboratory. November 1990. 158pp. 9012280125. ANL-90/33. 56198:315.

The COMMIX-1C computer program, an extended version of previous single-phase COMMIX codes, is designed to analyze steady-state/ transient, single-phase three-dimensional fluid flow with heat transfer in reactor components and multicomponent systems. The concepts of volume porosity, directional surface porosity, distributed resistance, and distributed heat source or sink is used to model a flow domain with stationary structures. The new porous- medium formulation permits a simulation of either a single component or a multicomponent engineering system. The conservation equations of mass, momentum, and energy based on the new porous-medium formulation are solved as a boundary-value problem in space and an initial-value problem in time. Volume 1 of this report, entitled "Equations and Numerics," describes in detail the basic equations, formulations, solution procedures, flow-modulated skew-upwind discretization scheme, models to describe the auxiliary phenomena, etc. Volume 2, entitled "User's Guide and Manual," contains the flow charts, available options, input instructions, sample problems, etc.

NUREG/CR-5649 V02: COMMIX-1C: A THREE-DIMENSIONAL TRANSIENT SINGLE-PHASE COMPUTER PROGRAM FOR THERMAL-HYDRAULIC ANALYSIS OF SINGLE-COMPONENT AND MULTICOMPONENT ENGINEERING SYSTEMS.User's Guide And Manual. DOMANUS,H.M.; CHA,Y.S.; CHIEN,T.H.; et al. Argonne National Laboratory. November 1990. 199pp. 9012280137. ANL-90/33. 56222:047.

See NUREG/CR-5649 V01 abstract.

NUREG/CR-5653: RECRITICALITY IN A BWR FOLLOWING A CORE DAMAGE EVENT. SCOTT,W.B.; HARRISON,D.G.; LIBBY,R.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1990. 167pp. 9012280221. PNL-7476. 56227:349.

This report describes the results of a study conducted by Pacific Northwest Laboratory to assist the U.S. Nuclear Regulatory Commission in evaluating the potential for recriticality in boiling water reactors (BWRs) during certain low probability severe accidents. Based on a conservative bounding analysis, this report concludes that there is a potential for recriticality in BWRs if core reflood occurs after control blade melting has begun but prior to significant fuel rod melting. However, a recriticality event will most likely not generate a pressure pulse significant enough to fail the vessel. Instead, a quasi-steady power level would result and the containment pressure and temperature would increase until the containment failure pressure is reached, unless actions are taken to terminate the event. Two strategies are identified that would aid in regaining control of the reactor and terminate the recriticality event before containment failure pressures are reached. The first strategy involves initiating boration injection at or before the time of core reflood if the potential for control blade melting exists. The second strategy involves initiating residual heat removal suppression pool cooling to remove the heat load generated by the recriticality event and thus extend the time available for boration.

NUREG/CR-5659: CONTROL ROOM HABITABILITY SYSTEM REVIEW MODELS. GILPIN,H. Science Applications International Corp. (formerly Science Applications, Inc.). December 1990. 143pp. 9101040284. SAIC-90/1054. 56270:274.

This report provides a method of calculating control room operator doses from postulated reactor accidents and toxic chemical spills as part of the resolution of TMI Action Plan III.D.3.4. The computer codes contained in this report use source concentrations calculated by either TACT5, FPFP, or EXTRAN, and transport them via user-defined flow rates to the control room envelope. The codes compute doses to six organs from up to 150 radionuclides (or 1 toxic chemical) for time steps as short as one second. Supporting codes written in Clipper assist in data entry and manipulation, and graphically display the results of the FORTRAN calculations.

NUREG/IA-0011: TRAC-PF1/MOD1 POST-TEST CALCULATIONS OF THE OECD LOFT EXPERIMENT LP-SB-1. ALLEN,E.J. United Kingdom Atomic Energy Authority. April 1990. 128pp. 9007120212. AEEW-R 2254. 54466:163.

Analysis of the small, hot leg break, OECD LOFT Experiment LP-SB-1, using the "best-estimate" computer code TRAC-PF1/MOD1 is presented. Descriptions of the LOFT facility and the LP-SB-1 experiment are given and development of the TRAC-PF1/MOD1 input model is detailed. The calculations performed in achieving the steady state conditions, from which the experiment was initiated, and the specification of experimental boundary conditions are outlined.

NUREG/IA-0012: RELAP5/MOD2 CALCULATIONS OF OECD-LOFT TEST LP-SB-01. HALL,P.C.; BROWN,G. Central Electricity Generating Board. January 1990. 40pp. 9002120283. GD/PE-N/544. 52597:215.

To assist CEGB in assessing the capabilities and status of RELAP5/MOD2, the code has been used to simulate SBLOCA test LP-SB-01 carried out in the LOFT experimental reactor under the OECD LOFT programme. This test simulated a 1.0% hot leg break in a PWR, with early tripping of the primary coolant circulating pumps. This report compares the results of the RELAP5/MOD2 analysis with experimental measurements.

NUREG/IA-0013: RELAP5/MOD2 CALCULATIONS OF OECD-LOFT TEST LP-SB-03. HARWOOD,C.; BROWN,G. Central Electricity Generating Board. January 1990. 48pp. 9002120316. GD/PE-N/535. 52611:083.

This report compares the results of the RELAP5/MOD2 analysis with experimental measurements. A simulation of test LP-SB-03 was previously carried out at GDCD using the RELAP5/MODI code. RELAP5/MOD2 was developed from RELAP5/MODI and contains more sophisticated hydraulic models and constitutive relationships. Comparison of the RELAP5/MOD2 and MOD1 calculations show that RELAP5/MOD2 perfoms better than RELAP5/MODI in a number of key areas; notably mass errors are much reduced, there is improved numerical stability, and improved separator modelling and modelling of accumulator injection.

NUREG/IA-0018: RELAP5/MOD2 ASSESSMENT,OECD-LOFT SMALL BREAK EXPERIMENT LP-SB-03. GUNTAY,S. Paul Scherrer Institute. April 1990. 100pp. 9006080163. 54084:136.

An analysis of the experimental results and post-test calculations using RELAP5/MOD2 carried out for OECD-LOFT small break experiment LP-SB-3 are presented. Experiment LP-SB-3 was conducted on March 5, 1984 in the Loss-of-Fluid Test (LOFT) facility located at the Idaho National Engineering Laboratory (INEL). The experiment simulated a small cold leg break, with concurrent loss of high pressure injection system, and cooldown and recovery by feed and bleed of the steam generator secondary side and accumulator injection respectively. This report documents a short post-test analysis of the experiment emphasizing the results of additional analysis performed during the course of this task. RELAP5/MOD2 input model and results of the post-test calculation are documented. Included in the report is the results of a sensitivity analysis which show the predicted thermal-hydraulic response to a different input model.

NUREG/IA-0019: TRAC-PF1/MOD1 POST-TEST CALCULATIONS OF THE OECD LOFT EXPERIMENT LP-SB-2. PELAYO,F. United Kingdom Atomic Energy Authority. December 1990. 165pp. 9012280144. AEEW-R-2202. 56196:315.

An analysis of the OECD-LOFT-LP-SB-2 experiment making use of TRAC-PF1/MOD1 is described in the report. LP-SB-2 experiment studies the effect of a delayed pump trip in a small break LOCA scenario with a 3-inch equivalent diameter break in the hot leg of a commercial PWR operating at full power. The experiment was performed on July 14, 1983, in the LOFT facility at the Idaho National Engineering Laboratory under the auspices of the Organization for Economic Cooperation and Development (OECD). This analysis presents an evaluation of the code capability in reproducing the complex phenomena which determined the LP-SB-2 transient evolution. The analysis comprises the results obtained from two different runs. The first run is described in detail analyzing the main variables over two time spans: short and longer term. Several conclusions are drawn and then a second run testing some of these conclusions is shown.

NUREG/IA-0021: RELAP5/MOD2 CALCULATIONS OF OECD LOFT TEST LP-SB-2. HALL,P.C. Central Electricity Generating Board. April 1990. 43pp. 9005040095. GD/PE-N/606. 53622:176.

To help in assessing the capabilities of RELAP5/MOD2 for PWR Fault Analysis, the code is being used by CEGB to simulate several small LOCA and pressurized transient experiments in the LOFT experimental reactor. The present report describes an analysis of small LOCA test LP-SB-02, which simulated a 1% hot leg break LOCA in a PWR, with delayed tripping of the primary coolant pumps. This test was carried out under the OECD LOFT Program.

NUREG/IA-0022: TRAC-PF1/MOD1 POST-TEST CALCULATIONS OF THE OECD LOFT EXPERIMENT LP-SB-3. ALLEN,E.J.; NEILL,A.P. United Kingdom Atomic Energy Authority. April 1990. 100pp. 9006290072. AEEW-R 2275. 54319:001.

Analysis of the small, cold leg break, OECD LOFT Experiment LP-SB-3 using the best-estimate computer code TRAC-PF1/MOD1 is presented. Descriptions of the LOFT facility and the LP-SB-3 experiment are given and development of the TRAC-PF1/MOD1 input model is detailed. The calculations performed in achieving the steady state conditions, from which the experiment was initiated, and the specification of experimental boundary conditions are outlined.

NUREG/IA-0023 V01: ASSESSMENT OF TRAC-PF1/MOD1 VERSION 14.3 USING SEPARATE EFFECTS CRITICAL FLOW AND BLOWDOWN EXPERIMENTS.Volume 1:Text And Tables. SPINDLER,B.; PELLISSIER,M. France, Govt. of. January 1990. 141pp. 9006110015. SETH/LEML88-138. 54084:236.

Independent assessment of the TRAC code was conducted at the Centre d'Etudes Nucleaires de Grenoble of the Commissariate a l'Energie Atomique (France) in the frame of the ICAP. This report presents the results of the assessment of TRAC-PF1/MOD1 version 14.3 using critical flow steady state tests (MOBY-DICK, SUPER-MOBY-DICK), and blowdown tests (CANNON, SUPER-CANNON, VERTICAL-CANON, MARVIKEN, OMEGA-TUBE, OMEGA-BUNDLE).

NUREG/IA-0023 V02: ASSESSMENT OF TRAC-PF1/MOD1 VERSION 14.3 USING SEPARATE EFFECTS CRITICAL FLOW AND BLOWDOWN EXPERIMENTS.Volume 2: Figures. SPINDLER,B.; PELLISSIER,M. France, Govt. of. January 1990. 238pp. 9006110013. SETH/LEML88-138. 54083:264.

See NUREG/IA-0023,V01 abstract.

NUREG/IA-0030: ASSESSMENT OF RELAP5/MOD2 CODE USING LOSS OF OFFSITE POWER TRANSIENT DATA OF KNU #1 PLANT. CHUNG,B-D.; KIM,H-J. Korea Advanced Energy Research Institute. LEE,Y-J. Seoul National Univ., Seoul, Republic of Korea. April 1990. 100pp. 9005040134. 53620:288.

This report presents a code assessment study, based on a real plant transient that occurred on June 9, 1981, at the KNU #1 (Korea Nuclear Unit Number 1). KNU #1 is a two-loop Westinghouse PWR plant of 587 Mwe. The loss of offsite power transient occurred at the 77.5% reactor power with 0.5%/hr power ramp. The real plant data were collected from available on-line plant records and computer diagnostics. The transient was simulated by RELAP5/MOD2/36.05 and the results were compared with the plant data to assess the code weaknesses and strengths. Some nodalization studies were performed to contribute to developing a guideline for PWR nodalization for the transient analysis.

NUREG/IA-0031: ICAP ASSESSMENT OF RELAP5/MOD2, CYCLE 36.05 AGAINST LOFT SMALL BREAK EXPERIMENT L3-7. LEE,E.-J.; CHUNG,B.-D.; KIM,H.-J. Korea Advanced Energy Research Institute. April 1990. 171pp. 9005040112. 53621:028.

The LOFT small break (1 in-dia) experiment L3-7 has been analyzed using the reactor thermal hydraulic analysis code RELAP5/ MOD2, Cycle 36.05. The base calculation (Case A) was completed and compared with the experimental data. Three types of sensitivity studies (Cases B, C, and D) were carried out to investigate the effects of (1) break discharge coefficient Cd, (2) pump two-phase difference multiplier, and (3) high pressure injection system (HPIS) capacity on major thermal and hydraulic (T/H) parameters. A nodalization study (Case E) was conducted to assess the phenomena with a simplified nodalization.

NUREG/IA-0032: ASSESSMENT OF RELAP5/MOD2 CYCLE 36.04 USING LOFT LARGE BREAK EXPERIMENT L2-5. BANG,Y.S.; LEE,S.Y.; KIM,H-J. Korea Advanced Energy Research Institute. April 1990. 183pp. 9005040105. 53621:199.

The LOFT L2-5 LBLOCA experiment was simulated using the RELAP5/ MOD2 Cycle 36.04 code to assess its capability to predict the phenomena in LBLOCA. One base case calculation and three cases of different nodalizations were carried out. The effect of different nodalization was studied in the area of the downcomer and core. For a sensitivity study, another calculation was executed using an updated version of RELAP5/MOD2 Cycle 36.04. A split downcomer with one crossflow junction and two core channels were found to be effective in describing the ECC bypass and hot channel behavior. And the updated version was found to be effective in overcoming the code deficiency in the interfacial friction and reflood quenching.

NUREG/IA-0033: ASSESSMENT OF RELAP5/MOD2 CYCLE 36.04 AGAINST LOFT SMALL BREAK EXPERIMENT L3-6. ERIKSSON,J. Sweden, Govt. of. July 1990. 103pp. 9008140510. STUDSVIKNP87128. 54926:326.

The LOFT small break experiment L3-6 has been analyzed as part of Sweden's contribution to the International Thermal-Hydraulic Code Assessment and Applications Program (ICAP). Three calculations, of which two were sensitivity studies, were carried out. The following quantities were varied: (1) the content of secondary side fluid and the feed water valve closure, and (2) the two-phase characteristics of the main pumps. All three predictions agreed reasonably well with most of the measured data. The sensitivity calculations resulted only in marginal improvements. The predicted and measured data are compared on plots and their differences are quantified over intervals in real time.

NUREG/IA-0034: ASSESSMENT STUDY OF RELAP5/MOD2 CYCLE 36.04 BASED ON PRESSURIZER SAFETY AND RELIEF VALVE TESTS. STUBBE,E.J.; VANHOENACKER,L. TRACTEBEL. July 1990. 86pp. 9008080178. 54871:157.

This report presents a code assessment study based on full size relief and assisted safety valve (called SEBIM) tests performed on the CUMULUS valve test rig operated by Electricite de France (EDF). The increased awareness that the pressurizer safety and relief valves are not reliable under water blowdown conditions, has led to the design, testing and installation of so called assisted safety valves of which the SEBIM (TM) valves are an example. These valves, used in tandem, are gradually replacing the safety and relief valves on pressurizers in some European PWR's. Before installation at the plant, the Belgian safety authorities requested a thorough full scale testing of these valves on a test rig (CUMULUS) equipped with sufficient diagnostics to measure the characteristics of the valve. The Belgian architect-engineering firm TRACTEBEL was called upon to specify, order and test these valves for installation at the DOEL 1 and DOEL 2 power plants. These tests do not provide sufficient data of high quality to justify an assessment study of the code RELAP-5 MOD-2 CYCLE 36 in the ICAP framework which is the subject of this report.

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Secondary Report Number Index

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

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NUREG/CR-4551 V6R1P2: EVALUATION OF SEVERE ACCIDENT RISKS: GRAND GULF,UNIT 1. Appendices.

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NUREG/CR-4551 V3R1P1: EVALUATION OF SEVERE ACCIDENT RISKS: SURRY UNIT 1.Main Report.
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SMITH,R.E.
NUREG/CR-5475: MODEL FEASIBILITY STUDY OF RADIOACTIVE PATHWAYS FROM ATMOSPHERE TO SURFACE WATER.

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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.

NUREG/CR-5583: PREDICTION OF CHECK VALVE PERFORMANCE AND DEGRADATION IN NUCLEAR POWER PLANT SYSTEMS - WEAR AND IMPACT TESTS.Final Report,September 1988 - April 1990.

Aging And Maintenance
NUREG/CR-5510: EVALUATIONS OF CORE MELT FREQUENCY EFFECTS DUE TO COMPONENT AGING AND MAINTENANCE.

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Aging Stressor
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NUREG/CR-5519 V01: AGING OF CONTROL AND SERVICE AIR COMPRESSORS AND DRYERS USED IN NUCLEAR POWER PLANTS.

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NUREG-1363 V02: ATOMIC SAFETY AND LICENSING BOARD PANEL ANNUAL REPORT FISCAL YEAR 1989.

Anticipated Transient Without Scram
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Apache Leap Tuff
NUREG/CR-5596: UNSATURATED FRACTURED ROCK CHARACTERIZATION METHODS AND DATA SETS AT THE APACHE LEAP TUFF SITE.

Atlantic Seaboard
NUREG/CR-5613: PALEOLIQUEFACTION FEATURES ALONG THE ATLANTIC SEABOARD.

Atmospheric Release
NUREG/CR-2850 V09: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1987.

Atomic Safety And Licensing Board Panel
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NUREG/CR-5616: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE DIABLO CANYON UNIT 1 NUCLEAR POWER PLANT.

Auxiliary Feedwater Pump
NUREG/CR-5298: CLOSEOUT OF IE BULLETIN 85-01: STEAM BINDING OF AUXILIARY FEEDWATER PUMPS.

Auxiliary Feedwater System
NUREG/CR-5617: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE J.M. FARLEY NUCLEAR POWER PLANT.

BWR
NUREG-1372: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC ISSUE C-8, "MAIN STEAM ISOLATION VALVE LEAKAGE AND LCS FAILURE."
NUREG/CR-0672 ADD04: TECHNOLOGY, SAFETY AND COSTS OF DECOMMISSIONING A REFERENCE BOILING WATER REACTOR POWER STATION.Comparison Of Two Decommissioning Cost Estimates Developed For The Same Commercial Nuclear Reactor Power Station.
NUREG/CR-4671: THE DF-4 FUEL DAMAGE EXPERIMENT IN ACRR WITH A BWR CONTROL BLADE AND CHANNEL BOX.
NUREG/CR-5397: VALUE-IMPACT ANALYSIS OF REGULATORY OPTIONS FOR RESOLUTION OF GENERIC ISSUE C-8:MSIV LEAKAGE AND LCS FAILURE.
NUREG/CR-5421: LAPUR USER'S GUIDE.
NUREG/CR-5449: DETERMINATION OF THE NEUTRON AND GAMMA FLUX DISTRIBUTION IN THE PRESSURE VESSEL AND CAVITY OF A BOILILNG WATER REACTOR.
NUREG/CR-5528: AN ASSESSMENT OF BWR MARK II CONTAINMENT CHALLENGES,FAILURE MODES, AND POTENTIAL IMPROVEMENTS IN PERFORMANCE.
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NUREG/CR-5605: LAPUR BENCHMARK AGAINST IN-PHASE AND OUT-OF-PHASE STABILITY TESTS.
NUREG/CR-5653: RECRITICALITY IN A BWR FOLLOWING A CORE DAMAGE EVENT.

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Boric Acid Corrosion
NUREG/CR-5576: SURVEY OF BORIC ACID CORROSION OF CARBON STEEL COMPONENTS IN NUCLEAR PLANTS.

Boron Flushing
NUREG/CR-5573: BORON FLUSHING DURING A BWR ANTICIPATED TRANSIENT WITHOUT SCRAM.

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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.

ADVISORY COMMITTEE(S)

ADVISORY COMMITTEE ON NUCLEAR WASTE
- NUREG-1393: THE INCINERATION OF LOW-LEVEL RADIOACTIVE WASTE.A Report For The Advisory Committee On Nuclear Waste.
- NUREG-1423 V01: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON NUCLEAR WASTE.July 1988 - June 1990.

ACRS - ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
- NUREG-1125 V11: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS.1989 Annual.

ATOMIC SAFETY BOARD(S) & PANEL(S)

ATOMIC SAFETY AND LICENSING BOARD PANEL
- NUREG-1363 V02: ATOMIC SAFETY AND LICENSING BOARD PANEL ANNUAL REPORT FISCAL YEAR 1989.

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- NUREG-1410: LOSS OF VITAL AC POWER AND THE RESIDUAL HEAT REMOVAL SYSTEM DURING MID-LOOP OPERATIONS AT VOGTLE UNIT 1 ON MARCH 20, 1990.
- NUREG-1425: WELDING AND NONDESTRUCTIVE EXAMINATION ISSUES AT SEABROOK NUCLEAR STATION.An Independent Review Team Report.

REGION 1 (POST 820201)
- NUREG-0837 V09 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. October-December 1989.
- NUREG-0837 V10 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. January-March 1990.
- NUREG-0837 V10 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. April-June 1990.
- NUREG-0837 V10 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. July-September 1990.

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- NUREG-1395 DRFT: INDUSTRY PERCEPTIONS OF THE IMPACT OF THE U.S. NUCLEAR REGULATORY COMMISSION ON NUCLEAR POWER PLANT ACTIVITIES. Draft Report.

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- NUREG-0940 V08 N04: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,October-December 1989.
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OFC OF PERSONNEL (POST 870413)
- NUREG-0325 R13: U.S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS.April 1, 1990.
- NUREG-0325 R14: U.S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS.August 15, [illegible]

EDO - OFFICE OF ADMINISTRATION (PRE 870413 & P[illegible])

OFFICE OF ADMINISTRATION (POST 890205)
- NUREG-1145 V06: U.S. NUCLEAR REGULATORY COMMISSION 1989 ANNUAL REPORT.

DIVISION OF FREEDOM OF INFORMATION & PUBLICATIONS SERVICES (POST 890205
- NUREG-0304 V14 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1989.
- NUREG-0304 V15 N01: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1990,January-March.
- NUREG-0304 V15 N02: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Second Quarter 1990,April-June.
- NUREG-0304 V15 N03: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Third Quarter 1990,July-September.
- NUREG-0540 V11 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. October 1-31,1989.
- NUREG-0540 V11 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.NOVEMBER 1-30, 1989.
- NUREG-0540 V11 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.DECEMBER 1-31,1989.
- NUREG-0540 V12 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31,1990.
- NUREG-0540 V12 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.February 1-28, 1990.
- NUREG-0540 V12 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.March 1-31, 1990.
- NUREG-0540 V12 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.April 1-30, 1990.
- NUREG-0540 V12 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.May 1-31, 1990.
- NUREG-0540 V12 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. June 1-30, 1990.
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- NUREG-0540 V12 N08: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. August 1-31,1990.
- NUREG-0540 V12 N09: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.September 1-30,1990.
- NUREG-0540 V12 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. October 1-31, 1990.
- NUREG-0650 R01: PUBLISHING DOCUMENTS IN THE NUREG SERIES.
- NUREG-0750 CI02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January 1,1980 Through December 31,1985.
- NUREG-0750 V30 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-September 1989.
- NUREG-0750 V30 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-December 1989.
- NUREG-0750 V30 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1989.Pages 1-84.
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- NUREG-0750 V30 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1989. Pages 167-229.
- NUREG-0750 V30 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1989. Pages 231-323.
- NUREG-0750 V30 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1989. Pages 325-708.
- NUREG-0750 V30 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1989. Pages 709-811.
- NUREG-0750 V31 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January-March 1990.
- NUREG-0750 V31 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January-June 1990.
- NUREG-0750 V31 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1990. Pages 1-130.
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- NUREG-0750 V31 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1990.Pages 197-332.
- NUREG-0750 V31 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1990.Pages 333-370.
- NUREG-0750 V31 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1990.Pages 371-481.
- NUREG-0750 V31 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1990.Pages 483-604.
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- NUREG-0750 V32 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1990.Pages 1-55.
- NUREG-0750 V32 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1990. Pages 56-128.

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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.

EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)

OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 860720)

NUREG/IA-0011: TRAC-PF1/MOD1 POST-TEST CALCULATIONS OF THE OECD LOFT EXPERIMENT LP-SB-1.

NUREG/IA-0012: RELAP5/MOD2 CALCULATIONS OF OECD-LOFT TEST LP-SB-01.

NUREG/IA-0013: RELAP5/MOD2 CALCULATIONS OF OECD-LOFT TEST LP-SB-03.

NUREG/IA-0018: RELAP5/MOD2 ASSESSMENT,OECD-LOFT SMALL BREAK EXPERIMENT LP-SB-03.

NUREG/IA-0019: TRAC-PF1/MOD1 POST-TEST CALCULATIONS OF THE OECD LOFT EXPERIMENT LP-SB-2.

NUREG/IA-0021: RELAP5/MOD2 CALCULATIONS OF OECD LOFT TEST LP-SB-2.

NUREG/IA-0022: TRAC-PF1/MOD1 POST-TEST CALCULATIONS OF THE OECD LOFT EXPERIMENT LP-SB-3.

NUREG/IA-0023 V01: ASSESSMENT OF TRAC-PF1/MOD1 VERSION 14.3 USING SEPARATE EFFECTS CRITICAL FLOW AND BLOW-DOWN EXPERIMENTS.Volume 1:Text And Tables.

NUREG/IA-0023 V02: ASSESSMENT OF TRAC-PF1/MOD1 VERSION 14.3 USING SEPARATE EFFECTS CRITICAL FLOW AND BLOW-DOWN EXPERIMENTS.Volume 2: Figures.

NUREG/IA-0030: ASSESSMENT OF RELAP5/MOD2 CODE USING LOSS OF OFFSITE POWER TRANSIENT DATA OF KNU #1 PLANT.

NUREG/IA-0031: ICAP ASSESSMENT OF RELAP5/MOD2, CYCLE 36.05 AGAINST LOFT SMALL BREAK EXPERIMENT L3-7.

NUREG/IA-0032: ASSESSMENT OF RELAP5/MOD2 CYCLE 36.04 USING LOFT LARGE BREAK EXPERIMENT L2-5.

NUREG/IA-0033: ASSESSMENT OF RELAP5/MOD2 CYCLE 36.04 AGAINST LOFT SMALL BREAK EXPERIMENT L3-6.

DIVISION OF SYSTEMS RESEARCH (POST 880717)

NUREG/IA-0034: ASSESSMENT STUDY OF RELAP5/MOD2 CYCLE 36.04 BASED ON PRESSURIZER SAFETY AND RELIEF VALVE TESTS.

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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 V08N12: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of December 1989.

NUREG/CR-2000 V09 N1: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of January 1990.

NUREG/CR-2000 V09 N2: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of February 1990.

NUREG/CR-2000 V09 N3: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of March 1990.

NUREG/CR-2000 V09 N4: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of April 1990.

NUREG/CR-2000 V09 N5: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of May 1990.

NUREG/CR-2000 V09 N6: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of June 1990.

NUREG/CR-2000 V09 N7: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of July 1990.

NUREG/CR-2000 V09 N8: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of August 1990.

NUREG/CR-2000 V09 N9: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of September 1990.

NUREG/CR-2000 V09N10: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of October 1990.

NUREG/CR-2000 V09N11: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of November 1990.

DIVISION OF SAFETY PROGRAMS (POST 870413)

NUREG/CR-4674 V09: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1988 A STATUS REPORT.Main Report And Appendix A.

NUREG/CR-4674 V10: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1988 A STATUS REPORT.Appendixes B And C.

NUREG/CR-4674 V11: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1989 A STATUS REPORT.Main Report And Appendix A.

NUREG/CR-4674 V12: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1989 A STATUS REPORT.Appendices B And C.

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

DIVISION OF COMPUTER & TELECOMMUNICATIONS SERVICES (POST 890205)

NUREG/CR-2850 V09: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1987.

EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS

DIVISION OF SAFEGUARDS & TRANSPORTATION (POST 870413)

NUREG/CR-5366: HTAS2: A THREE-DIMENSIONAL TRANSIENT SHIPPING CASK ANALYSIS TOOL.

NUREG/CR-5399: SURVEY OF STATE AND TRIBAL EMERGENCY RESPONSE CAPABILITIES FOR RADIOLOGICAL TRANSPORTATION INCIDENTS.

NUREG/CR-5468: QADS: A MULTIDIMENSIONAL POINT KERNEL ANALYSIS MODULE.

DIVISION OF INDUSTRIAL & MEDICAL NUCLEAR SAFETY (POST 870729)

NUREG/CR-5566: EVALUATION OF HEALTH EFFECTS IN SEQUOYAH FUELS CORPORATION WORKERS FROM ACCIDENTAL EXPOSURE TO URANIUM HEXAFLUORIDE.

NUREG/CR-5574: DETERMINATION OF THE CHEMICAL FORM OF TRITIUM IN SELF-LUMINOUS SIGNS.

DIVISION OF HIGH-LEVEL WASTE MANAGEMENT (POST 870413)

NUREG/CR-1667: RISK METHODOLOGY FOR GEOLOGIC DISPOSAL OF RADIOACTIVE WASTE. Scenario Selection Procedure.

NUREG/CR-4735 V06: EVALUATION AND COMPILATION OF DOE WASTE PACKAGE TEST DATA. Biannual Report: August 1988 - January 1989.

NUREG/CR-5211: UNCERTAINTIES ASSOCIATED WITH PERFORMANCE ASSESSMENT OF HIGH-LEVEL RADIOACTIVE WASTE REPOSITORIES.A Summary Report.

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