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Annual Compilation for 1994

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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
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NRC Originating Organization Index (Staff Reports)
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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), (9) the microfiche address (for NRC internal use).

Grant Report

NUREG/GR-0013: APPLICATIONS OF A NEW MAGNETIC MONITORING TECHNIQUE TO IN SITU EVALUATION OF FATIGUE DAMAGE IN FERROUS COMPONENTS. JILES, D.C.; BINER, S.B.; GOVINDARAJU, M.; et al. Iowa State Univ., Ames, IA. June 1994. 41 pp. 9407250286. 80328:195.

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), (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:

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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 or the work being reported.

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

All these report codes are controlled and assigned by the staff of the Technical Publications Section of the NRC Division of Freedom of Information and 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 V18: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of December 31, 1993. (Gray Book I) HARTFIELD, R.A. Office of Information Resources Management (Post 890205). March 1994. 350pp. 9404110367. 78819:014.

The Nuclear Regulatory Commission's annual summary of licensed nuclear power reactor data is based primarily on the report of operating data submitted by licensees for each unit for the month of December because that report contains data for the month of December, the year to date (in this case calendar year 1993) and cumulative data, usually from the date of commercial operation. The data is not independently verified, but various computer checks are made. The report is divided into two sections. The first contains summary highlights and the second contains data on each individual unit in commercial operation. Section 1 capacity and availability factors are simple arithmetic averages. Section 2 items in the cumulative column are generally as reported by the licensee and notes as to the use of weighted averages and starting dates other than commercial operation are provided.

NUREG-0040 V17 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, October-December 1993. (White Book) * Division of Reactor Inspection & Licensee Performance (Post 921004). February 1994. 68pp. 9403150242. 78501:288.

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

NUREG-0040 V18 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, January-March 1994. (White Book) * Division of Reactor Inspection & Licensee Performance (Post 921004). April 1994. 256pp. 9405310289. 79587:032.

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 January through March 1994.

NUREG-0040 V18 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, April-June 1994. (White Book) * Division of Reactor Inspection & Licensee Performance (Post 921004). October 1994. 183pp. 9411290139. 81878:220.

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 April through June 1994.

NUREG-0040 V18 N03: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, July-September 1994. (White Book) * Office of Nuclear Reactor Regulation, Director (Post 870411). November 1994. 123pp. 9412300175. 82158:001.

This periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Sec-

tion, that have been distributed to the inspected organizations during the period from July through September 1994.

NUREG-0090 V16 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. July-September 1993. * Office for Analysis & Evaluation of Operational Data, Director. March 1994. 31pp. 9405050338. 79162:316.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event that the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report covers the period from July 1 through September 30, 1993. This report discusses two abnormal occurrences at NRC-licensed facilities. One involved a medical sodium iodide misadministration and one involved a 1981 fatal radiation exposure of a radiographer. One industrial radiographer overexposure event and four medical misadministrations that were reported by the Agreement States are also discussed, based on information provided by the Agreement States as of November 1, 1993. The report also contains information updating four previously reported abnormal occurrences at NRC-licensed facilities and three reported by the Agreement States, and includes information on two other events of interest. Appendix D has been added to this report which includes events submitted by Agreement States that are likely to be categorized as abnormal occurrences. For these events, insufficient information was available as of November 1, 1993, to positively identify them as abnormal occurrences.

NUREG-0090 V16 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. October-December 1993. * Office for Analysis & Evaluation of Operational Data, Director. April 1994. 43pp. 9405310231. 79589:191.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event that the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report covers the period from October 1 through December 31, 1993. This report discusses six abnormal occurrences at NRC-licensed facilities. Five involved medical brachytherapy misadministrations, and one involved an overexposure to a nursing infant. Seven abnormal occurrences that were reported by the Agreement States are also discussed, based on information provided by the Agreement States as of February 28, 1994. Of these events, three involved brachytherapy misadministrations, one involved a teletherapy misadministration, one involved a theft of radioactive material during transport and improper disposal, and two involved lost sources.

NUREG-0090 V17 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. January-March 1994. * Office for Analysis & Evaluation of Operational Data, Director. August 1994. 33pp. 9409090134. 80817:073.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence (AO) as an unscheduled incident or event that the Nuclear Regulatory Commission determines to

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be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report provides a description of those events that have been determined to be abnormal occurrences during the period of January 1 through March 31, 1994. This report addresses seven AOs at NRC-licensed facilities. One involved inoperable main steam isolation valves at a boiling water reactor, four involved medical brachytherapy misadministrations, one involved a medical teletherapy misadministration, and one involved four lost reference sources. One AO that was reported by an Agreement State is also discussed; the information is current as of April 25, 1994. This event involved a therapeutic radiopharmaceutical misadministration. The report also contains updates on seven abnormal occurrences previously reported by NRC licensees and one abnormal occurrence previously reported by an Agreement State licensee. For the period January 1 to March 31, 1994, no new "Other Events of Interest" were reported but an update to a therapeutic misadministration previously reported as an "Other Event of Interest" is included.

NUREG-0090 V17 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. April-June 1994. * Office for Analysis & Evaluation of Operational Data, Director. October 1994. 41pp. 9412130086. 82006:305.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence (AO) as an unscheduled incident or event that the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report provides a description of those events that have been determined to be abnormal occurrences during the period of April 1 through June 30, 1994. This report addresses seven abnormal occurrences (AOs) at NRC-licensed facilities. Five involved medical brachytherapy misadministrations, one involved a medical teletherapy misadministration, and one involved a medical sodium iodide misadministration. Four AOs were reported by the Agreement States as of August 3, 1994. Two involved medical brachytherapy misadministrations, one involved a radiation burn received by an industrial radiographer, and one involved a lost well logging source. The report also contains updates of seven AOs previously reported by NRC licensees and five AOs previously reported by Agreement State licensees. Three "Other Events of Interest" are also reported. One involved a deliberate coverup of an error in a diagnostic radiopharmaceutical administration at an NRC licensee, one involved an Order Suspending License and Demand for Information at an NRC licensee, and one involved an overexposure of an industrial radiographer at an Agreement State licensee.

NUREG-0304 V18 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1993. * Division of Freedom of Information & Publications Services (890206-940714). March 1994. 125pp. 9405050332. 79208:147.

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

NUREG-0304 V19 N01: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1994, January-March. * Division of Freedom of Information & Publications Services (890206-940714). June 1994. 48pp. 9407250316. 80328:300.

See NUREG-0304,V18,N04 abstract.

NUREG-0304 V19 N02: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Second Quarter 1994, April-June. * Division of Freedom of Information & Publications Services (Post 940714). September 1994. 46pp. 9410130043. 81264:001.

See NUREG-0304,V18,N04 abstract.

NUREG-0325 R17: U.S. NUCLEAR REGULATORY COMMISSION ORGANIZATION CHARTS AND FUNCTIONAL STATEMENTS. October 3, 1994. * Ofc of Personnel (Post 870413). October 1994. 82pp. 9411160041. 81756:001.

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

NUREG-0383 V01 R17: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Report Of NRC Approved Packages. * Division of Industrial & Medical Nuclear Safety (Post 870729). October 1994. 513pp. 9411290132. 81875:001.

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 R17: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Certificates Of Compliance. * Division of Industrial & Medical Nuclear Safety (Post 870729). October 1994. 583pp. 9411290134. 81877:001.

See NUREG-0383,V01,R17 abstract.

NUREG-0383 V03 R14: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Report Of NRC Approved Quality Assurance Programs For Radioactive Materials Packages. * Division of Industrial & Medical Nuclear Safety (Post 870729). October 1994. 151pp. 9411290135. 81876:150.

See NUREG-0383,V01,R17 abstract.

NUREG-0390 V08 N01: TOPICAL REPORT REVIEW STATUS. (Blue Book) * Program Management, Policy Development & Analysis Staff (Post 870411). May 1994. 30pp. 9406200334. 79831:298.

This report provides industry with procedures for submitting topical reports, guidance on how the U.S. Nuclear Regulatory Commission (NRC) will process and respond to topical report submissions, and an accounting of all topical reports currently being reviewed by the NRC staff. This report will be published semiannually.

NUREG-0430 V13: LICENSED FUEL FACILITY STATUS REPORT. Inventory Difference Data. July 1, 1992 - June 30, 1993. (Gray Book II) JOY,D.R. Office of Nuclear Material Safety & Safeguards. February 1994. 19pp. 9403150260. 78501:269.

NRC is committed to the periodic publication of licensed fuel facility 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 special nuclear material.

NUREG-0498 S01 DFC: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT UNITS 1 AND 2. Draft Report For Comment. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority) * Associate Director for Advanced Reactors & License Renewal (ADAR) (Post 941001). November 1994. 185pp. 9412130097. 82007:001.

The Final Environmental Statement (FES) issued in 1978 represents the Nuclear Regulatory Commission's (NRC's) previous environmental review related to the operation of Watts Bar Nuclear Plant (WBN). The purpose of this NRC review is to discuss the effects of observed changes in environment and to evaluate the changes in environmental impacts that have occurred as a result of changes in the WBN Plant design and proposed methods of operations since the last environmental review. A full scope of environmental topics has been evaluated, including regional demography, land and water use, meteorology, terrestrial and aquatic ecology, radiological and non-radiological impacts on humans and the environment, socioeconomic impacts, and environmental justice. The staff concluded that there are no significant changes in the environmental impacts since the NRC 1978 FES-OL from changes in plant design, proposed methods of operation, or changes in the environment. The applicant's preoperational and operational monitoring programs were reviewed and found to be appropriate for establishing baseline conditions and ongoing assessments of environmental impacts. The staff also conducted an analysis of plant operation with severe accident mitigation design alternatives (SAMDAs) and concluded that none of the SAMDAs, beyond the three procedural changes that the applicant committed to implement, would be cost-beneficial for further mitigating environmental impacts.

NUREG-0525 V02 R02: SAFEGUARDS SUMMARY EVENT LIST (SSEL). January 1, 1990 Through December 31, 1993. FADDEN, M.; YARDUMIAN, J. Operations Branch. July 1994. 124pp. 9410130051. 81264:047.

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

NUREG-0540 V15 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30, 1993. * Division of Freedom of Information & Publications Services (890206-940714). January 1994. 251pp. 9402150231. 78121:163.

This document is a monthly publication containing descriptions of information received and generated by the U.S. Nuclear Regulatory Commission (NRC). This information includes (1) docketed material associated with civilian nuclear power plants and other uses of radioactive materials, and (2) nondocketed material received and generated by NRC pertinent to its role as a regulatory agency. The following indexes are included: Personal Author, Corporate Source, Report Number, and Cross Reference of Enclosures to Principal Documents.

NUREG-0540 V15 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. December 1-31, 1993. * Division of Freedom of Information & Publications Services (890206-940714). February 1994. 371pp. 9404010178. 78719:033.
See NUREG-0540, V15, N11 abstract.

NUREG-0540 V16 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31, 1994. * Division of Freedom of Information & Publications Services (890206-940714). March 1994. 288pp. 9404010174. 78718:109.
See NUREG-0540, V15, N11 abstract.

NUREG-0540 V16 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. February 1-28, 1994. * Division of Freedom of Information & Publications Services (890206-940714). April 1994. 304pp. 9405040142. 79106:001.
See NUREG-0540, V15, N11 abstract.

NUREG-0540 V16 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. March 1-31, 1994. * Division of Freedom of Information & Publications Services (890206-940714). May 1994. 400pp. 9405310285. 79586:001.
See NUREG-0540, V15, N11 abstract.

NUREG-0540 V16 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30, 1994. * Division of Freedom of Information & Publications Services (890206-940714). June 1994. 325pp. 9406290304. 80013:136.
See NUREG-0540, V15, N11 abstract.

NUREG-0540 V16 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. May 1-31, 1994. * Division of Freedom of Information & Publications Services (Post 940714). July 1994. 325pp. 9407250168. 80327:001.
See NUREG-0540, V15, N11 abstract.

NUREG-0540 V16 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. June 1-30, 1994. * Division of Freedom of Information & Publications Services (Post 940714). October 1994. 331pp. 9412090047. 81986:001.
See NUREG-0540, V15, N11 abstract.

NUREG-0540 V16 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. July 1-31, 1994. * Division of Freedom of Information & Publications Services (Post 940714). November 1994. 320pp. 9412070108. 81958:288.
See NUREG-0540, V15, N11 abstract.

NUREG-0540 V16 N08: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. August 1-31, 1994. * Division of Freedom of Information & Publications Services (Post 940714). November 1994. 391pp. 9412070105. 81957:260.
See NUREG-0540, V15, N11 abstract.

NUREG-0540 V16 N09: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. September 1-30, 1994. * Division of Freedom of Information & Publications Services (Post 940714). November 1994. 322pp. 9412300185. 82159:321.
See NUREG-0540, V15, N11 abstract.

NUREG-0540 V16 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. October 1-31, 1994. * Division of Freedom of Information & Publications Services (Post 940714). December 1994. 321pp. 9501180150. 82346:001.
See NUREG-0540, V15, N11 abstract.

NUREG-0711: HUMAN FACTORS ENGINEERING PROGRAM REVIEW MODEL. O'HARA, J.M.; HIGGINS, J.C.; STUBLER, W.F.; et al. Brookhaven National Laboratory. July 1994. 97pp. 9408220036. 80637:001.

The staff of the Nuclear Regulatory Commission is performing nuclear power plant design certification reviews based on a design process plan that describes the human factors engineering (HFE) program elements that are necessary and sufficient to develop an acceptable detailed design specification and an acceptable implemented design. There are two principal reasons for this approach. First, the initial design certification applications submitted for staff review did not include detailed design information. Second, since human performance literature and industry experiences have shown that many significant human factors issues arise early in the design process, review of the design process activities and results is important to the evaluation of an overall design. However, current regulations and guidance documents do not address the criteria for design process review. Therefore, the HFE Program Review Model (HFE PRM) was developed as a basis for performing design certification reviews that include design process evaluations as well as review

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of the final design. A central tenet of the HFE PRM is that the HFE aspects of the plant should be developed, designed, and evaluated on the basis of a structured top-down system analysis using accepted HFE principles. The HFE PRM consists of ten component elements. Each element is divided into four sections: Background, Objective, Applicant Submittals, and Review Criteria. This report describes the development of the HFE PRM and gives a detailed description of each HFE review element.

NUREG-0750 V37: NUCLEAR REGULATORY COMMISSION ISSUANCES. Opinions And Decisions Of The Nuclear Regulatory Commission With Selected Orders. January-June 1993. * Division of Freedom of Information & Publications Services (890206-940714). June 1994. 572pp. 9409010177. 80723:001.

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

NUREG-0750 V38: NUCLEAR REGULATORY COMMISSION ISSUANCES. Opinions And Decisions Of The Nuclear Regulatory Commission With Selected Orders. July-December 1993. * Division of Freedom of Information & Publications Services (890206-940714). June 1994. 436pp. 9409010179. 80724:208.
See NUREG-0750, V37 abstract.

NUREG-0750 V38 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-September 1993. * Division of Freedom of Information & Publications Services (890206-940714). January 1994. 41pp. 9403140325. 78454:291.

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

NUREG-0750 V38 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-December 1993. * Division of Freedom of Information & Publications Services (890206-940714). April 1994. 60pp. 9405040123. 79103:096.
See NUREG-0750, V38, I01 abstract.

NUREG-0750 V38 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1993. Pages 187-288. * Division of Freedom of Information & Publications Services (890206-940714). February 1994. 109pp. 9405240153. 79484:001.
See NUREG-0750, V37 abstract.

NUREG-0750 V38 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1993. Pages 289-391. * Division of Freedom of Information & Publications Services (890206-940714). February 1994. 109pp. 9403140316. 78477:148.
See NUREG-0750, V37 abstract.

NUREG-0750 V39 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January -March 1994. * Division of Freedom of Information & Publications Services (Post 940714). July 1994. 41pp. 9408150162. 80537:291.
See NUREG-0750, V38, I01 abstract.

NUREG-0750 V39 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-June 1994. * Division of Freedom of Information & Publications Services (Post 940714). September 1994. 68pp. 9410130089. 81266:182.
See NUREG-0750, V38, I01 abstract.

NUREG-0750 V39 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1994. Pages 1-45. * Division of Freedom of Information & Publications Services (890206-940714). April 1994. 52pp. 9405040127. 79103:157.
See NUREG-0750, V37 abstract.

NUREG-0750 V39 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1994. Pages 47-90. * Division of Freedom of Information & Publications Services (890206-940714). April 1994. 50pp. 9405040131. 79103:209.
See NUREG-0750, V37 abstract.

NUREG-0750 V39 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1994. Pages 91-186. * Division of Freedom of Information & Publications Services (890206-940714). May 1994. 105pp. 9406200339. 79831:001.
See NUREG-0750, V37 abstract.

NUREG-0750 V39 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1994. Pages 187-247. * Division of Freedom of Information & Publications Services (890206-940714). June 1994. 69pp. 9407010320. 80030:294.
See NUREG-0750, V37 abstract.

NUREG-0750 V39 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1994. Pages 249-284. * Division of Freedom of Information & Publications Services (Post 940714). July 1994. 42pp. 9407250180. 80327:324.
See NUREG-0750, V37 abstract.

NUREG-0750 V39 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1994. Pages 285-390. * Division of Freedom of Information & Publications Services (Post 940714). August 1994. 112pp. 9408250025. 80660:001.
See NUREG-0750, V37 abstract.

NUREG-0750 V40 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-September 1994. * Division of Freedom of Information & Publications Services (Post 940714). December 1994. 33pp. 9501130244. 82314:293.
See NUREG-0750, V38, I01 abstract.

NUREG-0750 V40 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1994. Pages 1-41. * Division of Freedom of Information & Publications Services (Post 940714). September 1994. 48pp. 9410130023. 81262:307.
See NUREG-0750, V37 abstract.

NUREG-0750 V40 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1994. Pages 43-132. * Division of Freedom of Information & Publications Services (Post 940714). October 1994. 98pp. 9411080019. 81647:133.
See NUREG-0750, V37 abstract.

NUREG-0750 V40 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1994. Pages 133-145. * Division of Freedom of Information & Publications Services (Post 940714). October 1994. 20pp. 9411160001. 81755:323.
See NUREG-0750, V37 abstract.

NUREG-0750 V40 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1994. Pages 147-167. * Division of Freedom of Information & Publications Services (Post 940714). November 1994. 27pp. 9412140189. 82032:272.
See NUREG-0750, V37 abstract.

NUREG-0837 V13 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. October-December 1993. STRUCKMEYER, R. Region 1 (Post 820201). March 1994. 328pp. 9404060261. 78758:001.

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

NUREG-0837 V14 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. January-March 1994. STRUCKMEYER, R. Region 1 (Post 820201). June 1994. 231pp. 9407250293. 80329:001.

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 first quarter of 1994.

NUREG-0837 V14 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. April-June 1994. STRUCKMEYER, R. Region 1 (Post 820201). August 1994. 231pp. 9409260062. 81042:112.

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 1994.

NUREG-0837 V14 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. July-September 1994. STRUCKMEYER, R. Region 1 (Post 820201). December 1994. 231pp. 9501180140. 82346:322.

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 1994.

NUREG-0847 S13: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority) TAM, P.S. Division of Reactor Projects - I/II (Post 870411). April 1994. 154pp. 9405310281. 79588:001.

Supplement No. 13 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 with (1) additional information submitted by the applicant since Supplement No. 12 was issued, and (2) matters that the staff had under review when Supplement No. 12 was issued.

NUREG-0847 S14: 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 (DRPE) Post 941001. December 1994. 132pp. 9412300174. 82158:127.

Supplement No. 14 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 with (1) additional information submitted by the applicant since Supplement No. 13 was issued, and (2) matters that the staff had under review when Supplement No. 13 was issued.

NUREG-0933 S17: A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT, R. Division of Safety Issue Resolution (880717-941217). September 1994. 307pp. 9411290141. 81879:039.

The report presents the safety priority ranking 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 resolution 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 V12 N04: NRC REGULATORY AGENDA. Quarterly Report, October-December 1993. * Division of Freedom of Information & Publications Services (890206-940714). February 1994. 130pp. 9403150227. 78501:135.

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

action, or is considering action, and all petitions for rulemaking which have been received by the Commission and are pending disposition by the Commission. The Regulatory Agenda is updated and issued each quarter.

NUREG-0936 V13 N01: NRC REGULATORY AGENDA. Quarterly Report, January-March 1994. * Division of Freedom of Information & Publications Services (890206-940714). June 1994. 58pp. 9407250298. 80328:235.

See NUREG-0936, V12, N04 abstract.

NUREG-0936 V13 N02: NRC REGULATORY AGENDA. Quarterly Report, April-June 1994. * Division of Freedom of Information & Publications Services (Post 940714). August 1994. 68pp. 9410060320. 81223:167. See NUREG-0936, V12, N04 abstract.

NUREG-0940 V12 N03: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED. Quarterly Progress Report, July-September 1993. * Ofc of Enforcement (Post 870413). December 1993. 601pp. 9402220162. 78193:062.

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

NUREG-0940 V12 N04: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED. Quarterly Progress Report, October-December 1993. * Ofc of Enforcement (Post 870413). March 1994. 387pp. 9404010220. 78721:266.

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

NUREG-0940 V13N01P01: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED REACTOR LICENSEES. Quarterly Progress Report, January-March 1994. * Ofc of Enforcement (Post 870413). June 1994. 210pp. 9407260127. 80345:099.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January - March 1994) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to reactor 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 V13N01P02: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MEDICAL LICENSEES. Quarterly Progress Report, January-March 1994. * Ofc of Enforcement (Post 870413). June 1994. 145pp. 9407260176. 80346:001.

See NUREG-0940, V13, N01, P01 abstract.

NUREG-0940 V13N01P03: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED INDUSTRIAL LICENSEES. Quarterly Progress Report, January-March 1994. * Ofc of Enforcement (Post 870413). June 1994. 118pp. 9407260183. 80346:146.

See NUREG-0940, V13, N01, P01 abstract.

6 Main Citations and Abstracts

NUREG-0940 V13N02P01: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED REACTOR LICENSEES. Quarterly Progress Report, April-June 1994. * Ofc of Enforcement (Post 870413). August 1994. 200pp. 9409270322. 81046:108.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1994) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to reactor 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 V13N02P02: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MEDICAL LICENSEES. Quarterly Progress Report, April-June 1994. * Ofc of Enforcement (Post 870413). August 1994. 354pp. 9410130039. 81263:001.

See NUREG-0940, V13, N02, P01 abstract.

NUREG-0940 V13N02P03: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED INDUSTRIAL LICENSEES. Quarterly Progress Report, April-June 1994. * Ofc of Enforcement (Post 870413). August 1994. 177pp. 9410070244. 81232:001.

See NUREG-0940, V13, N02, P01 abstract.

NUREG-0940 V13N03P01: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, REACTOR LICENSEES. Quarterly Progress Report, July-September 1994. * Ofc of Enforcement (Post 870413). December 1994. 315pp. 9501180207. 82344:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (July - September 1994) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to reactor 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 V13N03P02: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, MEDICAL LICENSEES. Quarterly Progress Report, July-September 1994. * Ofc of Enforcement (Post 870413). December 1994. 316pp. 9501180211. 82343:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (July - September 1994) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to medical 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 V13N03P03: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, MATERIAL LICENSEES (NON-MEDICAL). Quarterly Progress Report, July-September 1994. * Ofc of Enforcement (Post 870413). December 1994. 328pp. 9501180215. 82342:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (July - September 1994) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to material licensees (non-medical) 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 R07 S01: OPERATOR LICENSING EXAMINER STANDARDS. * Division of Reactor Controls & Human Factors (Post 921004). June 1994. 120pp. 9407070304. 80112:284.

The Operator Licensing Examiner Standards provide policy and guidance to NRC examiners and establish the procedures and practices for examining licensees and applicants for reactor operator and senior reactor operator licenses at power reactor facilities pursuant to Part 55 of Title 10 of the Code of Federal Regulations (10 CFR 55). The Examiner Standards are intended to assist NRC examiners and facility licensees to better understand the initial and requalification examination processes and to ensure the equitable and consistent administration of examinations to all applicants. These standards are not a substitute for the operator licensing regulations and are subject to revision or other internal operator licensing policy changes. Revision 7 was published in January 1993 and became effective in August 1993. Supplement 1 is being issued primarily to implement administrative changes to the requalification examination program resulting from the amendment to 10 CFR 55 that eliminated the requirement for every licensed operator to pass an NRC-conducted requalification examination as a condition for license renewal. The supplement does not substantially alter either the initial or requalification examination processes and will become effective 30 days after its publication is noticed in the Federal Register. The corporate notification letters issued after the effective date will provide facility licensees with at least 90 days notice that the examinations will be administered in accordance with the revised procedures.

NUREG-1022 R01 DR FC: EVENT REPORTING GUIDELINES 10CFR50.72 AND 50.73. Second Draft For Comment. ALLISON, D.P.; HARPER, M.R.; ISRAEL, S.; et al. Office for Analysis & Evaluation of Operational Data, Director. February 1994. 182pp. 9402220125. 78192:041.

Revision 1 to NUREG-1022 clarifies the immediate notification requirements of Title 10 of the Code of Federal Regulations, Part 50, Section 50.72 (10 CFR 50.72), and the 30-day written licensee event report (LER) requirements of 10 CFR 50.73 for nuclear power plants. This revision was initiated to improve the reporting guidelines related to 10 CFR 50.72 and 50.73 and to consolidate these guidelines into a single reference document. A first draft of this document was noticed for public comment in the Federal Register on October 7, 1991 (56 FR 50598). This document updates and supersedes NUREG-1022 and its Supplements 1 and 2 (published in September 1983, February 1984, and September 1985, respectively). It does not change the reporting requirements of 10 CFR 50.72 and 50.73.

NUREG-1100 V10: BUDGET ESTIMATES. Fiscal Year 1995. * Division of Budget & Analysis (Post 890205). February 1994. 217pp. 9402220139. 78191:075.

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

NUREG-1125 V15: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS. 1993 Annual. * ACRS - Advisory Committee on Reactor Safeguards. April 1994. 156pp. 9405040139. 79104:001.

This compilation contains 47 ACRS reports submitted to the Commission, Executive Director for Operations, or to the Office of Nuclear Regulatory Research, during calendar year 1993. 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 categorized by the most appropriate generic subject area and by chronological order within subject area.

NUREG-1145 V10: U.S. NUCLEAR REGULATORY COMMISSION 1993 ANNUAL REPORT. * Office of Administration, Director (Post 940714). September 1994. 306pp. 9410130036. 81262:001.

This report covers the major activities, events, decisions, and planning that took place during Fiscal Year 1993 within the U.S. Nuclear Regulatory Commission (NRC) or involving the NRC.

NUREG-1200 R03: STANDARD REVIEW PLAN FOR THE REVIEW OF A LICENSE APPLICATION FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY. * Division of Low-Level Waste Management & Decommissioning (870413-940402). April 1994. 582pp. 9405100233. 79279:287.

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

NUREG-1214 R13: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. ALLENSPACH, F. Division of Reactor Inspection & Licensee Performance (Post 921004). March 1994. 94pp. 9404110359. 78851:200.

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. Section 2 presents a chronological listing of all SALP report ratings for each operating facility since February 1, 1989.

NUREG-1242 V03 PT01: NRC REVIEW OF ELECTRIC POWER RESEARCH INSTITUTE'S ADVANCED LIGHT WATER REACTOR UTILITY REQUIREMENTS DOCUMENTS. Passive Plant Designs. Chapter 1. Project Number 669. * Associate Director for Advanced Reactors & License Renewal (Post 910918). August 1994. 450pp. 9410060313. 81222:001.

The staff of the U.S. Nuclear Regulatory Commission has prepared Volume 3 (Parts 1 and 2) of a safety evaluation report (SER), "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements - Passive Plant Designs," to document the results of its review of the Electric Power Research Institute's "Advanced Light Water Reactor Utility Requirements Document." This SER gives the results of the staff's review of Volume III of the Requirements Document for passive plant designs, which consists of 13 chapters and contains utility design requirements for nuclear power plants for which passive features will be used in their design (approximately 600 megawatts-electric).

NUREG-1242 V03 PT02: NRC REVIEW OF ELECTRIC POWER RESEARCH INSTITUTE'S ADVANCED LIGHT WATER REACTOR UTILITY REQUIREMENTS DOCUMENT. Passive Plant Designs. Chapters 2-13. Project Number 669. * Associate Director for Advanced Reactors & License Renewal (Post 910918). August 1994. 642pp. 9410070247. 81230:001.

See NUREG-1242, V03, P01 abstract.

NUREG-1266 V08: NRC SAFETY RESEARCH IN SUPPORT OF REGULATION - FY 1993. * Office of Nuclear Regulatory Research (860720-941217). June 1994. 100pp. 9407070294. 80113:209.

This report, the ninth 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 1993. A special emphasis on accomplishments in nuclear power plant aging research reflects recognition that a number of plants are entering the final portion of their original 40-year operating licenses and that, in addition to current aging effects a focus on safety considerations for license renewal becomes timely. The primary purpose of performing regulatory research is to develop and provide the Commission and its staff with the technical bases for regulatory decisions on the safe operation of licensed nuclear reactors and facilities, to find unknown or unexpected safety problems, and to develop data and related information for the purpose of revising the Commission's rules, regulatory guides, or other guidance.

NUREG-1272 V08 N01: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA. 1993 Annual Report - Power Reactors. * Office for Analysis & Evaluation of Operational Data, Director. December 1994. 261pp. 9501180134. 82348:001.

This annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) describes activities conducted during 1993. The report is published in two parts. NUREG-1272, Vol. 8, No. 1, covers power reactors and presents an overview of the operating experience of the nuclear power industry from the NRC perspective, including comments about the trends of some key performance measures. The report also includes the principal findings and issues identified in AEOD studies over the past year and summarizes information from such sources as licensee event reports, diagnostic evaluations, and reports to the NRC's Operations Center. NUREG-1272, Vol. 8, No. 2 covers nuclear materials and presents a review of the events and concerns during 1993 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations. Both reports also contain a discussion of the Incident Investigation Team program and summarize both the Incident Investigation Team and Augmented Inspection Team reports. Each volume contains a list of the AEOD reports issued from 1980 through 1993.

NUREG-1275 V10: OPERATING EXPERIENCE FEEDBACK REPORT - RELIABILITY OF SAFETY-RELATED STEAM TURBINE-DRIVEN STANDBY PUMPS. Commercial Power Reactors. BOARDMAN, J.R. Division of Safety Programs (Post 870413). October 1994. 91pp. 9411280281. 81851:135.

This report documents a detailed analysis of failure initiators, causes and design features for steam turbine assemblies (turbines with their related components, such as governors and valves) which are used as drivers for standby pumps in the auxiliary feedwater systems of U.S. commercial pressurized water reactor plants, and in the high pressure coolant injection and reactor core isolation cooling systems of U.S. commercial boiling water reactor plants. These standby pumps provide a redundant source of water to remove reactor core heat as specified in individual plant safety analysis reports. The period of review for this report was from January 1974 through December 1990 for licensee event reports (LERs) and January 1985 through December 1990 for Nuclear Plant Reliability Data System (NPRDS) failure data. This study confirmed the continuing validity of conclusions of earlier studies by the U.S. Nuclear Regulatory Commission and by the U.S. nuclear industry that the most significant factors in failures of turbine-driven standby pumps have been the failures of the turbine-drivers and their controls. Inadequate maintenance and the use of inappropriate vendor technical in-

formation were identified as significant factors which caused recurring failures.

NUREG-1307 R04: REPORT ON WASTE BURIAL CHARGES. Escalation Of Decommissioning Waste Disposal Costs At Low-Level Waste Burial Facilities. * Division of Regulatory Applications (870413-941217). June 1994. 54pp. 9408250008. 80658:306.

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

NUREG-1323 R00: LICENSEE APPLICATION REVIEW PLAN FOR A GEOLOGIC REPOSITORY FOR SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE. * Division of Waste Management (NMSS 940403). September 1994. 564pp. 9411290144. 81880:001.

In developing the LARP at this time, there are numerous benefits to be realized by the NRC staff. First, the LARP will help structure and coordinate the staff's efforts to develop a well-integrated review capability supported by the staff's independent technical assessment capability and the results of NRC-sponsored research. Second, identifying technical issues most important to geologic repository performance will help focus and prioritize the pre-license application reviews, independent technical assessment capabilities, and research needed by the staff. Third, the regulatory and technical basis for the staff's pre-license application reviews and guidance will be improved due to the focus on regulatory and technical issues. The LARP is divided into three parts. Part A contains the License Application Review Strategy, which gives general guidance to the staff in conducting its license application reviews. Part B contains eight individual review plans, to be used by the staff to review general information in the license application. Part C contains 89 individual review plans, distributed among ten chapters, which the staff will use to review the Safety Analysis Report, the principal part of the license application in which DOE provides the information needed to demonstrate compliance with the technical requirements of 10 CFR Part 60. The organization of the individual review plans in Parts B and C is consistent with the organization of the license application as specified in the draft regulatory guide "Format and Content for the License Application for the Waste Repository." Finally, each individual review plan has a standard format consisting of the following six sections: (1) Applicable [10 CFR Part 60] Regulatory Requirements; (2) Review Strategy; (3) Review Procedures and Acceptance Criteria; (4) Implementation; (5) Example Evaluation Findings; and (6) References.

NUREG-1350 V06: NUCLEAR REGULATORY COMMISSION INFORMATION DIGEST. 1994 Edition. STADLER, L. Division of Budget & Analysis (Post 890205). March 1994. 128pp. 9406060130. 79618:009.

The Nuclear Regulatory Commission Information Digest (digest) provides a summary of information about the U.S. Nuclear Regulatory Commission (NRC), NRC's regulatory responsibilities, the activities NRC licensees, and general information on domestic and worldwide nuclear energy. The digest, published annually, is a compilation of nuclear- and NRC-related data and is designed to provide a quick reference to major facts about the agency and the industry it regulates. In general, the data cover 1975 through 1993, with exceptions noted. Information on

generating capacity and average capacity factor for operating U.S. commercial nuclear power reactors is obtained from monthly operating reports that are submitted directly to the NRC by the licensee. This information is reviewed by the NRC for consistency only and no independent validation and/or verification is performed.

NUREG-1368: PREAPPLICATION SAFETY EVALUATION REPORT FOR THE POWER REACTOR INNOVATIVE SMALL MODULE (PRISM) LIQUID-METAL REACTOR. Final Report. DONOGHUE, J.E.; DONOHEW, J.N.; GOLUB, G.R.; et al. Associate Director for Advanced Reactors & License Renewal (Post 910918). February 1994. 400pp. 9404010163. 78716:001.

This preapplication safety evaluation report (PSER) presents the results of the preapplication design review for the Power Reactor Innovative Small Module (PRISM) liquid-metal (sodium)-cooled reactor, Nuclear Regulatory Commission (NRC) Project No. 674. The PRISM conceptual design was submitted by the U.S. Department of Energy in accordance with the NRC's "Statement of Policy for the Regulation of Advanced Nuclear Power Plants" (51 Federal Register 24643). This policy provides for the early Commission review and interaction with designers and licensees. The PRISM reactor design is a small, modular, pool-type, liquid-metal (sodium)-cooled reactor. The standard plant design consists of three identical power blocks with a total electrical output rating of 1395 MWe. Each power block comprises three reactor modules, each with a thermal rating of 471 MWt. Each module is located in its own below-grade silo and is connected to its own intermediate heat transport system and steam generator system. The reactors utilize a metallic-type fuel, a ternary alloy of U-Pu-Zr. The design includes passive reactor shutdown and passive decay heat removal features. The PSER is the NRC's preliminary evaluation of the safety features in the PRISM design, including the projected research and development programs required to support the design and the proposed testing needs. Because the NRC review was based on a conceptual design, the PSER did not result in an approval of the design. Instead it identified certain key safety issues, provided some guidance on applicable licensing criteria, assessed the adequacy of the preapplicant's research and development programs, and concluded that no obvious impediments to licensing the PRISM design had been identified.

NUREG-1415 V06 N02: OFFICE OF THE INSPECTOR GENERAL. Semiannual Report, October 1, 1993 - March 31, 1994. NORTON, L.; BARCHI, T.; HUBER, D. Office of the Inspector General (Post 890417). April 1994. 46pp. 9409200338. 80954:277.

The Inspector General is required by the IG Act of 1978, as amended, to prepare a semiannual report to Congress which summarizes program activities. The 6-month reporting period ends March 31 and September 30th. The IG's report is submitted to the Chairman of the NRC not later than April 30 and October 31, respectively. The Chairman comments on the IG's report and prepares his own, as required by the Act, and submits both reports to Congress no later than November 30 and May 31, respectively.

NUREG-1415 V07 N01: OFFICE OF THE INSPECTOR GENERAL. Semiannual Report, April 1 - September 30, 1994. DONOVAN, D.; MULLEY, G.; BODENSTEINER, J.; et al. Office of the Inspector General (Post 890417). October 1994. 50pp. 9412300183. 82159:204.

See NUREG-1415, V06, N02 abstract.

NUREG-1416: OPERATIONAL EXPERIENCE AND MAINTENANCE PROGRAMS OF TRANSAMERICA DELAVAL, INC., DIESEL GENERATORS. RAJAN, J. Division of Engineering (Post 921004). May 1994. 42pp. 9405310249. 79589:001.

In response to concerns prompted by a crankshaft failure in a Transamerica Delaval, Inc., (TDI) emergency diesel generator (EDG) at Shoreham in August 1983, 11 (now 8) U.S. nuclear utility owners formed a TDI Diesel Generator Owners' Group to

address the related operational and regulatory issues. The technical resolution involved implementation of programs focused on the design review of a large set of important engine components to ensure their adequacy from a manufacturing and operational standpoint. The most critical periodic maintenance/surveillance actions for certain components, such as connecting rods, crankshafts, cylinder blocks, cylinder heads, piston skirts, and turbochargers were incorporated as license conditions. On the basis of its review of the substantial operational experience of the TDI engines accumulated since 1985 and the inspection results of the EDG components, the NRC staff has concluded that there is adequate justification in removing the component-based license conditions, and the TDI EDGs can be regulated in a manner similar to other EDGs within the nuclear industry without the special requirements imposed in 1985.

NUREG-1426 V02: COMPILATION OF REPORTS FROM RESEARCH SUPPORTED BY THE MATERIALS ENGINEERING BRANCH, DIVISION OF ENGINEERING. 1991-1993. HISER, A.L. Division of Engineering (870413-941217). June 1994. 39pp. 9408030136. 80425:196.

Since 1965, the Materials Engineering Branch, Division of Engineering, of the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research, and its predecessors dating back to the Atomic Energy Commission (AEC), has sponsored research programs concerning the integrity of the primary system pressure boundary of light water reactors. The components of concern in these research programs have included the reactor pressure vessel (RPV), steam generators, and the piping. These research programs have covered a broad range of topics, including fracture mechanics analysis and experimental work for RPV and piping applications, inspection method development and qualification, and evaluation of irradiation effects to RPV steels. This report provides as complete a listing as practical of formal technical reports submitted to the NRC by the investigators working on these research programs. This listing includes topical, final and progress reports, and is segmented by topic area. In many cases a report will cover several topics (such as in the case of progress reports of multi-faceted programs), but is listed under only one topic. Therefore, in searching for reports on a specific topic, other related topic areas should be checked also. The previous volume to this report covers the period 1965 - 1990.

NUREG-1435 S03: STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS. TMI Action Plan Requirements. Unresolved Safety Issues. Generic Safety Issues. Other Multiplant Action Issues. * Program Management, Policy Development & Analysis Staff (Post 870411). December 1993. 165pp. 9402220173. 78197:187.

As part of ongoing U.S. Nuclear Regulatory Commission (NRC) efforts to ensure the quality and accountability of safety issue information, a program was established whereby an annual NUREG report would be published on the status of licensee implementation and NRC verification of safety issues in major NRC requirement areas. This information was compiled and reported in three NUREG volumes. Volume 1, published in March 1991, addressed the status of Three Mile Island (TMI) Action Plan Requirements. Volume 2, published in May 1991, addressed the status of unresolved safety issues (USIs). Volume 3, published in June 1991, addressed the implementation and verification status of generic safety issues (GSIs). Supplement 1, published in December 1991 combined these volumes into a single report and provided updated information as of September 30, 1991. Supplement 2, published in December 1992, provided updated information on TMI, USI, and GSI issues and included status of all other Multiplant Actions (MPAs). This annual NUREG report provides updated information on TMI, USI, and GSI and other MPAs as of September 30, 1993. The data contained in these NUREG reports are a product of the NRC's Safety Issues Management System (SIMS) database, which is maintained by the Project Management Staff in the Office of Nuclear Reactor Regulation and by NRC regional

personnel. This report is to provide a comprehensive description of the implementation and verification status of TMI Action Plan Requirements, USIs, GSIs, and other MPAs that have been resolved and involve implementation of an action or actions by licensees. This report makes the information available to other interested parties, including the public. An additional purpose of this report is to serve as a follow-on to NUREG-0933, "A Prioritization of Generic Safety Issues," which tracks safety issues up until requirements are approved for imposition at licensed plants or until the NRC issues a request for action by licensees.

NUREG-1460 R01: GUIDE TO NRC REPORTING AND RECORD-KEEPING REQUIREMENTS. Compiled From Requirements In Title 10 Of The U.S. Code Of Federal Regulations As Codified On December 31, 1993. COLLINS, M.; SHELTON, B. Office of Information Resources Management (Post 890205). July 1994. 254pp. 9408180211. 80625:001.

This compilation includes in the first two sections the reporting and recordkeeping requirements applicable to U.S. Nuclear Regulatory Commission (NRC) licensees and applicants and to members of the public. It includes those requirements codified in Title 10 of the Code of Federal Regulations, Chapter I, on December 31, 1993. It also includes, in a separate section, any of those requirements that were superseded or discontinued between January 1992 and December 1993. Finally, the appendix lists mailing and delivery addresses for NRC Headquarters and Regional Offices mentioned in the compilation.

NUREG-1462 V01: FINAL SAFETY EVALUATION REPORT RELATED TO THE CERTIFICATION OF THE SYSTEM 80+ DESIGN. Chapters 1-14. Docket No. 52-002. (Asea Brown Boveri-Combustion Engineering) * Associate Director for Advanced Reactors & License Renewal (Post 910918). August 1994. 583pp. 9409200306. 80953:001.

This final safety evaluation report (FSER) documents the technical review of the System 80+ standard design by the U.S. Nuclear Regulatory Commission (NRC) staff. The application for the System 80+ design was submitted by Combustion Engineering, Inc., now Asea Brown Boveri-Combustion Engineering (ABB-CE) as an application for design approval and subsequent design certification pursuant to 10 CFR § 52.45. System 80+ is a pressurized water reactor with a rated power of 3914 megawatts thermal (MWt) and a design power of 3992 MWt at which accidents are analyzed. Many features of the System 80+ are similar to those of ABB-CE's System 80 design from which it evolved. Unique features of the System 80+ design include: a large spherical, steel containment; an in-containment refueling water storage tank; a reactor cavity flooding system, hydrogen ignitors, and a safety depressurization system for severe accident mitigation; a combustion gas turbine for an alternate ac source; and an advanced digitally based control room. On the basis of its evaluation and independent analyses, the NRC staff concludes that ABB-CE's application for design certification meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant to the System 80+ standard design.

NUREG-1462 V02: FINAL SAFETY EVALUATION REPORT RELATED TO THE CERTIFICATION OF THE SYSTEM 80+ DESIGN. Chapters 15-22 And Appendices. Docket No. 52-002. (Asea Brown Boveri-Combustion Engineering) * Associate Director for Advanced Reactors & License Renewal (Post 910918). August 1994. 557pp. 9409200314. 80956:001.

See NUREG-1462, V01 abstract.

NUREG-1470 V03: FINANCIAL STATEMENT FOR FISCAL YEAR 1993. * Office of the Controller (Post 890205). August 1994. 110pp. 9409090026. 80816:001.

The Chief Financial Officers Act of 1990 requires the NRC Chief Financial Officer to prepare and submit an annual financial statement to the Director of the Office of Management and Budget (OMB). The OMB has replaced the requirement for the CFO's Annual Report with the annual financial statement. The

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annual financial statement was previously included in the Chief Financial Officer's Annual Report. This report is the third annual report for the NRC and includes an overview of the NRC, the audited principal financial statements and audit reports for fiscal year 1993, and supplemental financial and management information.

NUREG-1471: CONCEPT OF OPERATIONS WITH ORGANIZATION CHARTS. NRC Incident Response. HIMES, J.; JACKSON, K.; LOPRESTI, F. Division of Operational Assessment (Post 870413). February 1994. 69pp. 9403140209. 78456:175.

The U.S. Nuclear Regulatory Commission (NRC) regulates nuclear power plants and certain other civilian nuclear facilities and materials to protect the public health and safety and to preserve environmental quality. While the foremost objective of regulation is to prevent accidents, the NRC is also prepared to help its licensees and State and local governments mitigate the consequences of any that might occur. This document describes the NRC concept, purposes, and organization for performing essential functions during a Federal response to a severe reactor accident, with an emphasis on State and Federal coordination.

NUREG-1475: APPLYING STATISTICS. LURIE, D. Office of the Controller (Post 890205). MOORE, R.H. Bonneville Power Administration. February 1994. 594pp. 9405310242. 79584:001.

Applying Statistics is both a reference book and a textbook on statistical methods. Although the majority of the book's examples are taken from the field of nuclear regulation, the statistical principles that they embody are applicable across a wide spectrum of scientific, technical, and life problems. Many of these more widely based problems are addressed directly in the text and illustrated with appropriate data. The 21 chapters are interspersed with "For discussion:" sections to promote the reader's insights and intuitive understanding of the appropriate techniques. Each chapter is introduced by a list of items covered in the chapter and is concluded by a list of concepts to be remembered. A special section entitled "Afterword" lists areas of statistics that are not covered in the book; it gives a brief description of each of these areas along with one or two applicable and recommended references. Applying Statistics also provides an extensive set of statistical tables. Each table is explained and its use illustrated by at least one example; thus, the tables can be used independently of the rest of the material. The book's notational conventions, as they apply to the eight common statistical distributions that support the various analyses, are displayed on the inside front and back covers and the flyleaves.

NUREG-1478: NON-POWER REACTOR OPERATOR LICENSING EXAMINER STANDARDS. * Division of Reactor Controls & Human Factors (Post 921004). June 1994. 200pp. 9408030122. 80424:001.

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

NUREG-1484 V01: FINAL ENVIRONMENTAL IMPACT STATE-

MENT FOR THE CONSTRUCTION AND OPERATION OF CLAIBORNE ENRICHMENT CENTER, HOMER, LOUISIANA. Docket No. 70-3070. Louisiana Energy Services, L.P. Environmental Impact Statement. ZEITOUN, A. Science Applications International Corp. (formerly Science Applications, Inc.). August 1994. 390pp. 9409070096. 80790:001.

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

NUREG-1484 V02: FINAL ENVIRONMENTAL IMPACT STATEMENT FOR THE CONSTRUCTION AND OPERATION OF CLAIBORNE ENRICHMENT CENTER, HOMER, LOUISIANA. Docket No. 70-3070. Louisiana Energy Services, L.P. Comments And Responses. ZEITOUN, A. Science Applications International Corp. (formerly Science Applications, Inc.). August 1994. 480pp. 9409070125. 80792:001.

Volume 2 contains the comments received on the Draft EIS and the responses to those comments.

NUREG-1486: FINAL SAFETY EVALUATION REPORT TO LICENSE THE CONSTRUCTION AND OPERATION OF A FACILITY TO RECEIVE, STORE AND DISPOSE OF 11E(2) BYPRODUCT MATERIAL NEAR CLIVE, UTAH. Docket No. 40-8989. (Envirocare of Utah, Inc.) ABUEID, R.; BRUMMETT, E.; BYKOSKI, L.; et al. Division of Low-Level Waste Management & Decommissioning (870413-940402). January 1994. 116pp. 9402220120. 78191:291.

The Final Safety Evaluation Report (FSER) summarizes the U.S. Nuclear Regulatory Commission (NRC) staff's review of Envirocare of Utah, Inc.'s (Envirocare's) application for a license to receive, store, and dispose of uranium and thorium byproduct material (as defined in Section 11e.(2) of the Atomic Energy Act of 1954, as amended) at a site near Clive, Utah. Envirocare proposes to dispose of high-volume, low-activity Section 11e.(2) byproduct material in separate earthen disposal cells on a site where the applicant currently disposes of naturally occurring radioactive material (NORM), low-level waste, and mixed waste under license by the Utah Department of Environmental Quality. The NRC staff review of the December 23, 1991, license application, as revised by page changes dated July 2 and August 10, 1992, April 5, 7, and 10, 1993, and May 3, 6, 7, 11, and 21, 1993, has identified open issues in geotechnical engineering, water resources protection, radon attenuation, financial assurance, and radiological safety. The NRC will not issue a license for the proposed action until Envirocare adequately resolves these open issues.

NUREG-1488: REVISED LIVERMORE SEISMIC HAZARD ESTIMATES FOR SIXTY-NINE NUCLEAR POWER PLANT SITES EAST OF THE ROCKY MOUNTAINS. Final Report. SOBEL, P. Division of Engineering (Post 921004). April 1994. 108pp. 9405310204. 79589:235.

The draft version of this report presented updated Lawrence Livermore National Laboratory (LLNL) probabilistic seismic hazard analysis estimates for 69 nuclear power plant sites in the region of the United States east of the Rocky Mountains. LLNL performed a re-elicitation of seismicity and ground motion experts to improve their estimates of uncertainty in seismicity parameters and ground motion models. Using these revised inputs, LLNL updated the seismic hazard estimates documented in NUREG/CR-5250 (1989). These updated hazard estimates will be used in future NRC actions. The draft was issued for public comment in October 1993. By the end of the public comment period, February 28, 1994, comments had been received from two nuclear industry companies. The comments from these companies neither contested nor suggested amendments to the technical data conveyed in the report. Rather, they both suggest changes in the Individual Plant External Event Examination (IPEEE) program scope. This report is not the forum for discussion of the IPEEE program. Possible modification to the scope of the IPEEE will be examined in its own setting. Therefore, there are no technical differences between the draft report and this final report. Any information as to modifications to the IPEEE program will be provided to the public via an NRC general communication.

NUREG-1489: A REVIEW OF NRC STAFF USES OF PROBABILISTIC RISK ASSESSMENT. * Ofc of the Executive Director for Operations. March 1994. 274pp. 9405040223. 79107:001.

The NRC staff uses probabilistic risk assessment (PRA) and risk management as important elements of its licensing and regulatory processes. In October 1991, the NRC's Executive Director for Operations established the PRA Working Group to address concerns identified by the Advisory Committee on Reactor Safeguards with respect to unevenness and inconsistency in the staff's current uses of PRA. After surveying current staff uses of PRA and identifying needed improvements, the Working Group defined a set of basic principles for staff PRA use and identified three areas for improvements: guidance development, training enhancements, and PRA methods development. For each area of improvement, the Working Group took certain actions and recommended additional work. The Working Group recommended integrating its work with other recent PRA-related activities the staff completed and improving staff interactions with PRA users in the nuclear industry. The Working Group took two key actions by developing general guidance for two uses of PRA within the NRC (that is, screening or prioritizing reactor safety issues and analyzing such issues in detail) and developing guidance on basic terms and methods important to the staff's uses of PRA.

NUREG-1491: SAFETY EVALUATION REPORT FOR THE CLAIBORNE ENRICHMENT CENTER, HOMER, LOUISIANA. Docket No. 70-3070, Louisiana Energy Services, L.P. * Division of Fuel Cycle Safety & Safeguards (Post 930207). January 1994. 300pp. 9402150293. 78120:017.

This report documents the U.S. Nuclear Regulatory Commission (NRC) staff safety review and evaluation of the Louisiana Energy Services, L.P. (LES, the applicant) application for a license to possess and use byproducts, source, and special nuclear material and to enrich natural uranium to a maximum of five percent U-235 by the gas centrifuge process. The plant, to be known as the Claiborne Enrichment Center (CEC), would be constructed near the town of Homer in Claiborne Parish, Louisiana. At full production in a given year, the plant will receive approximately 4,700 tonnes of feed UF(6), and produce 870 tonnes of low-enriched UF(6), and 3,830 tonnes of depleted UF(6) tails. Facility construction, operation, and decommissioning are expected to last five, thirty, and seven years, respectively. The objective of the review is to evaluate the potential adverse impacts of operation of the facility on worker and public health and safety under both normal operating and accident conditions. The review also considers the management organization, administrative programs, and financial qualifications provided to assure safe design and operation of the facility. The

NRC staff concludes that the applicant's descriptions, specifications, and analyses provide an adequate basis for safety review of facility operations and that construction and operation of the facility does not pose an undue risk to public health and safety.

NUREG-1492 DFC: REGULATORY ANALYSIS ON CRITERIA FOR THE RELEASE OF PATIENTS ADMINISTERED RADIOACTIVE MATERIAL. Draft Report For Comment. SCHNEIDER, S.; MCGUIRE, S.A. Division of Regulatory Applications (870413-941217). BEHLING, U.H.; et al. S. Cohen & Associates, Inc. May 1994. 78pp. 9406290327. 80014:210.

The Nuclear Regulatory Commission (NRC) has received two petitions to amend its regulations in 10 CFR Parts 20 and 35 as they apply to doses received by members of the public exposed to patients released from a hospital after they have been administered radioactive material. While the two petitions are not identical, they both request that the NRC establish a dose limit of 5 millisieverts (0.5 rem) per year for individuals exposed to patients who have been administered radioactive materials. This Regulatory Analysis evaluates three alternatives. Alternative 1 is for the NRC to amend its patient release criteria in 10 CFR 35.75 to use the more stringent dose limit of 1 millisievert (0.1 rem) per year in 10 CFR 20.1301(a) for its patient release criteria. Alternative 2 is for the NRC to continue using the existing patient release criteria in 10 CFR 35.75 of 1,110 megabecquerels (30 millicuries) of activity or a dose rate at one meter from the patient of 0.05 millisievert (5 millirem) per hour. Alternative 3 is for the NRC to amend the patient release criteria in 10 CFR 35.75 to specify a dose limit of 5 millisieverts (0.5 rem) for patient release. The evaluation demonstrates that, except for a few diagnostic procedures using iodine-131, diagnostic procedures are unaffected by the choice of alternative. Only some therapeutic administrations of radioactive material could be affected by the choice of alternative. The evaluation indicates that Alternative 1 would cause a prohibitively large increase in the national health care cost from retaining patients in a hospital longer and would cause significant personal and psychological costs to patients and their families. The choice of Alternatives 2 or 3 would affect only thyroid cancer patients treated with iodine-131. For those patients, Alternative 3 would result in less hospitalization than Alternative 2. Alternative 3 has a potential decrease in national health care cost of \$30,000,000 per year but would increase the potential collective dose from released therapy patients by about 2,700 person-rem per year, mainly to family members. Alternative 3 would also have personal and psychological benefits for the patients and their families.

NUREG-1494: STAFF TECHNICAL POSITION ON CONSIDERATION OF FAULT DISPLACEMENT HAZARDS IN GEOLOGIC REPOSITORY DESIGN. MCCONNELL, K.I.; LEE, M.P. Division of Waste Management (NMSS 940403). September 1994. 37pp. 9411080076. 81649:261.

Nuclear Regulatory Commission regulations for the disposal of spent nuclear fuel and high-level radioactive waste in a geologic repository recognize that fault displacement is a potentially adverse condition (10 CFR 60.122(c)(11) and 60.122(c)(20)). However, they do not prohibit designing the geologic repository against the effects of such a potentially adverse condition. This Staff Technical Position recognizes the acceptability of designing the geologic repository to take into account the attendant effects (e.g., displacement) of faults of regulatory concern and expresses the staff's views on what is needed from the U.S. Department of Energy if it chooses to locate structures, systems, and components important to safety or important to waste isolation in areas that contain faults of regulatory concern.

NUREG-1495: OVERALL REVIEW STRATEGY FOR THE NUCLEAR REGULATORY COMMISSION'S HIGH-LEVEL WASTE REPOSITORY PROGRAM. JOHNSON, R.L. Division of Waste Management (NMSS 940403). November 1994. 39pp. 9412070075. 81956:200.

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The Overall Review Strategy gives general guidance to the Nuclear Regulatory Commission staff for conducting its license application and pre-license application reviews. These reviews are in support of the Commission's construction authorization decision for a geologic repository for the disposal of high-level radioactive waste. Objectives and strategies are defined that focus the staff's reviews on determining compliance with requirements of 10 CFR Part 60. These strategies define how the staff prioritizes its reviews on those key technical uncertainties considered to be most important to repository performance. Strategies also give guidance for developing, in an integrated way, the License Application Review Plan together with supporting performance assessments, analyses, and research.

NUREG-1496 V1 DFC: GENERIC ENVIRONMENTAL IMPACT STATEMENT IN SUPPORT OF RULEMAKING ON RADIOLOGICAL CRITERIA FOR DECOMMISSIONING OF NRC-LICENSED NUCLEAR FACILITIES. Main Report. Draft Report For Comment. * Division of Regulatory Applications (870413-941217). August 1994. 170pp. 9408300003. 80700:001.

The action being considered in this Draft Generic Environmental Impact Statement (GEIS) is an amendment to the Nuclear Regulatory Commission's (NRC) regulations in 10 CFR Part 20 to include radiological criteria for decommissioning of lands and structures at nuclear facilities. Under the National Environmental Policy Act (NEPA), all Federal agencies must consider the effect of their actions on the environment. To fulfill NRC's responsibilities under NEPA, the Commission is preparing this GEIS which analyzes alternative courses of action and the costs and impacts associated with those alternatives. In preparing the GEIS, the following approach was taken: (1) a listing was developed of regulatory alternatives for establishing radiological criteria for decommissioning; (2) for each alternative, a detailed analysis and comparison of incremental impacts, both radiological and nonradiological, to workers, members of the public, and the environment, and costs, was performed; and (3) based on the analysis of impacts and costs, preliminary recommendations were provided. Recommendations contained in the GEIS include those related to the definition of decommissioning, the scope of rulemaking, the radiological criteria, restrictions on use, citizen participation, use of the GEIS in site specific cases, and minimization of contamination.

NUREG-1496 V2 DFC: GENERIC ENVIRONMENTAL IMPACT STATEMENT IN SUPPORT OF RULEMAKING ON RADIOLOGICAL CRITERIA FOR DECOMMISSIONING OF NRC-LICENSED NUCLEAR FACILITIES. Appendices. Draft Report For Comment. * Division of Regulatory Applications (870413-941217). August 1994. 600pp. 9408300006. 80698:001.

See NUREG-1496,V01,DRF FC abstract.

NUREG-1497: INTERIM LICENSING CRITERIA FOR PHYSICAL PROTECTION OF CERTAIN STORAGE OF SPENT FUEL. DWYER,P.A. Division of Fuel Cycle Safety & Safeguards (Post 930207). November 1994. 20pp. 9412300178. 82159:182.

This document presents interim criteria to be used in the physical protection licensing of certain spent fuel storage installations. Installations that will be reviewed under this criteria are those that store power reactor spent fuel at decommissioned power reactor sites; independent spent fuel storage installations located outside of the owner-controlled area of operating nuclear power reactors; monitored retrievable storage installations owned by the Department of Energy, designed and constructed specifically for the storage of spent fuel; the proposed geologic repository operations area; or permanently shutdown power reactors still holding a Part 50 license. This criteria in this document does not apply to the storage of spent fuel within the owner-controlled area of operating nuclear power reactors.

NUREG-1499: REASSESSMENT OF THE NRC'S PROGRAM FOR PROTECTING ALLEGERS AGAINST RETALIATION. * Ofc of the Executive Director for Operations. January 1994. 200pp. 9402150379. 78123:349.

On July 6, 1993, the Nuclear Regulatory Commission's (NRC's) Executive Director for Operations established a review team to reassess the NRC's program for protecting allegers against retaliation. The team evaluated the current system, and solicited comments from various NRC offices, other Federal agencies, former allegers, and the public. This report is subject to agency review. The report summarizes current processes and gives an overview of current problems. It discusses: (1) ways in which licensees can promote a quality-conscious work environment, in which all employees feel free to raise concerns without fear of retaliation; (2) ways to improve the NRC's overall handling of allegations; (3) the NRC's involvement in the Department of Labor process; (4) related NRC enforcement practices; and (5) methods other than investigation and enforcement that may be useful in treating allegations of potential or actual discrimination. Recommendations are given in each area.

NUREG-1500: WORKING DRAFT REGULATORY GUIDE ON RELEASE CRITERIA FOR DECOMMISSIONING: NRC STAFF'S DRAFT FOR COMMENT. DAILY,M.C.; HUFFERT,A.M.; CARDILE,F.; et al. Division of Regulatory Applications (870413-941217). August 1994. 168pp. 9410130081. 81264:171.

The Nuclear Regulatory Commission's (NRC) regulations in 10 CFR Part 20 are being amended to include radiological criteria for decommissioning of lands and structures at nuclear facilities. 10 CFR Part 20, Subpart E establishes criteria for the remediation of contaminated sites or facilities that will allow their release for future use with or without restrictions. The criteria include a Total Effective Dose Equivalent (TEDE) limit of 15 mrem/year (0.15 mSv/y) that should not be exceeded by an average individual among those who could potentially receive the greatest exposure from any residual activity within a facility or on a site. The criteria also require a licensee to reduce any residual radioactivity to as-low-as-reasonably-achievable (ALARA) levels. This staff draft guide describes acceptable procedures for determining the predicted dose level (PDL) from any residual radioactivity at the site. It describes the basic features of the calculational models and the associated default assumptions and parameter values the NRC staff would find acceptable in calculating PDLs. Appendices A, B, and C provide numerical values that can be used to estimate the dose from residual radioactivity remaining at a site. Since 10 CFR Part 20, Subpart E introduces several new concepts, definitions and discussions are included in a regulatory position concepts section of the guide to assist licensees in understanding some of the philosophy underlying the rule.

NUREG-1501 DRFT: BACKGROUND AS A RESIDUAL RADIOACTIVITY CRITERION FOR DECOMMISSIONING. Appendix A To The Draft Generic Environmental Impact Statement In Support Of Rulemaking On Radiological Criteria For Decommissioning Of NRC.... HUFFERT,A.M.; MECK,R.A. Division of Regulatory Applications (870413-941217). MILLER,K.M. Energy,Dept.of, Environmental Measurements Laboratory. August 1994. 100pp. 9408250012. 80659:001.

This report was originally published as an appendix to the draft U.S. Nuclear Regulatory Commission (NRC) document entitled, "Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for Decommissioning of NRC-Licensed Nuclear Facilities." Because of the great interest in this report by members of the public, citizen and environmental organizations, academicians, licensees, and regulators, the NRC staff is publishing this report separately, so that it can be readily available to a diverse audience. This report was created to assist both the NRC staff and interested members of the public in evaluating background radiation (background) as a decommissioning criterion, by serving as a primer on background and providing information on the existing applications of background in regulatory criteria and standards. This report also discusses some of the methods available to measure and distinguish between the very low radiation levels associated with background and man-made sources of radiation. Two approach-

es are considered for applying background as a decommissioning criterion; these are the use of background dose rates and background radionuclide concentrations. This report concludes that the temporal and spatial variability of background produces a wide range of doses to United States residents, which prevents the application of background dose rates as a decommissioning criterion. Instead, this report recommends that local background radionuclide concentrations serve as a benchmark for decommissioning criteria, while taking into account the concept of reducing residual radioactivity to a level as low as is reasonably achievable.

NUREG-1502: ASSESSMENT OF DATABASES AND MODELING CAPABILITIES FOR THE CANDU 3 DESIGN. CARLSON,D.E.; MEYER,R.O. Division of Systems Research (880717-941217). July 1994. 107pp. 9408220041. 80637:099.

As part of the research program associated with the preliminary review of the CANDU 3 design, the NRC Office of Nuclear Regulatory Research (RES) has completed an assessment of databases and modeling capabilities that might be needed to support the CANDU 3 design. To ensure full coverage of the design, a detailed assessment methodology was developed by the RES staff and was implemented with help from research projects at three national laboratories. This report integrates and summarizes the database and modeling assessments, including major contributions from these laboratories.

NUREG-1503 V01: FINAL SAFETY EVALUATION REPORT RELATED TO THE CERTIFICATION OF THE ADVANCED BOILING WATER REACTOR DESIGN.Docket No. 52-001.(General Electric Nuclear Energy) * Associate Director for Advanced Reactors & License Renewal (Post 910918). July 1994. 876pp. 9408260011. 80681:001.

This Safety Evaluation Report (SER) documents the technical review of the U.S. Advanced Boiling Water Reactor (ABWR) standard design by the U.S. Nuclear Regulatory Commission (NRC) staff. The application for the ABWR design was submitted by GE Nuclear Energy. The NRC staff concludes that, subject to satisfactory resolution of the confirmatory items identified in Section 1.8 of this SER, GE's application for design certification meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant to the U.S. ABWR standard design.

NUREG-1503 V02: FINAL SAFETY EVALUATION REPORT RELATED TO THE CERTIFICATION OF THE ADVANCED BOILING WATER REACTOR DESIGN.Appendices. Docket No. 52-001.(General Electric Nuclear Energy) * Associate Director for Advanced Reactors & License Renewal (Post 910918). July 1994. 209pp. 9408250023. 80659:098.

See NUREG-1503,V01 abstract.

NUREG-1504: REVIEW CRITERIA FOR THE PHYSICAL FITNESS TRAINING REQUIREMENTS IN 10 CFR PART 73. BROWN,C. Division of Fuel Cycle Safety & Safeguards (Post 930207). September 1994. 18pp. 9410130302. 81276:081.

This document provides review criteria that will be used in reviewing and approving revised physical security plans submitted by licensees which are required to meet the physical fitness requirements in 10 CFR Part 73.

NUREG-1508: DRAFT ENVIRONMENTAL IMPACT STATEMENT TO CONSTRUCT AND OPERATE THE CROWNPOINT SOLUTION MINING PROJECT,CROWNPOINT, NEW MEXICO.Docket No. 40-8968.(Hydro Resources,Inc.) * Division of Waste Management (NMSS 940403). October 1994. 154pp. 9411160064. 81767:008.

This Draft Environmental Impact Statement (DEIS) addresses issuing a combined source and byproduct material license and minerals operating leases for Federal and Indian lands to Hydro Resources, Inc. This action would authorize the company to conduct in situ leach uranium mining in McKinley County, New Mexico. Such mining would involve drilling wells to the ore bodies, then recirculating ground water with added oxygen to

mobilize uranium found in the rock. Uranium would then be removed from the solution using ion exchange technology in processing plants located in three areas. A central plant would provide drying and packaging equipment for the entire project. This DEIS was prepared by an interagency review group, including the Nuclear Regulatory Commission, the Bureau of Indian Affairs, and the Bureau of Land Management. After weighing the environmental, technical, and other benefits of the Hydro Resources mining project against environmental and other costs, the reviewing agencies concluded that the appropriate action is to issue the requested license and leases authorizing the applicant to proceed with the project as discussed in this DEIS. The DEIS describes and evaluates (1) the purpose of and need for the proposed action, evaluated under the National Environmental Policy Act, and the agencies' implementing regulations, (2) alternatives considered, (3) existing environmental conditions, and (4) environmental consequences of the proposed action and proposed mitigating measures.

NUREG-1511: REACTOR PRESSURE VESSEL STATUS REPORT. STROSNIDER,J.; WICHMAN,K.; ELLIOT,B.; et al. Division of Engineering (Post 921004). December 1994. 174pp. 9412300173. 82158:251.

The Nuclear Regulatory Commission (NRC) issued Generic Letter 92-01 (GL 92-01) to obtain information needed to assess compliance with requirements and commitments regarding reactor vessel integrity in view of certain concerns raised in the NRC staff's review of reactor vessel integrity for the Yankee Nuclear Power Station. This report gives a brief description of the reactor pressure vessel (RPV), followed by a discussion of the radiation embrittlement of RPV beltline materials and the two indicators for measuring embrittlement, the end-of-license (EOL) reference temperature and the EOL upper-shelf energy. It also summarizes the GL 92-01 effort and presents, for all 37 boiling water reactor plants and 74 pressurized water reactor plants in the United States, the current status of compliance with regulatory requirements related to ensuring RPV integrity. The staff has evaluated the material data needed to predict neutron embrittlement of the reactor vessel beltline materials. These data will be stored in a computer database entitled the reactor vessel integrity database (RVID). This database will be updated annually to reflect the changes made by the licensees in future submissions and will be used by the NRC staff to assess the issues related to vessel structural integrity.

NUREG/CP-0127: PROCEEDINGS OF THE CSNI SPECIALISTS MEETING ON FUEL-COOLANT INTERACTIONS. * Organization for Economic Cooperation & Development. March 1994. 377pp. 9404110363. NEA/CSNI/R(93)8. 78818:001.

A specialists meeting on fuel-coolant interactions was held in Santa Barbara, CA from January 5-7, 1993. The meeting was sponsored by the United States Nuclear Regulatory Commission in collaboration with the Committee on the Safety of Nuclear Installation (CSNI) of the OECD Nuclear Energy Agency (NEA) and the University of California at Santa Barbara. The objectives of the meeting are to cross-fertilize on-going work, provide opportunities for mutual check points, seek to focus the technical issues on matters of practical significance and re-evaluate both the objectives as well as path of future research.

NUREG/CP-0133 V01: PROCEEDINGS OF THE TWENTY-FIRST WATER REACTOR SAFETY INFORMATION MEETING.Plenary Session; Advanced Reactor Research; Advanced Control System Technology; Advanced Instrumentation & Control Hardware; Human Factors.... MONTELEONE,S. Brookhaven National Laboratory. April 1994. 600pp. 9405100230. 79217:001.

This three-volume report contains 90 papers out of the 102 that were presented at the Twenty-First Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, during the week of October 25-27, 1993. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear

14 Main Citations and Abstracts

safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from France, Germany, Japan, Russia, Switzerland, Taiwan, and 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-0133 V02: PROCEEDINGS OF THE TWENTY-FIRST WATER REACTOR SAFETY INFORMATION MEETING. Severe Accident Research. MONTELEONE, S. Brookhaven National Laboratory. April 1994. 632pp. 9405100236. 79220:001.

See NUREG/CP-0133, V01 abstract.

NUREG/CP-0133 V03: PROCEEDINGS OF THE TWENTY-FIRST WATER REACTOR SAFETY INFORMATION MEETING. Primary System Integrity; Aging Research, Products & Applications; Structural & Seismic Engineering; Seismology & Geology. MONTELEONE, S. Brookhaven National Laboratory. April 1994. 454pp. 9405100242. 79218:236.

See NUREG/CP-0133, V01 abstract.

NUREG/CP-0135: WORKSHOP ON ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT. LOFARO, R.; GUNTHER, W.; VILLARAN, M.; et al. Brookhaven National Laboratory. May 1994. 309pp. 9406060111. BNL-NUREG-52409. 79619:001.

Questions concerning the Environmental Qualification (EQ) of electrical equipment used in commercial nuclear power plants have recently become the subject of significant interest to the U.S. Nuclear Regulatory Commission (NRC). Initial questions centered on whether compliance with the EQ requirements for older plants were adequate to support plant operation beyond 40 years. After subsequent investigation, the NRC Staff concluded that questions related to the differences in EQ requirements between older and newer plants constitute a potential generic issue which should be evaluated for backfit, independent of license renewal activities. EQ testing of electric cables was performed by Sandia National Laboratories (SNL) under contract to the NRC in support of license renewal activities. Results showed that some of the environmentally qualified cables either failed or exhibited marginal insulation resistance after a simulated plant life of 20 years during accident simulation. This indicated that the EQ process for some electric cables may be non-conservative. These results raised questions regarding the EQ process including the bases for conclusions about the qualified life of components based upon artificial aging prior to testing.

NUREG/CP-0136: PROCEEDINGS OF THE DIGITAL SYSTEMS RELIABILITY AND NUCLEAR SAFETY WORKSHOP. September 13-14, 1993, Rockville Crowne Plaza Hotel, Rockville, Maryland. WALLACE, D.R.; CUTHILL, B.B.; IPPOLITO, L.M.; et al. National Institute of Standards & Technology (formerly National Bureau of Standards). March 1994. 373pp. 9405050310. NIST SP 500-216. 79163:001.

The United States Nuclear Regulatory Commission (NRC), in cooperation with the National Institute of Standards and Technology conducted the Digital Systems Reliability and Nuclear Safety Workshop on September 13-14, 1993, in Rockville, Maryland. The workshop provided a forum for the exchange of information among experts within the nuclear industry, experts from other industries, regulators and academia. The information presented at this workshop provided in-depth exposure of the NRC staff and the nuclear industry to digital systems design safety issues and also provided feedback to the NRC from outside experts regarding identified safety issues, proposed regulatory positions, and intended research associated with the use of digital systems in nuclear power plants. Technical presentations provided insights on areas where current software engineering practices may be inadequate for safety-critical systems, on potential solutions for development issues, and on methods for reducing risk in safety-critical systems. This report contains an analysis of results of the workshop, the papers presented, panel presentations, and summaries of discussions at this workshop.

NUREG/CP-0137 V01: PROCEEDINGS OF THE THIRD NRC/ASME SYMPOSIUM ON VALVE AND PUMP TESTING. Held At The Hyatt Regency Hotel, Washington, DC, July 18-21, 1994. Session 1A - Session 2C. * EG&G Idaho, Inc. July 1994. 570pp. 9408250007. EGG-2742. 80657:001.

The 1994 Symposium on Valve and Pump Testing, jointly sponsored by the Board of Nuclear Codes and Standards of the American Society of Mechanical Engineers and by the Nuclear Regulatory Commission, provides a forum for the discussion of current programs and methods for inservice testing and motor-operated valve testing at nuclear power plants. The symposium also provides an opportunity to discuss the need to improve that testing in order to help ensure the reliable performance of pumps and valves. The participation of industry representatives, regulators, and consultants results 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-0137 V02: PROCEEDINGS OF THE THIRD NRC/ASME SYMPOSIUM ON VALVE AND PUMP TESTING. Held At The Hyatt Regency Hotel, Washington, DC, July 18-21, 1994. Session 3A - Session 4B. * EG&G Idaho, Inc. July 1994. 387pp. 9408030114. EGG-2742. 80424:197.

See NUREG/CP-0137, V01 abstract.

NUREG/CP-0138: PROCEEDINGS OF WORKSHOP I IN ADVANCED TOPICS IN RISK AND RELIABILITY ANALYSIS. Model Uncertainty: Its Characterization And Quantification. MOSLEH, A.; SMIDTS, C.; et al. Maryland, Univ. of, College Park, MD. SIU, N. EG&G Idaho, Inc. October 1994. 259pp. 9411220239. 81806:025.

This report contains the proceedings of an international workshop: Model Uncertainty: Its Characterization and Quantification, held October 20-22, 1993 in Annapolis, Maryland, USA. It includes: i) a brief introduction to the topic of model uncertainty and to the workshop; ii) contributed papers from 17 experts from a variety of organizations (government agencies, research laboratories, universities, and private industry), disciplines (physics, engineering, statistics, law, medicine, management science), and areas of interest (nuclear reactor safety, waste repository performance, fire safety, human health, environmental risk, decision analysis, and uncertainty analysis); and iii) summaries of discussions of three working groups formed during the workshop. The primary findings of the workshop as documented in these proceedings are as follows. First, a formal probabilistic framework for dealing with model uncertainty exists; some further development is needed to make it more practical. Second, less formal quantitative approaches also exist and have been applied in some problems. Third, coping strategies which don't require direct quantification of model uncertainty are available for immediate usage. Fourth, quantitative procedures that "average out" irreconcilable differences between models and modelers do not provide sufficient information regarding predictive uncertainty to risk managers. A prioritized research agenda for developing a standardized approach for treating model uncertainty is presented.

NUREG/CP-0139: TRANSACTIONS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING. MONTELEONE, S. Brookhaven National Laboratory. October 1994. 141pp. 9411040088. 81647:001.

This report contains summaries of papers on reactor safety research to be presented at the 22nd Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel, Bethesda, Maryland, October 24-26, 1994. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. Summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the nuclear industry, and from foreign governments and industry are also included. The summaries have been compiled in one report to

provide a basis for meaningful discussion and information exchange during the course of the meeting and are given in the order of their presentation in each session.

NUREG/CR-2850 V12: DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1990. BAKER, D.A. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1994. 194pp. 9412120234. PNL-4221. 81997:001.

Population and individual radiation dose commitments have been estimated from reported radionuclide releases from commercial power reactors operating during 1990. Fifty-year dose commitments for a one-year exposure from both liquid and atmospheric releases were calculated for four population groups (infant, child, teen-ager and adult) residing between 2 and 80 km from each of 72 reactor sites. This report tabulates the results of these calculations, showing the dose commitments for both water and airborne pathways for each age group and organ. Also included for each of the sites is an estimate of individual doses which are compared with 10 CFR Part 50, Appendix I design objectives. The total collective dose commitments (from both liquid and airborne pathways) for each site ranged from a high of 15 person-rem to a low of 0.002 person-rem for the sites with plants in operation and producing power during the year. The arithmetic mean was 1.1 person-rem. The total population dose for all sites was estimated at 78 person-rem for the 130 million people considered at risk. The individual dose commitments estimated for all sites were below the Appendix I design objectives.

NUREG/CR-2907 V12: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1991. TICHLER, J.; DOTY, K.; CONGEMI, J. Brookhaven National Laboratory. May 1994. 360pp. 9406210302. BNL-NUREG-51581. 79835:001.

Releases of radioactive materials in airborne and liquid effluents from commercial light water reactors during 1991 have been compiled and reported. Data on solid waste shipments as well as selected operating information have been included. This report supplements earlier annual reports issued by the former Atomic Energy Commission and the Nuclear Regulatory Commission. The 1991 release data are summarized in tabular form. Data covering specific radionuclides are summarized.

NUREG/CR-3145 V10: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION. Final Report, October 1986-September 1992. RUFF, L.; LAFORGE, R.; THORSON, R.; et al. Michigan, Univ. of, Ann Arbor, MI. February 1994. 115pp. 9403140217. 78456:242.

Earthquake activity in the Western Ohio-Indiana region has been monitored with a seismograph network consisting of nine stations located in west-central Ohio and four stations located in Indiana. Six local and regional earthquakes have been recorded from October 1990 to September 1992 with magnitudes ranging from 0.6 to 5.0. A total of 36 local and regional earthquakes have been recorded in the past 6-year period (October 1986 to September 1992). Overall a total of 78 local and regional earthquakes have been recorded since the network went into operation in 1977. There was a peak in seismicity in 1986, including the July 12, 1986 St. Marys' event ($m_b=4.5$), followed by an anomalously low level of seismicity for about 2 years. The most unusual feature of the seismicity in the past year is the occurrence of three earthquakes in Indiana. The locations of the felt earthquakes are scattered across central Indiana; an area that had been aseismic. Analysis of arrival time data accumulated over the past 14 years shows that the Anna region crustal structure is "slower" than the average mid-continent crustal structure. This implies that the proposed Keewenawan rift in the Anna region has a different structure than that of other Keewenawan rifts in the mid-continent.

NUREG/CR-3950 V09: FUEL PERFORMANCE REPORT FOR 1991. PAINTER, C.L.; ALVIS, J.M.; BEYER, C.E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1994. 123pp. 9409200326. PNL-5210. 80963:162.

This annual report, the fourteenth in a series, provides a brief description of fuel performance during 1991 in commercial nuclear power plants and an indication of trends. Brief summaries of fuel operating experience, fuel design changes, fuel surveillance programs, high-burnup experience, problem areas, and items of general significance are provided. References to more detailed information and related U.S. Nuclear Regulatory Commission evaluations are included.

NUREG/CR-4219 V10 N1: HEAVY SECTION STEEL TECHNOLOGY PROGRAM. Semiannual Progress Report For October 1992 - March 1993. PENNELL, W.E. Oak Ridge National Laboratory. September 1994. 149pp. 9410120224. ORNL/TM-9593. 81254:001.

The Heavy-Section Steel Technology (HSST) Program is conducted for the Nuclear Regulatory Commission by Oak Ridge National Laboratory (ORNL). The program focus is on the development and validation of technology for the assessment of fracture-prevention margins in commercial nuclear reactor pressure vessels. The HSST Program is organized in 12 tasks: (1) program management, (2) fracture methodology and analysis, (3) material characterization and properties, (4) special technical assistance (5) fracture analysis computer programs, (6) cleavage-crack initiation, (7) cladding evaluations, (8) pressurized-thermal shock technology, (9) analysis methods validation, (10) fracture evaluations, (11) warm prestressing, and (12) biaxial loading effects. The program tasks have been structured to place emphasis on the resolution fracture issues with near-term licensing significance. Resources to execute the research tasks are drawn from ORNL with subcontract support from universities and other research laboratories. Close contact is maintained with the sister Heavy-Section Steel Irradiation Program at ORNL and with related research programs both in the United States and abroad. This report provides an overview of principal developments in each of the 12 program tasks from October 1992 to March 1993.

NUREG/CR-4409 V05: DATA BASE ON DOSE REDUCTION RESEARCH PROJECTS FOR NUCLEAR POWER PLANTS. KHAN, T.A.; YU, C.K. Brookhaven National Laboratory. May 1994. 193pp. 9406200329. BNL-NUREG-51934. 79831:106.

This is the fifth volume in a series of reports that provide information on dose reduction research and health physics technology for nuclear power plants. The information is taken from two of several databases maintained by Brookhaven National Laboratory's ALARA Center for the U.S. Nuclear Regulatory Commission. The research section of the report covers dose reduction projects that are in the experimental or developmental phase. It includes topics such as steam generator degradation, decontamination, robotics, improvements in reactor materials, and inspection techniques. The section on health physics technology discusses dose-reduction efforts that are in place or in the process of being implemented at nuclear power plants. A total of 105 new or updated projects are described. All project abstracts from this report are made available to nuclear industry professionals with access to a fax machine through our ACEFAX system or a computer with a modem and the proper communications software through our ACE system. Detailed descriptions of how to access all of our databases electronically are in the appendices of the report.

NUREG/CR-4513 R01: ESTIMATION OF FRACTURE TOUGHNESS OF CAST STAINLESS STEELS DURING THERMAL AGING IN LWR SYSTEMS. CHOPRA, O.K. Argonne National Laboratory. August 1994. 83pp. 9409230297. ANL-93/22. 81010:229.

This report presents a revision of the procedure and correlations presented earlier in NUREG/CR-4513, ANL-90/42 (June

1991) for predicting the change in mechanical properties of cast stainless steel components due to thermal aging during service in light water reactors at 280-330 degrees C (535-625 degrees F). The correlations presented in this report are based on an **expanded database and have been optimized with mechanical-property data on cast stainless steels aged up to ~58,000 h at 290-350 degrees C (554-633 degrees F).** The correlations for estimating the change in tensile stress, including the Ramberg/Osgood parameters for strain hardening, are also described. The fracture toughness J-R curve, tensile stress, and Charpy-impact energy of aged cast stainless steels are estimated from known material information. Mechanical properties of a specific cast stainless steel are estimated from the extent and kinetics of thermal embrittlement. Embrittlement of cast stainless steels is characterized in terms of room-temperature Charpy-impact energy. The extent or degree of thermal embrittlement at "saturation," i.e., the minimum impact energy that can be achieved for a material after long-term aging, is determined from the chemical composition of the steel. Charpy-impact energy as a function of time and temperature of reactor service is estimated from the kinetics of thermal embrittlement, which are also determined from the chemical composition. The initial impact energy of the unaged steel is required for these estimations.

NUREG/CR-4551 V01 R1: EVALUATION OF SEVERE ACCIDENT RISKS: METHODOLOGY FOR THE CONTAINMENT, SOURCE TERM, CONSEQUENCE, AND RISK INTEGRATION ANALYSES. GORHAM, E.D.; BREEDING, R.J. Sandia National Laboratories. HELTON, J.C.; et al. Arizona State Univ., Tempe, AZ. December 1993. 279pp. 9402220215. SAND86-1309. 78198:001.

NUREG-1150 examines the risk to the public from five nuclear power plants. The NUREG-1150 plant studies are Level III probabilistic risk assessments (PRAs) and, as such, they consist of four analysis components: accident frequency analysis, accident progression analysis, and source term analysis, and consequence analysis. This volume summarizes the methods utilized in performing the last three components and the assembly of these analyses into an overall risk assessment. The NUREG-1150 analysis approach is based on the following ideas: (1) general and relatively fast-running models for the individual analysis components, (2) use of Monte Carlo techniques together with an efficient sampling procedure to propagate uncertainties, (3) use of expert panels to develop distributions for important phenomenological issues, and (4) automation of the overall analysis. Many features of the new analysis procedures were adopted to facilitate a comprehensive treatment of uncertainty in the complete risk analysis. Uncertainties in the accident frequency, accident progression and source term analyses were included in the overall uncertainty assessment. The uncertainties in the consequence analysis were not included in this assessment. A large effort was devoted to the development of procedures for obtaining expert opinion and the execution of the development of procedures for obtaining expert opinion and the execution of these procedures to quantify parameters and phenomena for which there is large uncertainty and divergent opinions in the reactor safety community.

NUREG/CR-4599 V03 N2: SHORT CRACKS IN PIPING AND PIPING WELDS. Semiannual Report, October 1992 - March 1993. WILKOWSKI, G.M.; BRUST, F.; FRANCINI, R.; et al. Battelle Memorial Institute, Columbus Laboratories. March 1994. 112pp. 9404040024. BMI-2173. 78736:256.

This is the sixth semiannual report of the U.S. Nuclear Regulatory Commission's 4-year research program "Short Cracks in Piping and Piping Welds" which began in March 1990. The objective is to verify and improve fracture analyses for circumferentially cracked nuclear piping with cracks sizes typically found during in-service flaw evaluations. Progress in the through-wall-cracked pipe efforts involved (1) verification of deformation plasticity under nonproportional loading, (2) evaluation of the effect of weld metal strength on various J-estimation schemes, and (3) development of new GE/EPRI functions. Surface-

cracked pipe evaluations involved (1) material characterization of B&W C-Mn-Mo submerged arc weld metal, and (2) 3D finite-element mesh refinement study. The toughness of the bimetallic weld fusion line was evaluated and showed unusual fracture behavior based on the results of the Charpy tests. The dynamic strain aging J-R tests confirmed the screening criterion developed earlier in the program. The results from this program to date necessitated several additional efforts. These were initiated and have been reported here. Presentation of the results from this program to the ASME Section XI Pipe Flaw Evaluation Working Group is also summarized here.

NUREG/CR-4639 V5R4P2: NUCLEAR COMPUTERIZED LIBRARY FOR ASSESSING REACTOR RELIABILITY (NUCLARR). Volume 5: Data Manual. Part 2: Human Error Probability (HEP) Data. REECE, W.J.; GILBERT, B.G.; RICHARDS, R.E. EG&G Idaho, Inc. September 1994. 453pp. 9410130115. EGG-2458. 81265:001.

This data manual contains a hard copy of the information in the Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR) Version 3.5 database, which is sponsored by the U.S. Nuclear Regulatory Commission. NUCLARR was designed as a tool for risk analysis. Many of the nuclear reactors in the U.S. and several outside the U.S. are represented in the NUCLARR database. NUCLARR includes both human error probability estimates for workers at the plants and hardware failure data for nuclear reactor equipment. Aggregations of these data yield valuable reliability estimates for probabilistic risk assessments and human reliability analyses. The data manual is organized to permit manual searches of the information if the computerized version is not available.

NUREG/CR-4639 V5R4P3: NUCLEAR COMPUTERIZED LIBRARY FOR ASSESSING REACTOR RELIABILITY (NUCLARR). Volume 5: Data Manual. Part 3: Hardware Component Failure Data. REECE, W.J.; GILBERT, B.G.; RICHARDS, R.E. EG&G Idaho, Inc. September 1994. 186pp. 9410130099. EGG-2458. 81267:001.

See NUREG/CR-4639, V05, R04, P02 abstract.

NUREG/CR-4667 V17: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, April 1993 - September 1993. CHOPRA, O.K.; CHUNG, H.M.; KARLSEN, T.; et al. Argonne National Laboratory. June 1994. 70pp. 9407130212. ANL-94/16. 80189:133.

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

NUREG/CR-4674 V17: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1992 A STATUS REPORT. Main Report And Appendix A. COX,D.F.; CLETCHER,J.W.; COPINGER,D.A.; et al. Oak Ridge National Laboratory. December 1993. 150pp. 9402220223. ORNL/NOAC-232. 78198:280.

Twenty-seven operational events with conditional probabilities of subsequent severe core damage of 1.0×10^{-6} or higher occurring at commercial light-water reactors during 1992 are considered to be precursors to potential core damage. These are described along with associated significance estimates, categorization, and subsequent analyses. 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 V18: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1992 A STATUS REPORT. Appendices B, C, D, E, F, And G. COX,D.F.; CLETCHER,J.W.; COPINGER,D.A.; et al. Oak Ridge National Laboratory. December 1993. 688pp. 9402220241. ORNL/NOAC-232. 78194:299.

See NUREG/CR-4674,V17 abstract.

NUREG/CR-4674 V19: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1993 A STATUS REPORT. Main Report And Appendices A-D. VANDEN HEUVEL,L.; CLETCHER,J.W.; COPINGER,D.A.; et al. Oak Ridge National Laboratory. September 1994. 200pp. 9411220300. ORNL/NOAC-232. 81806:272.

Sixteen operational events that affected sixteen commercial light-water reactors during 1993 and that are considered to be precursors to potential severe core damage are described. All these events had conditional probabilities of subsequent severe core damage greater than or equal to 1.0×10^{-6} . These events were identified by first computer-screening the 1993 licensee event reports from commercial light-water reactors to identify those that could potentially be precursors. Candidate precursors were then selected and evaluated in a process similar to that used in previous assessments. Selected events underwent engineering evaluation that identified, analyzed, and documented the precursors. Other events designated by the Nuclear Regulatory Commission (NRC) also underwent a similar evaluation. Finally, documented precursors were submitted for review by licensees and NRC headquarters and regional offices to ensure the plant design and its response to the precursor were correctly characterized. This study is a continuation of earlier work, which evaluated 1969-1981 and 1984-1992 events. The report discusses the general rationale for this study, the selection and documentation of events as precursors, and the estimation of conditional probabilities of subsequent severe core damage for events. This document is bound in two volumes: Volume 19 contains the main report and Appendices A-D; Volume 20 contains Appendices E and F.

NUREG/CR-4674 V20: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1993 A STATUS REPORT. Appendices E And F. VANDEN HEUVEL,L.; CLETCHER,J.W.; COPINGER,D.A.; et al. Oak Ridge National Laboratory. September 1994. 400pp. 9411220304. ORNL/NOAC-232. 81807:120.

See NUREG/CR-4674,V19 abstract.

NUREG/CR-4816 R02: PR-EDB: POWER REACTOR EMBRITTLEMENT DATA BASE,VERSION 2. Program Description. STALLMANN,F.W.; WANG,J.A.; KAM,F.B.K.; et al. Oak Ridge National Laboratory. January 1994. 122pp. 9402150283. ORNL/TM-10328. 78121:034.

Investigations of regulatory issues such as vessel integrity over plant life, vessel failure, and sufficiency of current codes, Standard Review Plans (SRPs) and Guides for license renewal can be greatly expedited by the use of a well-designed computerized data base. Also, such a data base is essential for the val-

idation of embrittlement prediction models by researchers. The Power Reactor Embrittlement Data Base (PR-EDB) is such a comprehensive collection of data for U.S. commercial nuclear reactors. The current version of the PR-EDB contains the Charpy test data that were irradiated in 252 capsules of 96 reactors and consists of 207 data points for heat-affected-zone (HAZ) materials (98 different HAZ), 227 data points for weld materials (105 different welds), 524 data points for base materials (136 different base materials), including 297 plate data points (85 different plates), 119 forging data points (31 different forging), and 108 correlation monitor materials data points (3 different plates). The data files are given in dBASE format and can be accessed with any personal computer using the DOS operating system. "User-friendly" utility programs are used to retrieve and select specific data, manipulate data, display data to the screen or printer, and to fit and plot Charpy impact data. The results of several studies investigated are presented in Appendix D.

NUREG/CR-4833: LARGE AREA SELF-POWERED GAMMA RAY DETECTOR. Phase II Development Of A Source Position Monitor For Use On Industrial Radiographic Units. LEVERT,F.E. K.E.M.P. Corp. January 1994. 47pp. 9402150301. 78122:053.

The purpose of this research was to develop a large area self-powered gamma detector (LASPGD) capable of detecting the movement of sealed radiation sources into and out of industrial radiographic units and to construct a prototype source position monitor (SPM) for these units utilizing the LASPGD. Prototype isotropic and directional LASPGDs, with solid and inert gas dielectrics, were developed and extensively tested using calibrated gamma sources (i.e., Cs-137 and Co-60). The sensitivities of the isotropic detectors, with inert gas dielectrics, were found to be approximately a factor of ten greater than those measured for the solid dielectric LASPGDs. Directionally sensitive self-powered detectors were found to exhibit a forward-to-back hemispherical sensitivity ratio of approximately 2 to 1. Industrial radiographic units containing Ir-192 sources with different activities were used to test the performance on the SPM. The SPM, which utilized a gas dielectric LASPGD, performed as designed. That is, the current generated in the LASPGD was converted to a voltage, amplified and used to control the on/off state of an incandescent lamp. The incandescent lamp, which functions as the source/out warning indicator, flashes at a rate of one flash per second when the source is in use (i.e. out of its shield).

NUREG/CR-4838: MICROCOMPUTER APPLICATIONS OF, AND MODIFICATIONS TO, THE MODULAR FAULT TREES. ZIMMERMAN,T.L.; GRAVES,N.L. Energy, Inc. PAYNE,A.C.; et al. Sandia National Laboratories. October 1994. 300pp. 9412070078. SAND88-1887. 81956:241.

The LaSalle Probabilistic Risk Assessment was the first major application of the modular logic fault trees after the IREP program. In the process of performing the analysis, many errors were discovered in the fault tree modules that led to difficulties in combining the modules to form the final system fault trees. These errors are corrected in the revised modules listed in this report. In addition, the application of the modules in terms of editing them and forming them into the system fault trees was inefficient. Two programs were written to help alleviate this problem: (1) MODEDIT - This program allows an operator to retrieve a file for editing, edit the file for the plant specific application, perform some general error checking while the file is being modified, and store the file for later use, and (2) INDEX - This program checks that the modules that are supposed to form one fault tree all link up appropriately before the files are loaded onto the mainframe computer. Lastly, the modules were not designed for relay type logic common in BWR designs but for solid state type logic. Some additional modules were defined for modeling relay logic.

NUREG/CR-4918 V07: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS. Progress Report On Field Experiments At A Humid Region Site, Beltsville, Maryland. SCHULZ, R.K. California, Univ. of, Los Angeles, CA. RIDKY, R.W. Maryland, Univ. of, College Park, MD. O'DONNELL, E. Waste Management Branch (Post 941217). December 1994. 34pp. 9501180233. 82341:180.

The project objective is to assess means for controlling waste infiltration through waste disposal unit covers in humid regions. Experimental work is being performed in large scale lysimeters (70'x45'x10') at Beltsville, MD and results of the assessment are applicable to disposal of LLW, uranium mill tailings, hazardous waste, and sanitary landfills. Three concepts are under investigation: (1) resistive layer barrier, (2) conductive layer barrier, and bioengineering water management. The resistive layer barrier consists of compacted earth (clay). The conductive layer barrier is a special case of the capillary barrier and it requires a flow layer (e.g. fine sandy loam) over a capillary break. As long as unsaturated conditions are maintained water is conducted by the flow layer to below the waste. This barrier is most efficient at low flow rates and is thus best placed below a resistive layer barrier. Such a combination of the resistive layer over the conductive layer barrier promises to be highly effective provided there is no appreciable subsidence. Bioengineering water management is a surface cover that is designed to accommodate subsidence. It consists of impermeable panels which enhance run-off and limit infiltration. Vegetation is planted in narrow openings between panels to transpire water from below the panels. This system has successfully dewatered two lysimeters thus demonstrating that this procedure could be used for remedial action ("drying out") existing water-logged disposal sites at low cost.

NUREG/CR-5128 R01: EVALUATION AND REFINEMENT OF LEAK-RATE ESTIMATION MODELS. PAUL, D.D.; AHMAD, J.; SCOTT, P.M.; et al. Battelle Memorial Institute, Columbus Laboratories. June 1994. 97pp. 9407260077. BMI-2164. 80345:001.

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

NUREG/CR-5161 V02: EVALUATION OF SAMPLING PLANS FOR INSERVICE INSPECTION OF STEAM GENERATOR TUBES. Comprehensive Analytical And Monte Carlo Simulation Results For Several Sampling Plans. KURTZ, R.J.; HEASLER, P.G.; BAIRD, D.B. Battelle Memorial Institute, Pacific Northwest Laboratory. February 1994. 72pp. 9404040034. PNL-6462. 78736:057.

This report summarizes the results of three previous studies to evaluate and compare the effectiveness of sampling plans for steam generator tube inspections. An analytical evaluation and Monte Carlo simulation techniques were the methods used to evaluate sampling plan performance. To test the performance of candidate sampling plans under a variety of conditions, ranges of inspection system reliability were considered along with different distributions of tube degradation. Results from the eddy current reliability studies performed with the retired-from-service Surry 2A steam generator were utilized to guide the selection of appropriate probability of detection and flaw sizing models for use in the analysis. Different distributions of tube degradation were selected to span the range of conditions that might exist in operating steam generators. The principal means

of evaluating sampling performance was to determine the effectiveness of the sampling plan for detecting and plugging defective tubes. A summary of key results from the eddy current reliability studies is presented. The analytical and Monte Carlo simulation analyses are discussed along with a synopsis of key results and conclusions.

NUREG/CR-5229 V06: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1993. Annual Report. MCCONNELL, J.W.; ROGERS, R.D.; JASTROW, J.D.; et al. EG&G Idaho, Inc. May 1994. 63pp. 9406070051. EGG-2577. 79684:208.

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

NUREG/CR-5247 V01 R2: RASCAL VERSION 2.1 USER'S GUIDE. SJOREEN, A.L. Oak Ridge National Laboratory. ATHEY, G.F. Athey Consulting. RAMSDALL, J.V.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1994. 196pp. 9501180194. ORNL-6820. 82345:001.

The Radiological Assessment System for Consequence Analysis, Version 2.1 (RASCAL 2.1) has been developed for use during response to radiological emergencies. The system supplements assessments based on plant conditions and quick estimates based on hand-calculational methods. The model is designed to provide a comparison to EPA Protective Action Guidance and thresholds for acute health effects. The model was developed to allow consideration of the dominant aspects of source term, transport, dose, and consequences. Revisions to RASCAL 2.0 have required the release of this new version of the system. Three new source-term calculations have been added to ST-DOSE in RASCAL 2.1. They are (1) source term based on the reactor containment monitor reading, (2) a source term for a spent fuel pool accident, and (3) an isotopic concentration source term. Field Measurements to Dose (FM-DOSE) calculations have been modified to include consideration of the effect of delay for re-entry on first-year and second-year dose, to incorporate a variable resuspension rate, and to compute a factor used to estimate first-year dose from R/hr measurements on the ground. Also, RASCAL 2.1 supports the saving of cases for later display or modification and the use of a mouse for user input.

NUREG/CR-5247 V02 R2: RASCAL VERSION 2.1 WORKBOOK. ATHEY, G.F. Athey Consulting. SJOREEN, A.L. Oak Ridge National Laboratory. MCKENNA, T.J. Incident Response Branch. December 1994. 148pp. 9501180200. 82345:197.

The Radiological Assessment System for Consequence Analysis, Version 2.1 (RASCAL 2.1) was developed for use by the NRC personnel who respond to radiological emergencies. This workbook complements the RASCAL 2.1 User's Guide (NUREG/CR-5247, Vol. 1, Rev. 2). The workbook contains exercises designed to familiarize the user with the computer-based tools of RASCAL through hands-on problem solving. The workbook contains four major sections. The first RASCAL familiarization exercise to acquaint the user with the operation of the

forms, menus, online help, and documentation. The latter three sections contain exercises in using the three tools of RASCAL Version 2.1: DECAY, FM-DOSE, and ST-DOSE. A discussion section describing how the tools could be used to solve the problems follows each set of exercises.

NUREG/CR-5314 V05: INSIGHTS FOR AGING MANAGEMENT OF MAJOR LWR COMPONENTS METAL CONTAINMENTS. SHAH,V.N.; SINHA,U.P. Idaho National Engineering Laboratory. SMITH,S.K. Ogden Environmental & Energy Services (Formerly Multiple Dynamics Corp). March 1994. 104pp. 9404060267. EGG-2562. 78760:001.

This report evaluates the available technical information and field experience related to management of aging damage to light water reactor metal containments. A generic aging management approach is suggested for the effective and comprehensive aging management of metal containments to ensure their safe operation. The major concern is corrosion of the embedded portion of the containment vessel and detection of this damage. The electromagnetic acoustic transducer and half-cell potential measurement are potential techniques to detect corrosion damage in the embedded portion of the containment vessel. Other corrosion-related concerns include inspection of corrosion damage on the inaccessible side of BWR Mark I and II containment vessels and corrosion of the BWR Mark I torus and emergency core cooling system piping that penetrates the torus, and transgranular stress corrosion cracking of the penetration bellows. Fatigue-related concerns include reduction in the fatigue life (a) of a vessel caused by roughness of the corroded vessel surface and (b) of bellows because of any physical damage. Maintenance of surface coatings and sealant at the metal-concrete interface is the best protection against corrosion of the vessel.

NUREG/CR-5344 R01: REPLACEMENT ENERGY COST ANALYSIS PACKAGE (RECAP): USER'S GUIDE. VANKUIKEN,J.C.; WILLING,D.L. Argonne National Laboratory. July 1994. 54pp. 9407250267. ANL/EES-TM-364. 80328:141.

A microcomputer program called the Replacement Energy Cost Analysis Package (RECAP) has been developed to assist the U.S. Nuclear Regulatory Commission (NRC) in determining the replacement energy costs associated with short-term shutdowns or deratings of one or more nuclear reactors. The calculations are based on the seasonal, unit-specific cost estimates for 1993-1996 previously published in NRC Report NUREG/CR-4012, Vol. 3 (1992), for all 112 U.S. reactors. Because the RECAP program is menu-driven, the user can define specific case studies in terms of such parameters as the units to be included, the length and timing of the shutdown or derating period, the unit capacity factors, and the reference year for reporting cost results. In addition to simultaneous shutdown cases, more complicated situations, such as overlapping shutdown periods or shutdowns that occur in different years, can be examined through the use of a present-worth calculation option.

NUREG/CR-5359: REVIEW OF ELASTIC STRESS AND FATIGUE-TO-FAILURE DATA FOR BRANCH CONNECTIONS AND TEES IN RELATION TO ASME DESIGN CRITERIA FOR NUCLEAR POWER PIPING SYSTEMS. RODABAUGH,E.C.; MOORE,S.E.; GWALTNEY,R.C. Oak Ridge National Laboratory. May 1994. 124pp. 9406060115. ORNL/TM-11152. 79618:200.

This is the third in a series of reports on the state-of-the-art design guidance for piping system branch connections and tees provided by Section III of the ASME Boiler and Pressure Vessel Code. The other reports covered primary or limit-loads and nozzle flexibility. The principal objective of this report, as with the others, was to identify needed improvements in the design methods and criteria of the Code based on the evaluation of the available information. This report does not propose changes in the design procedure of the Code. This report discusses the evaluation of stresses in branch connections and tees, correlation of these stresses with fatigue failures, and the code rules for protection against fatigue failure in design applications. Be-

cause of the extensive amount of available information, the report was divided into two parts. Part I discusses cyclic internal pressure loading and Part II discusses moment loadings for the branch and run. The cyclic pressure loading fatigue parameters are mostly based on leakage, whereas, if the parameters were based on crack initiation, different and possibly higher values would be developed. The fatigue evaluation procedure, which attempts to relate fatigue strength of piping components to strain-controlled, polished bar, and fatigue data appears to be inaccurate on the conservative side for high amplitude cycles and inaccurate on the unconservative side for low amplitude cycles.

NUREG/CR-5403: PREDICTING THE PRESSURE DRIVEN FLOW OF GASES THROUGH MICRO-CAPILLARIES AND MICRO-ORIFICES. ANDERSON,B.L.; CARLSON,R.W.; FISCHER,L.E. Lawrence Livermore National Laboratory. November 1994. 22pp. 9411220229. UCRL-ID-118245. 81806:001.

Experimentally measured gas flow rates obtained from the literature were compared to the predictions obtained using the constitutive flow equations given in ANSI N14.5, the "American National Standard for Radioactive Materials-Leakage Tests on Packages for Shipment," to determine whether the equations apply to the predictions of gas flow rates from leaking containment vessels used to transport radioactive materials. The equations were applied to both (1) the data set according to the recommendations given in ANSI N14.5, and (2) globally to the complete data set. It was found that: For flow rates $\leq 1 \text{ atm} \cdot \text{cm}^3/\text{s}$, the predictions obtained with the continuum and molecular flow equation provided good agreement with the experimental values and the choked flow equation resulted in over-prediction of the measured values. For flow rates $> 1 \text{ atm} \cdot \text{cm}^3/\text{s}$, the molecular and continuum flow equation over-predicted the measured flow rates and the predictions obtained with the choked flow equation agreed well with the experimental values. Since leakage rates from packages used to transport radioactive materials are almost always $< 1 \text{ atm} \cdot \text{cm}^3/\text{s}$, it is suggested that the continuum and molecular flow equation be used for gas flow rate predictions related to these applications.

NUREG/CR-5407: ASSESSMENT OF THE IMPACT OF DEGRADED SHEAR WALL STIFFNESSES ON SEISMIC PLANT RISK AND SEISMIC DESIGN LOADS. KLAMERUS,E.W.; BOHN,M.P. Sandia National Laboratories. JOHNSTON,J.J.; et al. EQE Engineering Consultants (formerly EQE Engineering, Inc.). February 1994. 588pp. 9404010183. SAND93-0234. 78720:042.

Test results sponsored by the USNRC have shown that reinforced shear wall (Seismic Category I) structures exhibit stiffnesses and natural frequencies which are smaller than those calculated in the design process. The USNRC has sponsored Sandia National Labs to perform an evaluation of the effects of the reduced frequencies on several existing seismic PRAs in order to determine the seismic risk implications inherent in these test results. This report presents the results for the re-evaluation of the seismic risk for three nuclear power plants: the Peach Bottom Atomic Power Station, the Zion Nuclear Power Plant, and Arkansas Nuclear One - Unit 1 (ANO-1). Increases in core damage frequencies for seismic initiated events at Peach Bottom were 25 to 30 percent (depending on whether LLNL or EPRI hazard curves were used). At the ANO-1 site, the corresponding increases in plant risk were 10 percent (for each set of hazard curves). Finally, at Zion, there was essentially no change in the computed core damage frequency when the reduction in shear wall stiffness was included. In addition, an evaluation of deterministic "design-like" structural dynamic calculations with and without the shear stiffness reductions was made. Deterministic loads calculated for these two cases typically increased on the order of 10 to 20 percent for the affected structures.

NUREG/CR-5535 V06: RELAP5/MOD3 CODE MANUAL. Validation Of Numerical Techniques In RELAP5/MOD3. SHIEH, A.S. EG&G Idaho, Inc. RANSOM, V.H. Purdue Univ., West Lafayette, IN. KRISHNAMURTHY Pacific-Nuclear Co. October 1994. 154pp. 9411160091. EGG-2705. 81759:138.

The RELAP code has been developed for best-estimate transient simulation of light-water reactor coolant systems during large and small break loss-of-coolant accidents and as well as operational transients. The code models the coupled behavior of the reactor coolant system and the core during a severe accident transient and models large- and small- break loss-of-coolant accidents and operational transients, such as anticipated transient without scram, loss of offsite power, loss of feedwater, and loss of flow. A generic modeling approach is used that permits as much of a particular system to be modeled as necessary. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater conditioning systems. RELAP5/MOD3 code documentation is divided into five volumes: Volume I provides modeling theory and associated numerical schemes; Volume II contains detailed instructions for code application and input data preparation; Volume III provides the results of developmental assessment cases that demonstrate and verify the models used in the code; Volume IV presents a detailed discussion of RELAP5 models and correlations; Volume V contains guidelines that have evolved over the past several years through use of the RELAP5 code; and Volume VI contains descriptions of numerical modeling of two-phase flow used in RELAP 5 and discussions on stability, accuracy, and convergence of the numerical techniques in RELAP5.

NUREG/CR-5535 V07: RELAP5/MOD3 CODE MANUAL. Summaries And Reviews Of Independent Code Assessment Reports. SLOAN, S.M.; SCHULTZ, R.R.; WILSON, G.E. EG&G Idaho, Inc. June 1994. 120pp. 9407260037. EGG-2596. 80344:207.

Summaries of RELAP5/MOD3 code assessments, a listing of the assessment matrix, and a chronology of the various versions of the code are given. Results from these code assessments have been used to formulate a compilation of some of the strengths and weaknesses of the code. These results are documented in the report. Volume 7 was designed to be updated periodically and to include the results of the latest code assessments as they become available. Consequently the user of Volume 7 should check that the latest revision is available.

NUREG/CR-5569 R01: HEALTH PHYSICS POSITIONS DATA BASE. KERR, G.D.; BORGES, T.; STAFFORD, R.S.; et al. Oak Ridge National Laboratory. February 1994. 300pp. 9403140176. ORNL/TM-12067. 78453:001.

The Health Physics Positions (HPPOS) Data Base of the Nuclear Regulatory Commission (NRC) is a collection of NRC staff positions on a wide range of topics involving radiation protection (health physics). It consists of 328 documents in the form of letters, memoranda, and excerpts from technical reports. The HPPOS Data Base was developed by the NRC Headquarters and Regional Offices to help ensure uniformity in inspections, enforcement, and licensing actions. Staff members of the Oak Ridge National Laboratory (ORNL) have assisted the NRC staff in summarizing the documents during the preparation of this NUREG report. These summaries are also being made available as a "stand alone" software package for IBM and IBM-compatible personal computers. The software package for this report is called HPPOS Version 2.0. A variety of indexing schemes were used to increase the usefulness of the NUREG report and its associated software. The software package and the summaries in the report are written in the context of "new" 10 CFR Part 20 (§§20.1001 - 20.2401). The purpose of this NUREG report is to allow interested individuals to familiarize themselves with the contents of the HPPOS Data Base and with the basis of many NRC decisions and regulations. The HPPOS summaries and original documents are intended to serve as a source of information for radiation protection programs at nuclear research

and power reactors, nuclear medicine, and other industries that either process or use nuclear materials.

NUREG/CR-5591 V02 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Semiannual Progress Report For October 1990 - March 1991. CORWIN, W.R. Oak Ridge National Laboratory. July 1994. 36pp. 9408180191. ORNL/TM-11568. 80613:001.

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, and in particular the fracture toughness properties, of typical pressure vessel steels as they relate to light water reactor pressure-vessel integrity. Effects of specimen size, material chemistry, product form and microstructure, irradiation fluence, flux, temperature and spectrum, and post-irradiation annealing are being examined on a wide range of fracture properties. Analyses of precleavage stable ductile tearing of high-copper welds 72W and 73W demonstrated that the size effects observed in the transition region are not due to substantial differences in ductile tearing behavior. Drop-weight tests, Charpy impact tests, and chemical analyses of the Midland reactor low upper-shelf welds were completed showing large variations in bulk copper content, transition temperature, and upper-shelf energy. Atom probe field ion microscopy analyses revealed no evidence of fine copper precipitates or clusters in the unirradiated Midland welds but a substantial depletion in the copper concentration in the matrix.

NUREG/CR-5591 V02 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Semiannual Progress Report For April-September 1991. CORWIN, W.R. Oak Ridge National Laboratory. October 1994. 37pp. 9411040083. ORNL/TM-11568. 81646:257.

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, and in particular the fracture toughness properties, of typical pressure vessel steels as they relate to light-water reactor pressure-vessel integrity. Effects of specimen size, material chemistry, product form and microstructure, irradiation fluence, flux, temperature and spectrum, and post-irradiation annealing are being examined on a wide range of fracture properties. The HSSI Program is arranged into 10 tasks: (1) program management, (2) K(lc) curve shift in high-copper welds, (3) K(la) curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K(lc) and K(la) curve shifts in low upper-shelf welds, (6) irradiation effects in a commercial low upper-shelf weld, (7) microstructural analysis of irradiation effects, (8) in-service aged material evaluations, (9) correlation monitor materials, and (10) special technical assistance. This report provides an overview of the activities within each of these tasks from April to September 1991.

NUREG/CR-5625: TECHNICAL SUPPORT FOR A PROPOSED DECAY HEAT GUIDE USING SAS2H/ORIGEN-S DATA. HERMANN, O.W.; PARKS, C.V.; RENIER, J.P. Oak Ridge National Laboratory. September 1994. 130pp. 9411080038. ORNL-6698. 81647:231.

Major revisions are proposed to the current U.S. Nuclear Regulatory Commission decay heat rate guide entitled "Regulatory Guide 3.54, Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation," using a new data base produced by the SAS2H analysis sequence of the SCALE-4 system. The data base for the proposed guide revision has been significantly improved by increasing the number and range of parameters that generally characterize pressurized-water-reactor (PWR) and boiling-water-reactor (BWR) spent fuel assemblies. Using generic PWR and BWR assembly models, calculations were performed with each model for six different burnups at each of three separate specific powers to produce heat rates at 20 cooling times in the range of 1 to 110 y. The proposed procedure specifies proper interpolation formulae for the tabu-

lated heat generation rates. Adjustment formulae for the interpolated values are provided to account for differences in initial (235)U enrichment and changes in the specific power of a cycle from the average value. Finally, safety factor formulae were derived as a function of burnup, cooling time, and type of reactor. The proposed guide revision was designed to be easier to use. Also, the complete data base and guide procedure is incorporated into an interactive code called LWRARC which can be executed on a personal computer. The report shows adequate comparisons of heat rates computed by SAS2H/ORIGEN-S and measurements for 10 BWR and 10 PWR fuel assemblies. The average differences of the computed minus the measured heat rates of fuel assemblies were $-07 \pm 2.6\%$ for the BWR and $1.5 \pm 1.3\%$ for the PWR. In addition, a detailed analysis of the proposed procedure indicated the method and equations to be valid.

NUREG/CR-5680 V01: THE IMPACT OF ENVIRONMENTAL CONDITIONS ON HUMAN PERFORMANCE. A Handbook Of Environmental Exposures. ECHEVERRIA,D.; BARNES,V.; BITTNER,A.; et al. Battelle Human Affairs Research Centers. September 1994. 110pp. 9411080032. 81650:001.

A comprehensive review of the technical literature was conducted regarding the impact of environmental conditions on human performance applicable to nuclear power plant workers. The environmental conditions considered were vibration, noise, heat, cold, and light. Research staff identified potential human performance deficits (e.g., decreased dexterity, impaired vision, hearing loss, memory deficiency) along a continuum of increasing occupational exposure, ranging from exposures that result in no deficit to exposures that resulted in significant performance problems. Specific deficits were included in the report if there was sound scientific evidence that environmental exposure resulted in those performance deficits. The levels associated with each deficit were then compared to the protection afforded by existing occupational exposure standards. Volume I is a handbook for use by NRC inspectors to help them determine the impact of specific environmental conditions on licensee personnel performance. It discusses the units used to measure each condition, discusses the effects of the condition on task performance, presents an example of the assessment of each condition in a nuclear power plant, and discusses potential methods for reducing the effects of exposure to the condition.

NUREG/CR-5680 V02: THE IMPACT OF ENVIRONMENTAL CONDITIONS ON HUMAN PERFORMANCE. A Critical Review Of The Literature. ECHEVERRIA,D.; BARNES,V.; BITTNER,A.; et al. Battelle Human Affairs Research Centers. September 1994. 248pp. 9411080026. 81650:109.
See NUREG/CR-5680,V01 abstract.

NUREG/CR-5726: REVIEW OF THE DIABLO CANYON PROBABILISTIC RISK ASSESSMENT. BOZOKI,G.E.; FITZPATRICK,R. Brookhaven National Laboratory. BOHN,M.P.; et al. Sandia National Laboratories. August 1994. 500pp. 9409260069. BNL-NUREG-52288. 81028:001.

This report details the review of the Diablo Canyon Probabilistic Risk Assessment (DCPRA). The study was performed under contract from the Probabilistic Risk Analysis Branch, Office of Nuclear Reactor Research, USNRC by Brookhaven National Laboratory. The DCPRA is a full scope Level I effort and although the review touched on all aspects of the PRA, the internal events and seismic events received the vast majority of the review effort. The report includes a number of independent systems analyses, sensitivity studies, importance analyses as well as conclusions on the adequacy of the DCPRA for use in the Diablo Canyon Long Term Seismic Program.

NUREG/CR-5758 V04: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY. Annual Summary Of Program Performance Reports CY 1993. WESTRA,C.; FORSLUND,C.; FIELD,I.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1994. 101pp. 9409230281. PNL-9985. 81006:184.

This report summarizes the data from the semiannual reports on Fitness for Duty programs submitted to the NRC by utilities for two reporting periods: January 1 through June 30, 1993, and July 1 through December 31, 1993. During 1993, licensees reported that they had conducted 242,966 tests for the presence of illegal drugs and alcohol. Of these tests, 1,512 (.62%) were confirmed positive. Positive test results varied by category of test and category of worker. The majority of positive test results (952) were obtained through pre-access training. Of tests conducted on workers having access to the protected area, there were 341 positive tests from random testing and 163 positive tests from for-cause testing. Follow-up testing of workers who had previously tested positive resulted in 56 positive tests. For-cause testing resulted in the highest percentage of positive tests: about 22 percent of for-cause tests were positive. This compares with a positive test rate of 1.04 percent of pre-access tests and .23 percent of random tests. Positive test rates also varied by category of worker. When all types of tests are combined (pre-access, random, for-cause, and follow-up testing), short-term contractor personnel had the highest positive test rate at .97 percent. Licensee employees and long-term contractors had lower combined positive test rates (.25% and .21%, respectively). Of the substances tested, marijuana was responsible for the highest percentage of positive test results (49.56%), followed by cocaine (23.41%) and alcohol (22.65%).

NUREG/CR-5812: MANAGING AGING IN NUCLEAR POWER PLANTS. Insights From NRC Maintenance Team Inspection Reports. FRESCO,A.; SUBUDHI,M.; GUNTHER,W.; et al. Brookhaven National Laboratory. December 1993. 204pp. 9402220158. BNL-NUREG-52309. 78196:259.

A plant's maintenance program is the principal vehicle through which age-related degradation is managed. From 1988 to 1991, the NRC evaluated the maintenance program of every nuclear power plant in the United States. Forty-four out of a total of 67 of the reports issued on these in-depth team inspections were reviewed for insights into the strengths and weaknesses of the programs as related to the need to understand and manage the effects of aging on nuclear plant systems, structures, and components. Relevant information was extracted from these inspection reports and sorted into several categories, including Specific Aging Insights, Preventive Maintenance, Predictive Maintenance and Condition Monitoring, Post Maintenance Testing, Failure Trending, Root Cause Analysis and Usage of Probabilistic Risk Assessment in the Maintenance Process. Specific examples of inspection and monitoring techniques successfully used by utilities to detect degradation due to aging have been identified. The information also was sorted according to systems and components, including: Auxiliary Feedwater, Main Feedwater, High Pressure Injection for both BWRs and PWRs, Service Water, Instrument Air, and Emergency Diesel Generator Air Start Systems, and emergency diesel generators, electrical components such as switchgear, breakers, relays, and motor control centers, motor operated valves and check valves. This information was compared to insights gained from the Nuclear Plant Aging Research (NPAR) Program. Attributes of plant maintenance programs where the NRC inspectors felt that improvement was needed to properly address the aging issue also are discussed.

NUREG/CR-5830: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE MCGUIRE NUCLEAR POWER PLANT. BUMGARDNER,J.D.; LLOYD,R.C.; MOFFITT,N.E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1994. 36pp. 9406200324. PNL-7784. 79830:001.

In a study sponsored by the U.S. Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory has developed and applied a methodology for deriving plant-specific risk-based inspection guidance for the auxiliary feedwater (AFW) system at pressurized water reactors that have not undergone probabilistic risk assessment (PRA). This methodology uses existing PRA re-

sults 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. McGuire was selected as one of a series of plants for study. The product of this effort is a prioritized listing of AFW failures which have occurred at the plant and at other PWRs. This listing is intended for use by NRC inspectors in the preparation of inspection plans addressing AFW risk-important components at the McGuire plant.

NUREG/CR-5850: ANALYSIS OF LONG-TERM STATION BLACKOUT WITHOUT AUTOMATIC DEPRESSURIZATION AT PEACH BOTTOM USING MELCOR (VERSION 1.8). MADNI, I.K. Brookhaven National Laboratory. May 1994. 127pp. 9406210270. BNL-NUREG-52319. 79837:001.

This report documents the results from MELCOR calculations of the Long-Term Station Blackout Accident Sequence, with failure to depressurize the reactor vessel, at the Peach Bottom (BWR Mark I) plant, and presents comparisons with Source Term Code Package calculations of the same sequence. STCP has calculated the transient out to 13.5 hours after core uncover. Most of the MELCOR calculations presented have been carried out to between 15 and 16.7 hours after core uncover. The results include the release of source terms to the environment. The results of several sensitivity calculations with MELCOR are also presented, which explore the impact of varying user-input modeling and timestep control parameters on the accident progression and release of source terms to the environment. Most of the calculations documented here were performed in FY1990 using MELCOR Version 1.8BC. However, the appendices also document the results of more recent calculations performed in FY1991 using MELCOR versions 1.8CZ and 1.8DNX.

NUREG/CR-5861: CRACK-SPEED RELATIONS INFERRED FROM LARGE SINGLE-EDGE-NOTCHED SPECIMENS OF A 533 B STEEL. SCHWARTZ, C.W. Oak Ridge National Laboratory. July 1994. 32pp. 9408180237. ORNL SUB79-77789. 80595:312.

A relationship between instantaneous crack-tip velocity, a dynamic stress-intensity factor, $K(I)$, and temperature T for A 533 B steel is estimated using dynamic crack position vs time data measured in a series of very large-scale crack-arrest tests. The corresponding dynamic stress intensity vs time history and the dynamic-arrest toughness for each test are obtained from generation-mode elastodynamic analyses based on cubic polynomial fits to the discrete crack-position data points. Application mode elastodynamic analytical predictions based on the proposed $a-K(I)-T$ relation are within 7% of experimentally measured arrested crack lengths and within 50% of measured arrest times. These predictions represent significant improvements over results obtained using previous preliminary estimates of the $a-K(I)-T$ relation for A 533 B steel. The influence of nonlinear material behavior on the results is also evaluated.

NUREG/CR-5904: FUNCTIONAL ISSUES AND ENVIRONMENTAL QUALIFICATION OF DIGITAL PROTECTION SYSTEMS OF ADVANCED LIGHT-WATER NUCLEAR REACTORS. KORSAN, K.; CLARK, R.L.; WOOD, R.T. Oak Ridge National Laboratory. April 1994. 87pp. 9405310256. ORNL/TM-12164. 79588:251.

A study of significant "new" technologies proposed for use in safety-related instrumentation and controls (I&C) systems of advanced light-water reactors (ALWRs) was performed as part of the Qualification of Advanced Instrumentation and Control Systems project conducted for the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission (NRC). Templates showing digital protection systems of some ALWR designs and the effect of expected environmental stressors on system components were developed to illustrate functional and

qualification issues. The study also identified optical fiber systems as technologies that are relatively new to the nuclear power plant environment, and examined the failure modes and age-related degradation mechanisms associated with fiber optic cables and components. The data were then used to propose a methodology for identifying circumstances in which accelerated aging should be used in an equipment qualification program for "new" I&C technologies. An analysis of the licensee event report database over a 10-y period (1982-1991) performed under this study showed that the fraction of EMI/RFI-related protection system events is significant compared to traditionally recognized environmental stressors such as elevated temperature. The problem is likely to be even more significant for ALWR safety systems due to the increased use of microprocessor-based technology and software. Thus, it appears that while safety systems in ALWRs will have to be qualified to the same environment as current LWRs, EMI/RFI emissions and susceptibility criteria and guidelines specific to the nuclear power plant environment should be considered. Specific EMI/RFI requirements are addressed in a companion document, NUREG/CR-5941, "Electromagnetic and Radio-Frequency Interference in Safety-Critical I&C System."

NUREG/CR-5908 V01: ADVANCED HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE. General Evaluation Model, Technical Development, And Guideline Description. O'HARA, J.M. Brookhaven National Laboratory. July 1994. 140pp. 9408250026. BNL-NUREG-52333. 80660:115.

Advanced control rooms will use advanced human-system interface (HSI) technologies that may have significant implications for plant safety in that they will affect the operator's overall role in the system, the method of information presentation, and the ways in which operators interact with the system. The U.S. Nuclear Regulatory Commission (NRC) reviews the HSI aspects of control rooms to ensure that they are designed to good human factors engineering principles and that operator performance and reliability are appropriately supported to protect public health and safety. The principal guidance available to the NRC, however, was developed more than ten years ago, well before these technological changes. Accordingly, the human factors guidance needs to be updated to serve as the basis for NRC review of these advanced designs. The purpose of this project was to develop a general approach to advanced HSI review and the human factors guidelines to support NRC safety reviews of advanced systems. This two-volume report provides the results of the project. Volume 1 describes the development of the Advanced HSI Design Review Guideline (DRG) including (1) its theoretical and technical foundation, (2) a general model for the review of advanced HSIs, (3) guideline development in both hard-copy and computer-based versions, and (4) the tests and evaluations performed to develop and validate the DRG. Volume 1 also includes a discussion of the gaps in available guidance and a methodology for addressing them. Volume 2 provides the guideline to be used for advanced HSI review and the procedures for their use.

NUREG/CR-5908 V02: ADVANCED HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE. Evaluation Procedures And Guidelines For Human Factors Engineering Reviews. O'HARA, J.M.; BROWN, W.S. Brookhaven National Laboratory. BAKER, C.C.; et al. Carlow International, Inc. July 1994. 277pp. 9408250004. BNL-NUREG-52333. 80656:001.

See NUREG/CR-5908, V01 abstract.

NUREG/CR-5919: REPOSITORY OPERATIONAL CRITERIA COMPARATIVE ANALYSIS. HAGEMAN, J.P.; CHOWDHURY, A.H. Center for Nuclear Waste Regulatory Analyses. June 1994. 69pp. 9406290301. CNWRA 92-007. 80013:065.

The objective of the "Repository Operational Criteria (ROC) Feasibility Studies" (or ROC task) was to conduct comprehensive and integrated analyses of repository design, construction,

and operations criteria in 10 CFR Part 60 regulations considering the interfaces among the components of the regulations and impacts of any potential changes to those regulations. The ROC task addresses regulatory criteria and uncertainties related to the preclosure aspects of the geologic repository. Those parts of 10 CFR Part 60 that require routine guidance or minor changes to the rule were addressed in Hageman and Chowdhury, 1992. The ROC task shows a possible need for further regulatory clarity, by major changes to the rule, related to the design bases and siting of a geologic repository operations area and radiological emergency planning in order to assure defense-in-depth. The analyses, presented in this report, resulted in the development and refinement of regulatory concepts and their supporting rationale for recommendations for potential major changes to 10 CFR Part 60 regulations.

NUREG/CR-5935: SUMMARY OF WORK COMPLETED UNDER THE ENVIRONMENTAL AND DYNAMIC EQUIPMENT QUALIFICATION RESEARCH PROGRAM (EDQP). STEELE,R.; BRAMWELL,D.L.; WATKINS,J.C.; et al. EG&G Idaho, Inc. February 1994. 79pp. 9403140229. EGG-2686. 78477:071.

This report documents the results of the main projects undertaken under the Environmental and Dynamic Equipment Qualification Research Program (EDQP) sponsored by the U.S. Nuclear Regulatory Commission (NRC) under FIN A6322. Lasting from fiscal year 1983 to 1987, the program dealt with environmental and dynamic (including seismic) equipment qualification issues for mechanical and electromechanical components and systems used in nuclear power plants. The research results have since been used by both the NRC and industry. The program included seven major research projects that addressed the following issues: (a) containment purge and vent valves performing under design basis loss of coolant accident loads, (b) containment piping penetrations and isolation valves performing under seismic loadings and design basis and severe accident containment wall displacements, (c) shaft seals for primary coolant pumps performing under station blackout conditions, (d) electrical cabinet internals responding to in-structure generated motion (rattling), and (e) in situ piping and valves responding to seismic loadings. Another project investigating whether certain containment isolation valves will close under design basis conditions was also started under this program. This report includes eight main sections, each of which provides a brief description of one of the projects, a summary of the findings, and an overview of the application of the results. A bibliography lists the journal articles, papers, and reports that document the research.

NUREG/CR-5939: THE EFFECTS OF AGE ON NUCLEAR POWER PLANT CONTAINMENT COOLING SYSTEMS. LOFARO,R.; SUBUDHI,M.; TRAVIS,R.; et al. Brookhaven National Laboratory. April 1994. 160pp. 9405310236. BNL-NUREG-52345. 79589:042.

A study was performed to assess the effects of aging on the performance and availability of containment cooling systems in U.S. commercial nuclear power plants. This study is part of the Nuclear Plant Aging Research (NPAR) program sponsored by the U.S. Nuclear Regulatory Commission. The objectives of this program are to provide an understanding of the aging process and how it affects plant safety so that it can be properly managed. This is one of a number of studies performed under the NPAR program which provide a technical basis for the identification and evaluation of degradation caused by age. The effects of age were characterized for the containment cooling system by reviewing and analyzing failure data from national databases, as well as plant-specific data. The predominant failure causes and aging mechanisms were identified, along with the components that failed most frequently. Current inspection, surveillance, and monitoring practices were also examined. A containment cooling system unavailability analysis was performed to examine the potential effects of aging by increasing failure rates for selected components. A commonly found containment spray system design and a commonly found fan cooler system design were modeled. Parametric failure rates for those compo-

nents in each system that could be subject to aging were accounted for in the model to simulate the time-dependent effects of aging degradation, assuming no provisions are made to properly manage it. System unavailability as a function of increasing component failure rates was then calculated.

NUREG/CR-5941: TECHNICAL BASIS FOR EVALUATING ELECTROMAGNETIC AND RADIO-FREQUENCY INTERFERENCE IN SAFETY-RELATED I&C SYSTEMS. EWING,P.D.; KORSAH,K. Oak Ridge National Laboratory. April 1994. 44pp. 9405310198. ORNL/TM-12221. 79590:001.

This report discusses the development of the technical basis for the control of upsets and malfunctions in safety-related instrumentation and control (I&C) systems caused by electromagnetic and radio-frequency interference (EMI/RFI) and power surges. The research was performed at the Oak Ridge National Laboratory (ORNL) and was sponsored by the U.S. NRC Office of Nuclear Regulatory Research (RES). The motivation for research stems from the safety-related issues that need to be addressed with the application of advanced I&C systems to nuclear power plants. Development of the technical basis centered around establishing good engineering practices to ensure that sufficient levels of electromagnetic compatibility (EMC) are maintained between the nuclear power plant's electronic and electromechanical systems known to be the source(s) of EMI/RFI and power surges. First, good EMC design and installation practices need to be established to control the impact of interference sources on nearby circuits and systems. These EMC good practices include circuit layouts, terminations, filtering, grounding, bonding, shielding, and adequate physical separation. Second, an EMI/RFI test and evaluation program needs to be established to outline the tests to be performed, the associated test methods to be followed, and carefully formulated acceptance criteria based on the intended environment to ensure that the circuit or system under test meets the recommended guidelines. Third, a program needs to be developed to perform confirmatory tests and evaluate the surge withstand capability (SWC) and of I&C equipment connected to or installed in the vicinity of power circuits within the nuclear power plant. By following these three steps, the design and operability of safety-related I&C systems against EMI/RFI and power surges can be evaluated, acceptance criteria can be developed, and appropriate regulatory guidance can be provided.

NUREG/CR-5960: STEAM EXPLOSIONS: FUNDAMENTALS AND ENERGETIC BEHAVIOR. THEOFANOUS,T.G.; YUEN,W.W.; ANGELINI,S.; et al. California, Univ. of, Santa Barbara, CA. January 1994. 250pp. 9403140221. 78454:001.

This report presents the results of a multifaceted research effort in the field of steam explosions. The scope ranges from the fundamentals to assessing the energetics in applications relevant to Severe Accidents in Light Water Reactors. The consideration of fundamentals is built around two key ideas: the water depletion phenomenon during premixing and the microinteractions, including fragmentation kinetics, during propagation. The application to reactor conditions includes consideration of in-vessel steam explosions in PWRs and ex-vessel explosions in all five containment designs in current plants (in the USA). The report is structured in three parts, dealing with premixing, propagation, and energetics, respectively.

NUREG/CR-5963: CONTINUOUS AE CRACK MONITORING OF A DISSIMILAR METAL WELDMENT AT LIMERICK UNIT 1. HUTTON,P.H.; FRIESEL,M.A.; DAWSON,J.F. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1993. 100pp. 9402220168. PNL-8844. 78197:095.

Acoustic emission (AE) technology for continuous surveillance of a reactor component(s) to detect crack initiation and/or crack growth has been developed at Pacific Northwest Laboratory (PNL), operated by Battelle Memorial Institute, under support from the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (U.S. NRC-RES). The technology was vali-

dated off-reactor in several major tests, but it had not been validated by monitoring crack growth on an operating reactor system. A flaw indication was identified during normal inservice inspection of piping at Philadelphia Electric Company (PECO) Limerick Unit 1 reactor during the 1989 refueling outage. Evaluation of the flaw indication showed that it could remain in place during the subsequent fuel cycle without compromising safety. The existence of this flaw indication offered a long sought opportunity to validate AE surveillance to detect and evaluate crack growth during reactor operation. Through the cooperation and support of PECO and the U.S. NRC-RES, AE instrumentation was installed by BNW and PECO under PECO Mod. No. 043-002 to monitor the flaw indication during two complete fuel cycles. This report discusses the results obtained from the AE monitoring over the period May 1989 to March 1992 (both fuel cycles).

NUREG/CR-5965: MODELING FIELD SCALE UNSATURATED FLOW AND TRANSPORT PROCESSES. GELHAR, L.W.; CELIA, M.A.; MCLAUGHLIN, D. Massachusetts Institute of Technology, Cambridge, MA. August 1994. 82pp. 9409230273. 81006:001.

A stochastic theory describing unsaturated flow and contamination transport in naturally heterogeneous soils has been enhanced by adopting a more realistic characterization of soil variability. The enhanced theory is used to predict field-scale effective properties and variances of tension and moisture content. Applications illustrate the important effects of small-scale heterogeneity on large-scale anisotropy and hysteresis and demonstrate the feasibility of simulating two-dimensional flow systems at time and space scales of interest in radioactive waste disposal investigations. Numerical algorithms for predicting field scale unsaturated flow and contaminant transport have been improved by requiring them to respect fundamental physical principles such as mass conservation. These algorithms are able to provide realistic simulations of systems with very dry initial conditions and high degrees of heterogeneity. Numerical simulation of the simultaneous movement of water and air in unsaturated soils has demonstrated the importance of air pathways for contaminant transport. The stochastic flow and transport theory has been used to develop a systematic approach to performance assessment and site characterization. Prediction uncertainties have been quantified by considering the role of both natural heterogeneity and measurement error. Hypothesis-testing techniques have been used to determine whether model predictions are consistent with observed data.

NUREG/CR-5967: DEVELOPMENT AND APPLICATION OF DEGRADATION MODELING TO DEFINE MAINTENANCE PRACTICES. STOCK, D.; SAMANTA, P. Brookhaven National Laboratory. VESELY, W. Science Applications International Corp. (formerly Science Applications, Inc.). June 1994. 63pp. 9406290294. BNL-NUREG-52353. 80013:001.

This report presents the development and application of component degradation modeling to analyze degradation effects on reliability and to identify aspects of maintenance practices that mitigate degradation and aging effects. Using continuous time Markov approaches, a component degradation model is discussed that includes information about degradation and maintenance. The component model commonly used in probabilistic risk assessments is a simple case of this general model. The parameters used in the general model have engineering interpretations and can be estimated using data and engineering experience. The generation of equations for specific models, the solution of these equations, and a methodology for estimating the needed parameters are all discussed. Applications in this report show how these models can be used to quantitatively assess the benefits that are expected from maintaining a component, the effects of different maintenance efficiencies, the merits of different maintenance policies, and the interaction of surveillance test intervals with maintenance practices.

NUREG/CR-5973 R01: CODES AND STANDARDS AND OTHER GUIDANCE CITED IN REGULATORY DOCUMENTS. ANKRUM, A.R.; NICKOLAUS, J.R.; VINTHER, R.W.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1994. 398pp. 9409230271. PNL-8462. 81009:001.

As part of the U.S. Nuclear Regulatory Commission (NRC) Standard Review Plan Update and Development Program, Pacific Northwest Laboratory developed a listing of industry consensus codes and standards and other government and industry guidance referred to in regulatory documents. In addition to updating previous information, Revision 1 adds citations from the NRC Inspection Manual and the Improved Standard Technical Specifications. This listing identifies the version of the code or standard cited in the regulatory document, and the current version of the code or standard. It also provides a summary characterization of the nature of the citation. This listing was developed from electronic searches of the Code of Federal Regulations and the NRC's Bulletins, Information Notices, Circulars, Generic Letters, Policy Statements, Regulatory Guides, and the Standard Review Plan (NUREG-0800).

NUREG/CR-5985: EVALUATION OF COMPUTER-BASED ULTRASONIC INSERVICE INSPECTION SYSTEMS. HARRIS, R.V.; ANGEL, L.J.; DOCTOR, S.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1994. 126pp. 9404040063. PNL-8919. 78737:001.

This report presents the principles, practices, terminology, and technology of computer-based ultrasonic testing for inservice inspection (UT/ISI) of nuclear power plants, with extensive use of drawings, diagrams, and UT images. The presentation is technical but assumes limited specific knowledge of ultrasonics or computers. The report is divided into 9 sections covering conventional UT, computer-based UT, and evaluation methodology. Conventional UT topics include coordinate axes, scanning, instrument operation, RF and video signals, and A-, B-, and C-scans. Computer-based topics include sampling, digitization, signal analysis, image presentation, SAFT, ultrasonic holography, transducer arrays, and data interpretation. An evaluation methodology for computer-based UT/ISI systems is presented, including questions, detailed procedures, and test block designs. Brief evaluations of several computer-based UT/ISI systems are given; supplementary volumes will provide detailed evaluations of selected systems.

NUREG/CR-5990: THE EFFECTS OF SOLAR-GEOMAGNETICALLY INDUCED CURRENTS ON ELECTRICAL SYSTEMS IN NUCLEAR POWER STATIONS. SUBUDHI, M. Brookhaven National Laboratory. CARROLL, D.P. Florida, Univ. of, Gainesville, FL. KASTURI, S. MOS, Inc. January 1994. 196pp. 9402150350. BNL-NUREG-52359. 78123:019.

This report presents the results of a study to evaluate the potential effects of geomagnetically induced currents (GICs) caused by the solar disturbances on the in-plant electrical distribution system and equipment in nuclear power stations. The plant-specific electrical distribution system for a typical nuclear plant is modeled using the ElectroMagnetic Transient Program (EMTP). The computer model simulates online equipment and loads from the station transformer in the switchyard of the power station to the safety-buses at 120 volts to which all electronic devices are connected for plant monitoring. The analytical model of the plant's electrical distribution system is studied to identify the transient effects caused by the half-cycle saturation of the station transformers due to GIC. This study provides results of the voltage harmonics levels that have been noted at various electrical buses inside the plant. The emergency circuits appear to be more susceptible to high harmonics due to the normally light load conditions. In addition to steady-state analysis, this model was further analyzed simulating various plant transient conditions (e.g., loss of load or large motor start-up) occurring during GIC events. Detail models of the plant's protective relaying system employed in bus transfer application were included in this model to study the effects of the harmonic

distortion of the voltage input. Potential harmonic effects on the uninterruptable power system (UPS) are qualitatively discussed as well.

NUREG/CR-5994: EMERGENCY DIESEL GENERATOR: MAINTENANCE AND FAILURE UNAVAILABILITY, AND THEIR RISK IMPACTS. SAMANTA,P.; KIM,I.; URYASEV,S.; et al. Brookhaven National Laboratory. November 1994. 210pp. 9412130030. BNL-NUREG-52363. 82006:001.

Emergency Diesel Generators (EDGs) provide on-site emergency alternating current (ac) electric power for a nuclear plant in the event that all off-site power sources are lost. Existing regulations establish requirements for designing and testing of these on-site power sources to reduce to an acceptable level the probability of losing all ac power sources. Operating experience with EDGs has raised questions about their testing and maintenance to achieve the EDG reliability levels and the total EDG unavailability experienced (fraction of time EDG is out-of-service due to testing, maintenance, and failures). In this report, recent operating experience is used to assess EDG unavailability due to testing, maintenance, and failures during reactor power operation and during plant shutdown. Recent data show an improvement in EDG reliability, but an increase in EDG unavailability due to maintenance, a significant portion of which is due to routinely scheduled maintenances. Probabilistic safety assessments (PSAs) of selected nuclear power plants are used to assess the risk impact of EDG unavailability due to maintenance and failure during power operation, and during different stages of plant shutdown. The results of these risk analyses suggest qualitative insights for scheduling EDG maintenance that will have minimal impact on risk of operating nuclear power plants.

NUREG/CR-6042: PERSPECTIVES ON REACTOR SAFETY. HASKIN,F.E. New Mexico, Univ. of, Albuquerque, NM. CAMP,A.L. Sandia National Laboratories. March 1994. 589pp. 9404110335. SAND93-0971. 78820:001.

The U.S. Nuclear Regulatory Commission (NRC) maintains a technical training center at Chattanooga, Tennessee to provide appropriate training to both new and experienced NRC employees. This document describes a one-week course in reactor safety concepts. The course consists of five modules: (1) historical perspective; (2) accident sequences; (3) accident progression in the reactor vessel; (4) containment characteristics and design bases; and (5) source terms and offsite consequences. The course text is accompanied by slides and videos during the actual presentation of the course.

NUREG/CR-6044: EXPERIMENTS TO INVESTIGATE DIRECT CONTAINMENT HEATING PHENOMENA WITH SCALED MODELS OF THE ZION NUCLEAR POWER PLANT IN THE SURTSEY TEST FACILITY. ALLEN,M.D.; PILCH,M.M.; BLANCHAT,T.K.; et al. Sandia National Laboratories. May 1994. 213pp. 9406210288. SAND93-1049. 79836:001.

The Surtsey Facility at Sandia National Laboratories (SNL) is used to perform scaled experiments that simulate hypothetical high-pressure melt ejection (HPME) accidents in a nuclear power plant (NPP). These experiments are designed to investigate the effect of specific phenomena associated with direct containment heating (DCH) on the containment load, such as the effect of physical scale, prototypic subcompartment structures, water in the cavity, and hydrogen generation and combustion. In the Integral Effects Test (IET) series, 1:10 linear scale models of the Zion NPP structures were constructed in the Surtsey vessel. The RPV was modeled with a steel pressure vessel that had a hemispherical bottom head, which had a 4-cm hole in the bottom head that simulated the final ablated hole that would be formed by ejection of an instrument guide tube in a severe NPP accident. Iron/alumina/chromium thermite was used to simulate molten corium that would accumulate on the bottom head of an actual RPV. The chemically reactive melt simulant was ejected by high-pressure steam from the RPV model into the scaled reactor cavity. Debris was then entrained

through the instrument tunnel into the subcompartment structures and the upper dome of the simulated reactor containment building. The results of the IET experiments are given in this report.

NUREG/CR-6051: EFFECTS OF TENSILE LOADING ON UPPER SHELF FRACTURE TOUGHNESS. JOYCE,J.A.; LINK,R.E. U.S. Naval Academy, Annapolis, MD. March 1994. 129pp. 9405050401. 79165:001.

The elastic-plastic experiments described in this document investigated whether upper-shelf initiation toughness and J resistance curves are dependent on T-stress and Q. Upper-shelf J resistance curves (J(lc)) and tearing resistance (T(mat)) results were obtained for a range of applied constraint. The J-Q and J-T stress loci were developed and compared with the expectations of the O'Dowd and Shih and the Betegon and Hancock analyses. Constraint was varied by changing the crack length and by changing the loading mode from bending to predominantly tensile. The conclusion is that J(lc) is apparently not dependent on T-stress or Q but the material tearing resistance is, with the tearing modulus increasing as constraint increases.

NUREG/CR-6053: COMPARISON OF MACCS USERS CALCULATIONS FOR THE INTERNATIONAL COMPARISON EXERCISE ON PROBABILISTIC ACCIDENT CONSEQUENCE ASSESSMENT CODES. NEYMOTIN,L. Brookhaven National Laboratory. April 1994. 184pp. 9405310158. BNL-NUREG-52380. 79592:001.

Over the past several years, the OECD/NEA and CEC sponsored an international program intercomparing a group of six probabilistic consequence assessment (PCA) codes designed to simulate health and economic consequences of radioactive releases into atmosphere of radioactive materials following severe accidents at nuclear power plants (NPPs): ARANO (Finland), CONDOR (UK), COSYMA (CEC), LENA (Sweden), MACCS (USA), and OSCAAR (Japan). In parallel with this effort, two separate groups performed similar calculations using the MACCS and COSYMA codes. Results produced in the MACCS Users Group (Greece, Italy, Spain, and USA) calculations and their comparison are contained in the present report. Version 1.5.11.1 of the MACCS code was used for the calculations. Good agreement between the results produced in the four participating calculations has been reached, with the exception of the results related to the ingestion pathway dose predictions. The main reason for the scatter in those particular results is attributed to the lack of a straightforward implementation of the specifications for agricultural production and counter-measures criteria provided for the exercise. A significantly smaller scatter in predictions of other consequences was successfully explained by differences in meteorological files and weather sampling, grids, rain distance intervals, dispersion model options, and population distributions.

NUREG/CR-6063: INTRAVAL PHASE II MODEL TESTING AT THE LAS CRUCES TRENCH SITE. HILLS,R.G. New Mexico State Univ., Las Cruces, NM. WIERENGA,P.J. Arizona, Univ. of, Tucson, AZ. LUIS,S.; et al. Massachusetts Institute of Technology, Cambridge, MA. January 1994. 143pp. 9402150366. 78123:209.

Several field experiments have been performed by scientists from the University of Arizona and New Mexico State University at the Las Cruces Trench Site to provide data to test deterministic and stochastic models for water flow and solute transport. These experiments were performed in collaboration with INTRAVAL, an international effort toward validation of geosphere models for the transport of radionuclides. During Phase I of INTRAVAL, qualitative comparisons between experimental data and model predictions were made using contour plots of water contents and solute concentrations. Detailed quantitative comparisons were not made. To provide data for more rigorous model testing, a third Las Cruces Trench experiment was designed by scientists from the University of Arizona and New

Mexico State University. Modelers from the Center for Nuclear Waste Regulatory Analysis, Massachusetts Institute of Technology, New Mexico State University, Pacific Northwest Laboratory, and the University of Texas provided predictions of water flow and tritium transport to New Mexico State University for analysis. The corresponding models assumed soil characterizations ranging from uniform to deterministically heterogeneous to stochastic. This report presents detailed quantitative comparisons to field data.

NUREG/CR-6074 V01: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING. Technical Report On The Findings Of Task 1. October 1991 - April 1993. DORNES, E.; CUTSHALL, W.; GONZALEZ, M. Southwest Research Institute. April 1994. 102pp. 9405040118. 04-4448-Y1T1. 79103:003.

This report covers the Task 1 activities for the Sealed Source and Device Design Safety Test Program. The NRC and Agreement States evaluate radiation safety information and register the designs of products containing radioactive materials. SwRI has been contracted to perform an examination of the licensing information submitted by vendors. The intent of this investigation is to verify vendor testing and analysis and to determine where further testing is warranted to ensure the radiation safety of the sealed source and device designs. Results of interviews with users are presented and recommendations are made for tests to be developed during Task 2.

NUREG/CR-6075: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN ZION. PILCH, M.M. Sandia National Laboratories. YAN, H.; THEOFANOUS, T.G. California, Univ. of, Santa Barbara, CA. December 1994. 396pp. 9501190286. SAND93-1535. 82359:001.

This report is the first step in the resolution of the Direct Containment Heating (DCH) issue for the Zion Nuclear Power Plant using the Risk Oriented Accident Analysis Methodology (ROAAM). This report includes the definition of a probabilistic framework that decomposes the DCH problem into three probability density functions that reflect the most uncertain initial conditions (UO(2) mass, zirconium oxidation fraction, and steel mass). Uncertainties in the initial conditions are significant, but our quantification approach is based on establishing reasonable bounds that are not unnecessarily conservative. To this end, we also make use of the ROAAM ideas of enveloping scenarios and "splintering." Two causal relations (CRs) are used in this framework: CR1 is a model that calculates the peak pressure in the containment as a function of the initial conditions, and CR2 is a model that returns the fluency of containment failure as a function of pressure within the containment. Uncertainty in CR1 is accounted for by the use of two independently developed phenomenological models, the Convection Limited Containment Heating (CLCH) model and the Two-Cell Equilibrium (TCE) model, and by probabilistically distributing the key parameter in both, which is the ratio of the melt entrainment time to the system blowdown time constant. The two phenomenological models have been compared with an extensive database including recent integral simulations at two different physical scales. The containment load distributions do not intersect the containment strength (fragility) curve in any significant way, resulting in containment failure probabilities less than 10^{-3} for all scenarios considered. Sensitivity analyses did not show any areas of large sensitivity.

NUREG/CR-6075 S01: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN ZION. PILCH, M.M.; ALLEN, M.D. Sandia National Laboratories. KNUDSON, D.L.; et al. Idaho National Engineering Laboratory. December 1994. 450pp. 9501180334. SAND93-1535. 82340:001.

Supplement 1 of NUREG/CR-6075 brings to closure the DCH issue for the Zion plant. It includes the documentation of the peer review process for NUREG/CR-6075, the assessments of four new splinter scenarios defined in working group meetings, and modeling enhancements recommended by the working

groups. In the four new scenarios, consistency of the initial conditions has been implemented by using insights from systems-level codes. SCDAP/RELAP5 was used to analyze three short-term station blackout cases with different lead rates. In all three cases, the hot leg or surge line failed well before the lower head and thus the primary system depressurized to a point where DCH was no longer considered a threat. However, these calculations were continued to lower head failure in order to gain insights that were useful in establishing the initial and boundary conditions. The most useful insights are that the RCS pressure is low at vessel breach, metallic blockages in the core region do not melt and relocate into the lower plenum, and melting of upper plenum steel is correlated with hot leg failure. The SCDAP/RELAP output was used as input to CONTAIN to assess the containment conditions at vessel breach. The containment-side conditions predicted by CONTAIN are similar to those originally specified in NUREG/CR-6075.

NUREG/CR-6076: TR-EDB: TEST REACTOR EMBRITTLEMENT DATA BASE, VERSION 1. STALLMANN, F.W.; WANG, J.A.; KAM, F.B.K. Oak Ridge National Laboratory. January 1994. 119pp. 9403140311. ORNL/TM-12415. 78516:001.

The Test Reactor Embrittlement Data Base (TR-EDB) is a collection of results from irradiations in materials test reactors. It complements the Power Reactor Embrittlement Data Base (PR-EDB), whose data are restricted to the results from the analysis of surveillance capsules in commercial power reactors. The rationale behind this restriction was the assumption that the results of test reactor experiments may not be applicable to power reactors and could, therefore, be challenged if such data were included. For this very reason the embrittlement predictions in the Reg Guide 1.99, Rev. 2 were based exclusively on power reactor data. However, test reactor experiments are able to cover a much wider range of materials and irradiation conditions that are needed to explore more fully a variety of models for prediction of irradiation embrittlement. These data are also needed for the study of effects of annealing for life extension of reactor pressure vessels that are difficult to obtain from surveillance capsule results. The current data collection of the TR-EDB contains primarily Charpy test data, which are accompanied in most cases by tensile tests for the same irradiation conditions. Information is available for 1,230 different irradiated sets, 797 of which are from base material (plates and forgings), 378 from welds, and 55 from heat-affected-zone materials. The chemistries of the investigated materials span also a fairly wide range, particularly in the content of copper and nickel, which are considered the most important contributors to embrittlement sensitivity. Complete chemistry information is available for 1,095 of the 1,230 samples (after discarding the HAZ information). The architecture of the TR-EDB is fully compatible with that of the PR-EDB so that the data from both databases can be easily merged if desired. The data files are given in dBASE format and can be accessed with any personal computer using the DOS operating system. "User-friendly" utility programs have been written to investigate the radiation embrittlement using this data base. The utility programs are used to retrieve and select specific data, manipulate data, display data to the screen or printer, and to fit and plot Charpy impact data.

NUREG/CR-6077: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-6. OSBORNE, M.F.; LORENZ, R.A.; TRAVIS, J.R.; et al. Oak Ridge National Laboratory. March 1994. 67pp. 9404080092. ORNL/TM-12416. 78794:197.

Test VI-6 was the sixth test in the VI series conducted in the vertical furnace. The fuel specimen was a 15.2-cm-long section of a fuel rod from the BR3 reactor in Belgium. The fuel had experienced a burnup of ~ 42 MWd/kg, with inert gas release during irradiation of $\sim 2\%$. The fuel specimen was heated in an induction furnace at 2300 K for 60 min, initially in hydrogen, then in a steam atmosphere. The released fission products were collected in three sequentially operated collection trains

designed to facilitate sampling and analysis. The fission product inventories in the fuel were measured directly by gamma-ray spectrometry, where possible, and were calculated by ORIGEN2. Integral releases were 75% for (85)Kr, 67% for (129)I, 64% for (125)Sb, 80% for both (134)Cs and (137)Cs, 14% for (154)Eu, 63% for Te, 32% for Ba, 13% for Mo, and 5.8% for Sr. Of the totals released from the fuel, 43% of the Cs, 32% of the Sb, and 98% of the Eu were deposited in the outlet end of the furnace. During the heatup in hydrogen, the Zircaloy cladding melted, ran down, and reacted with some of the UO₂ and fission products, especially Te and Sb. The total mass released from the furnace to the collection system, including fission products, fuel, and structural materials, was 0.57 g, almost equally divided between thermal gradient tubes and filters. The release behaviors for the most volatile elements, Kr and Cs, were in good agreement with the ORNL Diffusion Model.

NUREG/CR-6086: SELECTED FAULT TESTING OF ELECTRONIC ISOLATION DEVICES USED IN NUCLEAR POWER PLANT OPERATION. VILLARAN,M.; HILLMAN,K.; TAYLOR,J.; et al. Brookhaven National Laboratory. May 1994. 126pp. 9405310260. BNL-NUREG-52385. 79588:141.

Electronic isolation devices are used in nuclear power plants to provide electrical separation between safety and non-safety circuits and systems. Major fault testing in an earlier program indicated that some energy may pass through an isolation device when a fault at the maximum credible potential is applied in the transverse mode to its output terminals. During subsequent field qualification testing of isolators, concerns were raised that the worst case fault, that is, the maximum credible fault (MCF), may not occur with a fault at the maximum credible potential, but rather at some lower potential. The present test program investigates whether problems can arise when fault levels up to the MCF potential are applied to the output terminals of an isolator. The fault energy passed through an isolated device during a fault was measured to determine whether the levels are great enough to potentially damage or degrade performance of equipment on the input (Class 1E) side of the isolator.

NUREG/CR-6087: THE EFFECTS OF AGING ON BOILING WATER REACTOR CORE ISOLATION COOLING SYSTEMS. LEE,B.S. Brookhaven National Laboratory. October 1994. 218pp. 9411160020. BNL-NUREG-52390. 81756:083.

A study was performed to assess the effects of aging on the Reactor Core Isolation Cooling (RCIC) system in commercial Boiling Water Reactors (BWRs). This study is part of the Nuclear Plant Aging Research (NPAR) program sponsored by the U.S. Nuclear Regulatory Commission. The objectives of this program are to provide an understanding of the aging process and how it affects plant safety so that it can be properly managed. This is one of a number of studies performed under the NPAR program which provide a technical basis for the identification and evaluation of degradation caused by age. The failure data from national databases, as well as plant specific data were reviewed and analyzed to understand the effects of aging on the RCIC system. This analysis identified important components that should receive the highest priority in terms of aging management. The aging characterization provided information on the effects of aging on component failure frequency, failure modes, and failure causes. Current inspection, surveillance, and monitoring practices were also reviewed.

NUREG/CR-6088: SUMMARY OF 1991-1992 MISADMINISTRATION EVENT INVESTIGATIONS. OSTROM,L.T.; LEAHY,T.J.; NOVAK,S.D. EG&G Idaho, Inc. March 1994. 61pp. 9404040060. EGG-2707. 78736:193.

Investigation teams composed of representatives of the Idaho National Engineering Laboratory (INEL), the U.S. Nuclear Regulatory Commission (NRC), and subcontractors investigated and analyzed seven misadministration events selected by the NRC concerning medical radioisotopes. Each team was led by an INEL member and, depending on the nature of the event, in-

cluded three or more team members with appropriate expertise in radiation oncology, medical physics, nuclear medicine technology, risk analysis, and human factors. The investigations focused on causes of the event, consequences, mitigating actions, and corrective actions. The investigation produced seven major findings: 1) many misadministrations occurred primarily because procedures did not exist or because existing procedures that were not sufficiently detailed, comprehensive, specific, or clearly written; 2) although the NRC's quality management (QM) rule can prevent many misadministrations, most licensees in this study had not effectively implemented their QM programs; 3) the lack of substantial, direct involvement by radiation safety officers and authorized users was often a direct cause of misadministration; 4) a change in routine or the advent of a unique condition often predisposed misadministration; 5) hardware failures, though rare, had severe consequences, particularly when operating procedures, staff training, or other factors were not well implemented; 6) licensees' corrective actions were often narrow in focus; 7) the licensees lacked systematic methods for detecting and mitigating a misadministration once an error occurred.

NUREG/CR-6092: RISK ASSESSMENT FOR THE INTENTIONAL DEPRESSURIZATION STRATEGY IN PWRs. DINGMAN,S.E. Sandia National Laboratories. March 1994. 94pp. 9405040222. SAND93-1737. 79108:001.

An accident management strategy has been proposed in which the reactor coolant system is intentionally depressurized during an accident. The aim is to reduce the containment pressurization that would result from high pressure ejection of molten debris at vessel breach. Probabilistic risk assessment (PRA) methods were used to evaluate this strategy for the Surry nuclear power plant. Sensitivity studies were conducted using event trees that were developed for the NUREG-1150 study. It was found that depressurization (intentional or unintentional) had minimal impact on the containment failure probability at vessel breach for Surry because the containment loads assessed for NUREG-1150 were not a great threat to the containment survivability. An updated evaluation of the impact of intentional depressurization on the probability of having a high pressure melt ejection was then made that reflected analyses that have been performed since NUREG-1150 was completed. The updated evaluation confirmed the sensitivity study conclusions that intentional depressurization has minimal impact on the probability of a high pressure melt ejection. The updated evaluation did show a slight benefit from depressurization because depressurization delayed core melting, which led to a higher probability of recovering emergency core coolant injection, thereby arresting the core damage.

NUREG/CR-6093: AN ANALYSIS OF OPERATIONAL EXPERIENCE DURING LOW POWER AND SHUTDOWN AND A PLAN FOR ADDRESSING HUMAN RELIABILITY ASSESSMENT ISSUES. BARRIERE,M.; LUCKAS,W. Brookhaven National Laboratory. WHITEHEAD,D.W.; et al. Sandia National Laboratories. June 1994. 200pp. 9408030127. BNL-NUREG-52388. 80431:107.

Recent nuclear power plant events (e.g. Chernobyl, Diablo Canyon, and Vogtle) and U.S. Nuclear Regulatory Commission (NRC) reports (e.g. NUREG-1449) have led to concerns regarding human reliability during low power and shutdown (LP&S) conditions and limitations of human reliability analysis (HRA) methodologies in adequately representing the LP&S environment. As a result of these concerns, the NRC initiated two parallel research projects to assess the influence of LP&S conditions on human reliability through an analysis of operational experience at pressurized water reactors (PWRs) and boiling water reactors (BWRs). These research projects, performed by Brookhaven National Laboratory for PWRs, and Sandia National Laboratories for BWRs, identified unique aspects of human performance during LP&S conditions and provided a program plan for research and development necessary to improve existing

28 Main Citations and Abstracts

HRA methodologies. This report documents the results of the analysis of LP&S operating experience and describes the improved HRA program plan.

NUREG/CR-6094: CALCULATIONS IN SUPPORT OF A POTENTIAL DEFINITION OF LARGE RELEASE. HANSON, A.L.; DAVIS, R.E.; MUBAYI, V. Brookhaven National Laboratory. May 1994. 165pp. 9406200323. BNL-NUREG-52387. 79830:037.

The Nuclear Regulatory Commission has stated a hierarchy of safety goals with the qualitative safety goals as Level I of the hierarchy, backed up by the quantitative health objectives as Level II and the contemplated large release guideline as Level III. The large release guideline has been stated in qualitative terms as a magnitude of release of the core inventory whose frequency should not exceed 10(-6) per reactor year. However the Commission did not provide a quantitative specification of a large release. One recommendation was made that a large release be defined as having the potential for causing an off-site early fatality within one mile of the site to the average individual. This report focuses on an examination of releases which have a potential to lead to one prompt fatality. The basic information required to set up the calculations was derived from the simplified source terms which were obtained from approximations of the NUREG-1150 source terms. Since the calculation of consequences is affected by a large number of assumptions, a generic site with a population density and meteorology was specified. At this site, various emergency responses (including no response) were assumed based on information derived from earlier studies. For each of the emergency response assumptions, a set of calculations were performed with the simplified source terms. These included adjustments to the source terms, such as the timing of the release, the core inventory, and the release fractions of different radionuclides, to have a potential to lead to one mean prompt fatality in each case. Each of the source terms, so defined, has the potential to be a candidate for a large release.

NUREG/CR-6095: AGING, LOSS-OF-COOLANT ACCIDENT (LOCA), AND HIGH POTENTIAL TESTING OF DAMAGED CABLES. VIGIL, R.A. Science & Engineering Associates, Inc. JACOBUS, M.J. Sandia National Laboratories. April 1994. 74pp. 9405310297. SAND93-1803. 79585:231.

Experiments were conducted to assess the effects of high potential testing of cables and to assess the survivability of aged and damaged cables under loss-of-coolant accident (LOCA) conditions. High potential testing at 240 Vdc/mil on undamaged cables suggested that no damage was incurred on the selected virgin cables. During aging and LOCA testing, Okonite ethylene propylene rubber (EPR) cables with a bonded jacket experienced unexpected failures. The failures appear to be primarily related to the level of thermal aging and the presence of a bonded jacket that ages more rapidly than the insulation. For Brand Rex cross linked polyolefin (XLPO) cables, the results suggest that 7 mils of insulation remaining should give the cables a high probability of surviving accident exposure following aging. The voltage necessary to detect when 7 mils of insulation remain on unaged Brand Rex cables is approximately 35 kVdc. This voltage level would almost certainly be unacceptable to a utility for use as a damage assessment tool. However, additional tests indicated that a 35 kVdc voltage application would not damage virgin Brand Rex cables when tested in water. Although two Rockbestos silicone rubber cables failed during the accident test, no correlation between failures and level of damage was apparent.

NUREG/CR-6102: VALIDATION OF THE SCALE BROAD STRUCTURE 44-GROUP ENDF/B-Y CROSS-SECTION LIBRARY FOR USE IN CRITICALITY SAFETY ANALYSES. DEHART, M.D.; BOWMAN, S.M. Oak Ridge National Laboratory. September 1994. 147pp. 9411080072. ORNL/TM-12460. 81649:116.

This report documents the validation of the recently developed 44-group ENDF/B-V based cross-section library

(44GROUPNDF5). This cross-section set has been tested against its parent 238-group fine structure library (238GROUPNDF5) using a set of 33 benchmark problems in order to demonstrate that the collapsed set was an acceptable representation of 238GROUPNDF5. Validation of the library within the SCALE system was based on a comparison of calculated values of $k(\text{eff})$ with that of 93 experiments. The experiments primarily consisted of various configurations of light-water-reactor-type fuel representative of transportation and storage conditions. Additional experiments were included to allow comparison with results obtained in earlier validation of the 27GROUPNDF4 library. Results show that the broad 44-group structure is an acceptable representation of its parent 238-group library for thermal as well as hard fast spectrum systems. Accurate broad-group analyses of intermediate spectrum systems will require either a more detailed group structure in this energy range or a more appropriate collapsing spectrum. Further, validation calculations indicate that the 44-group library is an accurate tool in the prediction of criticality for arrays of light-water-reactor-type fuel assemblies as would be encountered in fresh or spent fuel transportation or storage environments.

NUREG/CR-6103: PRIORITIZATION OF REACTOR CONTROL COMPONENTS SUSCEPTIBLE TO FIRE DAMAGE AS A CONSEQUENCE OF AGING. LOWRY, W.; VIRGIL, R. Science & Engineering Associates, Inc. NOWLEN, S. Sandia National Laboratories. January 1994. 45pp. 9402150309. SAND93-7107. 78122:333.

The Fire Vulnerability of Aged Electrical Components Test Program is to identify and assess issues of plant aging that could lead to an increase in nuclear power plant risk because of fires. Historical component data and prior analyses are used to prioritize a list of components with respect to aging and fire vulnerability and the consequences of their failure on plant safety systems. The component list emphasizes safety system control components, but excludes cables, large equipment, and devices encompassed in the Equipment Qualification (EQ) program. The test program selected components identified in a utility survey and developed test and fire conditions necessary to maximize the effectiveness of the test program. Fire damage considerations were limited to purely thermal effects.

NUREG/CR-6104: SHEAR WALL ULTIMATE DRIFT LIMITS. DUFFEY, T.A. Consulting Engineer. GOLDMAN, A.; FARRAR, C.R. Los Alamos National Laboratory. April 1994. 116pp. 9405310177. LA-12649-MS. 79590:111.

Drift limits for reinforced-concrete shear walls are investigated by reviewing the open literature for appropriate experimental data. Drift values at ultimate load are determined for walls with aspect ratios ranging up to a maximum of 3.53 and undergoing different types of lateral loading (cyclic static, monotonic static, and dynamic). Based on the geometry of actual nuclear power plant structures exclusive of containment and concerns regarding their response during seismic (i.e., cyclic) loading, data are obtained from pertinent references for which the wall aspect ratio is less than or equal to approximately 1, and for which testing is cyclic in nature (typically displacement controlled). In particular, lateral deflections at ultimate load, and at points in the softening region beyond ultimate for which the load has dropped to 90, 80, 70, 60, and 50 percent of its ultimate value, are obtained and converted to drift information. The statistical nature of the data is also investigated. These data are shown to be lognormally distributed, and an analysis of variance is performed. The use of these statistics to estimate Probability of Failure for a shear wall structure is illustrated.

NUREG/CR-6105: HUMAN FACTORS ENGINEERING GUIDANCE FOR THE REVIEW OF ADVANCED ALARM SYSTEMS. O'HARA, J.M.; BROWN, W.S.; HIGGINS, J.C. Brookhaven National Laboratory. September 1994. 124pp. 9411080068. BNL-NUREG-52391. 81649:001.

This report provides guidance to support the review of the human factors aspects of advanced alarm system designs in nuclear power plants. The report is organized into three major sections. The first section describes the methodology and criteria that were used to develop the design review guidelines. Also included is a description of the scope, organization, and format of the guidelines. The second section provides a systematic review procedure in which important characteristics of the alarm system are identified, described, and evaluated. The third section provides the detailed review guidelines. The review guidelines are organized according to important characteristics of the alarm system including: alarm definition; alarm processing and reduction; alarm prioritization and availability; display; control; automated, dynamic, and modifiable characteristics; reliability, test, maintenance, and failure indication; alarm response procedures; and control-display integration and layout.

NUREG/CR-6107: SUMMARY OF MELCOR 1.8.2 CALCULATIONS FOR THREE LOCA SEQUENCES (AG,S2D & S3D) AT THE SURRY PLANT. KMETYK,L.N. Sandia National Laboratories. SMITH,L. Geo-Centers, Inc. March 1994. 200pp. 9404080073. SAND93-2042. 78794:001.

Activities involving regulatory implementation of updated source term information were pursued. These activities include the identification of the source term, the identification of the chemical form of iodine in the source term, and the timing of the source term's entrance into containment. These activities are intended to support a more realistic source term for licensing nuclear power plants than the current TID-14844 source term and current licensing assumptions. MELCOR calculations were performed to support the technical basis for the updated source term. This report presents the results from three MELCOR calculations of nuclear power plant accident sequences and presents comparisons with Source Term Code Package (STCP) calculations for the same sequences. The three low-pressure sequences were analyzed to identify the materials which enter containment (source terms) and are available for release to the environment, and to obtain timing of sequence events. The source terms include fission products and other materials such as those generated by core-concrete interactions. All three calculations, for both MELCOR and STCP, analyzed the Surry plant, a pressurized water reactor (PWR) with a subatmospheric containment design.

NUREG/CR-6112 DRF FC: IMPACT OF REDUCED DOSE LIMITS ON NRC LICENSED ACTIVITIES. Major Issues In The Implementation Of ICRP/NCRP Dose Limit Recommendations.Draft Report For Comment. MEINHOLD,C.B. Brookhaven National Laboratory. January 1994. 75pp. 9402220146. BNL-NUREG-52394. 78191:001.

This report summarizes information required to estimate, at least in qualitative terms, the potential impacts of reducing occupational dose limits below those given in 10 CFR 20 (Revised). The data from a questionnaire developed for this project and data from existing surveys were used to estimate the impact of three dose limit options; 10 mSv yr⁻¹, 20 mSv yr⁻¹, and a combination of an annual limit of 50 mSv yr⁻¹ coupled with a cumulative limit in rem equal to age in years. The overall conclusions of the study are: (1) Although 10 mSv yr⁻¹ is a reasonable limit for many licensees, such a limit could be extraordinarily difficult and potentially destructive to some, (2) Twenty mSv yr⁻¹ as a limit is possible for some of the latter groups, but for others it would prove difficult, (3) Fifty mSv yr⁻¹ and age in 10's of mSv would appear acceptable both in terms of the related lifetime risk of cancer and severe genetic effects to the most highly exposed and in terms of practicality of operation. This acceptability in some segments of the industry is based on the adoption of a "grandfather clause" for those exceeding the cumulative limit.

NUREG/CR-6114 V03: PERFORMANCE ASSESSMENT OF A HYPOTHETICAL LOW-LEVEL WASTE FACILITY.Groundwater Flow And Transport Simulation. TALBOTT,M.E.; GELHAR,L.W. Massachusetts Institute of Technology, Cambridge, MA. May 1994. 82pp. 9406200318. 79830:203.

Stochastic subsurface hydrologic theory is applied to data for a hypothetical low-level waste site to demonstrate the features of the hydraulic parameter estimation process, as developed by Gelhar and others. Effective values of hydraulic conductivity, macrodispersivity, and macrodispersivity enhancement are estimated from the data in this manner. A two-dimensional saturated flow and transport finite-element computer code is used to model the site. Four different isotope inputs and two types of input configurations contribute to an evaluation of model sensitivities. These sensitivities of the mean concentrations and the uncertainties around the mean are explored using an analytical model as an example. Results indicate that the spatial heterogeneity of isotope sorption, through its contribution to longitudinal dispersivity enhancement, has a large effect on the magnitude of concentration predictions, especially for isotopes with short half-lives in comparison to their retarded mean travel times. This observation indicates the need for accurate site data measurements that compliment the parameter estimation process. A comparison of simplified analytical screening models with the numerical model predictions shows that the analytical models tend to underestimate concentration levels at low times, potentially as a result of oversimplification of the flow field. Future models could address aspects that are neglected in this report, such as three-dimensionality or unsaturated flow and transport.

NUREG/CR-6116 V01: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Technical Reference Manual. RUSSELL,K.D.; ATWOOD,C.L.; GALYEAN,W.J.; et al. EG&G Idaho, Inc. July 1994. 131pp. 9408120223. EGG-2716. 80517:001.

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several micro-computer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. This volume provides information on the principles used in the construction and operation of Version 5.0 of the Integrated Reliability and Risk Analysis System (IRRAS) and the System Analysis and Risk Assessment (SARA) system. It summarizes the fundamental mathematical concepts of sets and logic, fault trees, and probability. This volume then describes the algorithms that these programs use to construct a fault tree and to obtain the minimal cut sets. It gives the formulas used to obtain the probability of the top event from the minimal cut sets, and the formulas for probabilities that are appropriate under various assumptions concerning reparability and mission time. It defines the measures of basic event importance that these programs can calculate. This volume gives an overview of uncertainty analysis using simple Monte Carlo sampling or Latin Hypercube sampling, and states the algorithms used by these programs to generate random-basic event probabilities from various distributions. Further references are given, and a detailed example of the reduction and quantification of a simple fault tree is provided in an appendix.

NUREG/CR-6116 V02: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Integrated Reliability And Risk Analysis System (IRRAS) Reference Manual. RUSSELL,K.D.; KVARFORDT,K.J.; SKINNER,N.L.; et al. EG&G Idaho, Inc. July 1994. 422pp. 9408150043. EGG-2716. 80542:032.

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several micro-computer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. 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 to report generation. Version 5.0 of IRRAS provides the same capabilities as earlier versions and adds the ability to perform location transformations, seismic analysis, and provides enhancements to the user interface as well as improved algorithm performance. Additionally, version 5.0 contains new alphanumeric fault tree and event tree editors, and a powerful set of macro-based rule editors. These editors are used for event tree rules, recovery rules, and end state partitioning.

NUREG/CR-6116 V03: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Integrated Reliability And Risk Analysis System (IRRAS) Tutorial Manual. VANHORN, R.L.; RUSSELL, K.D.; SKINNER, N.L. EG&G Idaho, Inc. July 1994. 174pp. 9408150049. EGG-2716. 80544:106.

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several microcomputer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. This volume is the tutorial manual for the Integrated Reliability and Risk Analysis System (IRRAS) Version 5.0, 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. A series of lessons is provided that guides the user through basic steps common to most analyses performed with IRRAS. The tutorial is divided into two major sections: basic and additional features. The basic section contains lessons that lead the student through development of a very simple problem in IRRAS, highlighting the program's most basic features. The additional features section contains lessons that expand on basic analysis features of IRRAS 5.0.

NUREG/CR-6116 V04: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Systems Analysis And Risk Assessment (SARA) Reference Manual. RUSSELL, K.D.; KVARFORDT, K.J.; SKINNER, N.L.; et al. EG&G Idaho, Inc. July 1994. 291pp. 9408150057. EGG-2716. 80541:105.

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several microcomputer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. This volume is the reference manual for the Systems Analysis and Risk Assessment (SARA) System Version 5.0, a microcomputer-based system used to analyze the safety issues of a "family" [i.e., a power plant, a manufacturing facility, any facility on which a probabilistic risk assessment (PRA) might be performed]. The SARA database contains PRA data primarily for the dominant accident sequences of a family and descriptive information about the family including event trees, fault trees, and system model diagrams. To simulate changes to family systems, SARA users change the failure rates of initiating and basic events and/or modify the structure of the cut sets that make up the event trees, fault trees, and systems. The user then evaluates the effects of these changes through the recalculation of the resultant accident sequence probabilities and importance measures. The results are displayed in tables and graphs that may be printed for reports.

NUREG/CR-6116 V05: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Systems Analysis And Risk Assessment (SARA) Tutorial Manual. SATTISON, M.B.; RUSSELL, K.D.; SKINNER, N.L. EG&G Idaho, Inc. July 1994. 110pp. 9408150072. EGG-2716. 80544:001.

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several microcomputer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. This volume is the tutorial manual for the Systems Analysis and Risk Assessment (SARA) System Version 5.0, a microcomputer-based system used to analyze the safety issues of a "family" [i.e., a power plant, a manufacturing facility, any facility on which a probabilistic risk assessment (PRA) might be performed]. A series of lessons is provided that guides the user through some basic steps common to most analyses performed with SARA. The example problems presented in the lessons build on one another, and in combination, lead the user through all aspects of SARA sensitivity analysis capabilities.

NUREG/CR-6116 V07: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Fault Tree, Event Tree, and Piping & Instrumentation Diagram (FEP) Editors Reference Manual. MCKAY, M.K.; SKINNER, N.L.; WOOD, S.T. EG&G Idaho, Inc. July 1994. 147pp. 9408150081. EGG-2716. 80537:144.

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several microcomputer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. The Fault Tree, Event Tree, and Piping & Instrumentation Diagram (FEP) editors allow the user to graphically build and edit fault trees, event trees, and piping & instrumentation diagrams (P&IDs). The software is designed to enable the independent use of the graphical-based editors found in the Integrated Reliability and Risk Assessment System (IRRAS). FEP is comprised of three separate editors (Fault Tree, Event Tree, and Piping & Instrumentation Diagram) and a utility module. This reference manual provides a screen-by-screen guide of the entire FEP System.

NUREG/CR-6116 V08: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Models And Results Database (MAR-D) Reference Manual. RUSSELL, K.D.; SKINNER, N.L. EG&G Idaho, Inc. July 1994. 134pp. 9408150094. EGG-2716. 80544:280.

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several microcomputer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. The primary function of MAR-D is to create a data repository for completed PRAs and Individual Plant Examinations (IPEs) by providing input, conversion, and output capabilities for data used by IRRAS, SARA, SETS, and FRANTIC software. As probabilistic risk assessments and individual plant examinations are submitted to the NRC for review, MAR-D can be used to convert the models and results from the study for use with IRRAS and SARA. Then, these data can be easily accessed by future studies and will be in a form that will enhance the analysis process. This reference manual provides an overview of the functions available within MAR-D and step-by-step operating instructions.

NUREG/CR-6120: CONTROLLED FIELD STUDY FOR VALIDATION OF VADOSE ZONE TRANSPORT MODELS. WIERENG, P.J.; WARRICK, A.W.; et al. Arizona, Univ. of, Tucson, AZ. HILLS, R.G. New Mexico State Univ., Las Cruces, NM. August 1994. 27pp. 9409060223. 80785:001.

Prediction of radionuclide migration through soil and groundwater requires models which have been tested under a variety of conditions. Unfortunately, many of the existing models have

not been tested in the field, partly because such testing requires accurate and representative data. This report provides the design of a large scale field experiment representative, in terms of surface area and depth of vadose zone, of an actual disposal area. Experiments are proposed which will yield documented data, of sufficient scale, to allow testing of a variety of models including effective media stochastic models and deterministic models. Details of the methodology and procedures to be used in the experiment are presented.

NUREG/CR-6121: COMPONENT EVALUATION FOR INTERSYSTEM LOSS-OF-COOLANT ACCIDENTS IN ADVANCED LIGHT WATER REACTORS. WARE, A.G. EG&G Idaho, Inc. July 1994. 113pp. 9408250010. EGG-2717. 80658:177.

Using the methodology outlined in NUREG/CR-5603 this report evaluates (on a probabilistic basis) design rules for components in ALWRs that could be subjected to intersystem loss-of-coolant accidents (ISLOCAs). The methodology is intended for piping elements, flange connections, on-line pumps and valves, and heat exchangers. The NRC has directed that the design rules be evaluated for BWR pressures of 7.04 MPa (1025 psig), PWR pressures of 15.4 MPa (2235 psig), and 177 degrees C (350 degrees F), and has established a goal of 90% probability that system rupture will not occur during an ISLOCA event. The results of the calculations in this report show that components designed for a pressure of 0.4 of the reactor coolant system operating pressure will satisfy the NRC survival goal in most cases. Specific recommendations for component strengths for BWR and PWR applications are made in the report. A peer review panel of nationally recognized experts was selected to review and critique the initial results of this program.

NUREG/CR-6122: STAFFING DECISION PROCESSES AND ISSUES. Case Studies Of Seven U.S. Nuclear Power Plants. MELBER, B.; ROUSSEL, A.; BAKER, K.; et al. Battelle Human Affairs Research Centers. March 1994. 53pp. 9404040039. PNL-8428. 78736:001.

The objective of this report is to identify how decisions are made regarding staffing levels and positions for U.S. nuclear power plants. In this report, a framework is provided for understanding the major forces driving staffing and the implications of staffing decisions for plant safety. The focus of this report is on driving forces that have led to changes in staffing levels and to the establishment of new positions between the mid-1980s and the early 1990s. Processes used at utilities and nuclear power plants to make and implement these staffing decisions are also discussed in the report. While general trends affecting the plant as a whole are presented, the major emphasis of this report is on staffing changes and practices in the operations department, including the operations shift crew. The findings in this report are based on interviews conducted at seven nuclear power plants and their parent utilities. A discussion of the key findings is followed by a summary of the implications of staffing issues for plant safety.

NUREG/CR-6123: AN INTERNATIONAL COMPARISON OF COMMERCIAL NUCLEAR POWER PLANT STAFFING REGULATIONS AND PRACTICE. 1980-1990. MELBER, B.; HAUTH, J.; TERRILL, E.; et al. Battelle Human Affairs Research Centers. March 1994. 70pp. 9404040052. PNL-8564. 78736:123.

A review of nuclear power plant staffing regulatory and industry practices in Canada, France, Germany, Japan, Sweden, and the United Kingdom is presented in this report. International trends in staffing regulatory approaches, industry practices, and issues of concern that have potential relevance for the continued development of nuclear power plant staffing policy in the United States are highlighted. The 1980s were marked by significant growth in nuclear power operations internationally; however, growth of nuclear power is not expected to continue in the 1990s except in France and Japan. A continuum of regulatory approaches was identified, ranging from prescriptive regulations applied to all licensees to staffing requirements agreed to in the plant operating licenses. Most of the changes observed in staff-

ing regulations and practices in the early 1980s were made in response to the accident at the Three Mile Island Unit 2 nuclear power plant in 1979. These changes included widespread issuance of new operator and licensing requirements and the establishment of national training centers. A period of relative stability followed these changes. Changes in the latter half of the 1980s focused on continuing improvements and additions to training curricula and methods, most notably increased reliance on simulator training.

NUREG/CR-6126: COGNITIVE SKILL TRAINING FOR NUCLEAR POWER PLANT OPERATIONAL DECISION-MAKING. MUMAW, R.J.; SWATZLER, D.; ROTH, E.M.; et al. Westinghouse Electric Corp. June 1994. 200pp. 9407010300. 80031:240.

Training for operator and other technical positions in the commercial nuclear power industry traditionally has focused on mastery of the formal procedures used to control plant systems and processes. However, decision-making tasks required of nuclear power plant operators involve cognitive skills (e.g., situation assessment, planning). Cognitive skills are needed in situations where formal procedures may not exist or may not be as prescriptive, as is the case in severe accident management (SAM). The Westinghouse research team investigated the potential cognitive demands of SAM on the control room operators and Technical Support Center staff who would be most involved in the selection and execution of severe accident control actions. A model of decision making, organized around six general cognitive processes, was developed to identify the types of cognitive skills that may be needed for effective performance. Also, twelve SAM scenarios were developed to reveal specific decision-making difficulties. Following the identification of relevant cognitive skills, 19 approaches for training individual and team cognitive skills were identified. A review of these approaches resulted in the identification of general characteristics that are important to effective training of cognitive skills.

NUREG/CR-6127: THE EFFECTS OF STRESS ON NUCLEAR POWER PLANT OPERATIONAL DECISION MAKING AND TRAINING APPROACHES TO REDUCE STRESS EFFECTS. MUMAW, R.J. Westinghouse Electric Corp. August 1994. 43pp. 9408250017. 80656:278.

Operational personnel may be exposed to significantly levels of stress during unexpected changes in plant state and plant emergencies. The decision making that identifies operational actions, which is strongly determined by procedures, may be affected by stress, and performance may be impaired. This report analyzes potential effects of stress in nuclear power plant (NPP) settings, especially in the context of severe accident management (SAM). First, potential sources of stress in the NPP setting are identified. This analysis is followed by a review of the ways in which stress is likely to affect performance, with an emphasis on performance of cognitive skills that are linked to operational decision making. Finally, potential training approaches for reducing or eliminating stress effects are identified. Several training approaches have the potential to eliminate or mitigate stress effects on cognitive skill performance. First, the use of simulated events for training can reduce the novelty and uncertainty that can lead to stress and performance impairments. Second, training to make cognitive processing more efficient and less reliant on attention and memory resources can offset the reductions in these resources that occur under stressful conditions. Third, training that targets crew communications skills can reduce the likelihood that communications will fail under stress.

NUREG/CR-6128: PIPING BENCHMARK PROBLEMS FOR THE ABB/CE SYSTEM 80+ STANDARDIZED PLANT. BEZLER, P.; DEGRASSI, D.; BRAVERMAN, J.; et al. Brookhaven National Laboratory. July 1994. 234pp. 9408150117. BNL-NUREG-52396. 80543:090.

To satisfy the need for verification of the computer programs and modeling techniques that will be used to perform the final piping analysis for the ABB/ Combustion Engineering System

80+ Standardized Plant, three benchmark problems were developed. The problems are representative piping systems subjected to representative dynamic loads with solutions developed using the methods being proposed for analysis for the System 80+ standard design. It will be required that the combined license licensees demonstrate that their solutions to these problems are in agreement with the benchmark problem set.

NUREG/CR-6132: BIAxIAL LOADING AND SHALLOW-FLAW EFFECTS ON CRACK-TIP CONSTRAINT AND FRACTURE TOUGHNESS. BASS,B.R.; BRYSON,J.W.; THEISS,T.J.; et al. Oak Ridge National Laboratory. January 1994. 74pp. 9403140303. ORNL/TM-12498. 78477:001.

A program to develop and evaluate fracture methodologies for the assessment of crack-tip constraint effects on fracture toughness of reactor pressure vessel (RPV) steels has been initiated in the Heavy-Section Steel Technology (HSST) Program. Crack-tip constraint is an issue that significantly impacts fracture mechanics technologies employed in safety assessment procedures for commercially licensed nuclear RPVs. The focus of studies described herein is on the evaluation of two stressed-based methodologies for quantifying crack-tip constraint (i.e., J-Q theory and a micromechanical scaling model based on critical stressed volumes) through applications to experimental and fractographic data. Data were utilized from single-edge notch bend (SENB) specimens and HSST-developed cruciform beam specimens that were tested in HSST shallow-crack and biaxial testing programs. Results from applications indicate that both the J-Q methodology and the micromechanical scaling model can be used successfully to interpret experimental data from the shallow- and deep-crack SENB specimen tests. When applied to the uniaxially and biaxially loaded cruciform specimens, the two methodologies showed some promising features, but also raised several questions concerning the interpretation of constraint conditions in the specimen based on near-tip stress fields. Fractographic data taken from the fracture surfaces of the SENB and cruciform specimens are used to assess the relevance of stress-based fracture characterizations to conditions at cleavage initiation sites. Unresolved issues identified from these analyses require resolution as part of a validation process for biaxial loading applications. This report is designated as HSST Report No. 142.

NUREG/CR-6133: FRAGMENTATION AND QUENCH BEHAVIOR OF CORIUM MELT STREAMS IN WATER. SPENCER,B.W.; WANG,K.; BLOMQUIST,C.A.; et al. Argonne National Laboratory. February 1994. 407pp. 9404010160. ANL-93/32. 78717:066.

The interaction of molten core materials with water has been investigated for the pour stream mixing mode. This interaction plays a crucial role during the later stages of in-vessel core melt progression inside a light water reactor such as during the TMI-2 accident. The key issues which arise during the molten core relocation include: i) the thermal attack and possible damage to the RPV lower head from the impinging molten fuel stream and/or the debris bed, ii) the molten fuel relocation pathways including the effects of redistribution due to core support structure and the reactor lower internals, iii) the quench rate of the molten fuel through the water in the lower plenum, iv) the steam generation and hydrogen generation during the interaction, v) the transient pressurization of the primary system, and vi) the possibility of a steam explosion. In order to understand these issues, a series of six experiments (designated CCM-1 through -6) was performed in which molten corium passed through a deep pool of water in a long, slender pour stream mode. Results discussed include the transient temperatures and pressures, the rate and magnitude of steam/hydrogen generation, and the posttest debris characteristics.

NUREG/CR-6138: USER'S GUIDE FOR SIMPLIFIED COMPUTER MODELS FOR THE ESTIMATION OF LONG-TERM PERFORMANCE OF CEMENT-BASED MATERIALS. PLANSKY,L.E.; SEITZ,R.R. EG&G Idaho, Inc. February 1994. 94pp. 9403140184. EGG-2719. 78456:092.

This report documents user instructions for several simplified subroutines and driver programs that can be used to estimate various aspects of the long-term performance of cement-based barriers used in low-level radioactive waste disposal facilities. The subroutines are prepared in a modular fashion to allow flexibility for a variety of applications. Three levels of codes are provided: the individual subroutines, interactive drivers for each of the subroutines, and an interactive main driver, CEMENT, that calls each of the individual drivers. The individual subroutines for the different models may be taken independently and used in larger programs, or the driver modules can be used to execute the subroutines separately or as a part of the main driver routine. A brief program description is included and user-interface instructions for the individual subroutines are documented in the main report. These are intended to be used when the subroutines are used as subroutines in a larger computer code. User instructions for the drivers and example interactive screens for the main drivers are provided in Appendix A. Examples showing the use of the individual driver routines to execute the different subroutines and test data are included in Appendix B. Programmer notes are provided in Appendix C and the configuration control system is discussed in Appendix D.

NUREG/CR-6139: CRACK-ARREST TESTS ON TWO IRRADIATED HIGH-COPPER WELDS. Phase II: Results Of Duplex-Type Specimens. ISKANDER,S.K.; CORWIN,W.R.; NANSTAD,R.K. Oak Ridge National Laboratory. March 1994. 138pp. 9403140206. ORNL/TM-12513. 78476:176.

The objective of the Heavy-Section Steel Irradiation Program Sixth Irradiation Series is to determine the effect of neutron irradiation on the shift and shape of the lower-bound curve to crack-arrest toughness data. Two submerged-arc welds with copper contents of 0.23 and 0.31 wt % were commercially fabricated in 200 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 1.9×10^{19} neutrons/cm² (> 1 MeV). This is the second report giving the results of the tests on irradiated duplex-type crack-arrest specimens. Charpy V-notch specimens irradiated in the same capsules as the crack-arrest specimens were also tested, and a 41-J transition temperature shift was determined from these specimens. "Mean" curves of the same form as the ASME K(Ia) curve were fit to all the data with only the reference temperature as a parameter. The shift between the mean curves agrees well with the 41-J transition temperature shift obtained from the Charpy V-notch specimen tests.

NUREG/CR-6142: TENSILE-PROPERTY CHARACTERIZATION OF THERMALLY AGED CAST STAINLESS STEELS. MICHAUD,W.F.; TOBEN,P.T.; SOPPET,W.K.; et al. Argonne National Laboratory. February 1994. 257pp. 9402250117. ANL-93/35. 78278:001.

The effect of thermal aging on tensile properties of cast stainless steels during service in light water reactors has been evaluated. Tensile data for several experimental and commercial heats of cast stainless steels are presented. Thermal aging increases the tensile strength of these steels. The high-C Mo-bearing CF-8M steels are more susceptible to thermal aging than the Mo-free CF-3 or CF-8 steels. A procedure and correlations are presented for predicting the change in tensile flow and yield stresses and engineering stress-vs.-strain curve of cast stainless steel as a function of time and temperature of service. The tensile properties of aged cast stainless steel are estimated from known material information, i.e., chemical composition and the initial tensile strength of the steel. The correlations described in this report may be used for assessing thermal embrittlement of cast stainless steel components.

NUREG/CR-6143 V02P1A: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Analysis Of Core Damage Frequency From Internal Events For Plant Operational State 5 During A Refueling Outage. Sections 1-9. WHITEHEAD, D.W. Sandia National Laboratories. DARBY, J. Science & Engineering Associates, Inc. YAKLE, J.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). June 1994. 284pp. 9408030161. SAND93-2440. 80442:001.

This document contains the accident sequence analysis of internally initiated events for Grand Gulf, Unit 1 as it operates in the Low Power and Shutdown Plant Operational State 5 during a refueling outage. The report documents the methodology used during the analysis, describes the results from the application of the methodology, and compares the results with the results from two full power analyses performed on Grand Gulf.

NUREG/CR-6143 V02P1B: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Analysis Of Core Damage Frequency From Internal Events For Plant Operational State 5 During A Refueling Outage. Section 10. WHITEHEAD, D.W. Sandia National Laboratories. DARBY, J. Science & Engineering Associates, Inc. YAKLE, J.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). June 1994. 997pp. 9408030170. SAND93-2440. 80426:001.

See NUREG/CR-6143, V02, P1A abstract.

NUREG/CR-6143 V02P1C: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Analysis Of Core Damage Frequency From Internal Events For Plant Operational State 5 During A Refueling Outage. Main Report. WHITEHEAD, D.W. Sandia National Laboratories. DARBY, J. Science & Engineering Associates, Inc. YAKLE, J.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). June 1994. 373pp. 9408030176. SAND93-2440. 80443:001.

See NUREG/CR-6143, V02, P1A abstract.

NUREG/CR-6143 V02PT2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Analysis Of Core Damage Frequency From Internal Events For Plant Operational State 5 During Refueling Outage. Internal.... DARBY, J. Science & Engineering Associates, Inc. WHITEHEAD, D.W.; STAPLE, B.D.; et al. Sandia National Laboratories. June 1994. 609pp. 9408030187. SAND93-2440. 80432:001.

This document contains the accident sequence analysis of internally initiated events for Grand Gulf, Unit 1 as it operates in the Low Power and Shutdown Plant Operational State 5 during a refueling outage. The report documents the methodology used during the analysis, describes the results from the application of the methodology, and compares the results with the results from two full power analyses performed on Grand Gulf.

NUREG/CR-6143 V02PT3: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Analysis Of Core Damage Frequency From Internal Events For Plant Operational State 5 During A Refueling Outage. Internal.... YAKLE, J. Science Applications International Corp. (formerly Science Applications, Inc.). DARBY, J. Science & Engineering Associates, Inc. WHITEHEAD, D.W.; et al. Sandia National Laboratories. June 1994. 834pp. 9408030235. SAND93-2440. 80429:001.

This report provides supporting documentation for various tasks associated with the performance of the probabilistic risk assessment for Plant Operational State 5 during a refueling outage at Grand Gulf, Unit 1 as documented in Volume 2, Part 1 of NUREG/CR-6143.

NUREG/CR-6143 V02PT4: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Analysis Of Core Damage Frequency From Internal Events For Plant Operational State 5 During A Refueling Outage. Internal.... FORESTER, J. Science Applications International Corp. (formerly Science Applications, Inc.). WHITEHEAD, D.W. Sandia National Laboratories. DARBY, J.; et al. Science & Engineering Associates, Inc. June 1994. 889pp. 9408030243. SAND93-2440. 80439:001.

See NUREG/CR-6143, V02, PT3 abstract.

NUREG/CR-6143 V03: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Analysis Of Core Damage Frequency From Internal Events For Plant Operational State 5 During A Refueling Outage. LAMBRIGHT, J. Sandia National Laboratories. ROSS, S.; LYNCH, J.; et al. Science & Engineering Associates, Inc. July 1994. 112pp. 9408180206. SAND93-2440. 80613:244.

This report presents the details of the analysis of core damage frequency due to fire during shutdown Plant Operational State 5 at the Grand Gulf Nuclear Station. Insights from previous fire analyses (Peach Bottom, Surry, LaSalle) were used to the greatest extent possible in this analysis. The fire analysis was fully integrated utilizing the same event trees and fault trees that were used in the internal events analysis. In assessing shutdown risk due to fire at Grand Gulf, a detailed screening was performed which included the following elements: (a) Computer-aided vital area analysis, (b) Plant inspections, (c) Credit for automatic fire protection systems, (d) Recovery of random failures, and (e) Detailed fire propagation modeling. This screening process revealed that all plant areas had a negligible ($<1.0E-8$ per year) contribution to fire-induced core damage frequency.

NUREG/CR-6143 V04: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Analysis Of Core Damage Frequency From Internally Induced Flooding Events For Plant Operational State 5 During a Refueling.... DANDINI, V.; STAPLE, B.D.; et al. Sandia National Laboratories. FORESTER, J. Science Applications International Corp. (formerly Science Applications, Inc.). July 1994. 472pp. 9409070061. SAND93-2440. 80787:001.

An estimate of the contribution of internal flooding to the mean core damage frequency at the Grand Gulf Nuclear Station was calculated for Plant Operational State 5 during a refueling outage. Pursuant to this objective, flood zones and sources were identified and flood volumes were calculated. Equipment necessary for the maintenance of plant safety was identified and its vulnerability to flooding was determined. Event trees and fault trees were modified or developed as required, and PRA quantification was performed using the IRRAS code. The mean core damage frequency estimate for GGNS during POS 5 was found to be $2.3 E-8$ per year.

NUREG/CR-6143 V05: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Analysis Of Core Damage Frequency From Seismic Events During Mid-Loop Operations. Main Report. BUDNITZ, R.J. Future Resources Associates, Inc. DAVIS, P.R. PRD Consulting. RAVINDRA, M.K.; et al. EQE, Inc. August 1994. 110pp. 9409230278. 81006:083.

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects being performed by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL). Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied. The objectives of the program are to assess the risks

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of severe accidents initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The objective of this report is to document the approach utilized in the Grand Gulf plant and discuss the results obtained. A parallel report for the Surry plant is prepared by SNL.

NUREG/CR-6144 V02P1A: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUT-DOWN OPERATIONS AT SURRY, UNIT 1. Analysis Of Core Damage Frequency From Internal Events During Mid-Loop Operations. Main Report (Chapters 1-6). CHU, T.L.; MUSICKI, Z.; KOHUT, P.; et al. Brookhaven National Laboratory. June 1994. 494pp. 9408120219. BNL-NUREG-52399. 80515:001.

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects being performed by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL). Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied. The objectives of the program are to assess the risks of severe accidents initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The objective of this report is to document the approach utilized in the Surry plant and discuss the results obtained. A parallel report for the Grand Gulf plant is prepared by SNL. This study shows that the core-damage frequency during mid-loop operation at the Surry plant is comparable to that of power operation. We recognize that there is very large uncertainty in the human error probabilities in this study. This study identified that only a few procedures are available for mitigating accidents that may occur during shutdown. Procedures written specifically for shutdown accidents would be useful.

NUREG/CR-6144 V02P1B: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUT-DOWN OPERATIONS AT SURRY, UNIT 1. Analysis Of Core Damage Frequency From Internal Events During Mid-Loop Operations. Main Report (Chapters 7-12). CHU, T.L.; MUSICKI, Z.; KOHUT, P.; et al. Brookhaven National Laboratory. June 1994. 630pp. 9408150181. BNL-NUREG-52399. 80538:001.

See NUREG/CR-6144, V02, P1A abstract.

NUREG/CR-6144 V02P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUT-DOWN OPERATIONS AT SURRY, UNIT 1. Analysis Of Core Damage Frequency From Internal Events During Mid-Loop Operations. Appendices A-D. CHU, T.L.; MUSICKI, Z.; KOHUT, P.; et al. Brookhaven National Laboratory. June 1994. 468pp. 9408150192. BNL-NUREG-52399. 80540:001.

See NUREG/CR-6144, V02, P1A abstract.

NUREG/CR-6144 V02P3A: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUT-DOWN OPERATIONS AT SURRY, UNIT 1. Analysis Of Core Damage Frequency From Internal Events During Mid-Loop Operations. Appendices E (Sections E.1-E.8). CHU, T.L.; MUSICKI, Z.; KOHUT, P.; et al. Brookhaven National Laboratory. June 1994. 560pp. 9408150228. BNL-NUREG-52399. 80555:001.

See NUREG/CR-6144, V02, P1A abstract.

NUREG/CR-6144 V02P3B: EVALUATION OF POTENTIAL

SEVERE ACCIDENTS DURING LOW POWER AND SHUT-DOWN OPERATIONS AT SURRY, UNIT 1. Analysis Of Core Damage Frequency From Internal Events During Mid-Loop Operations. Appendices E (Sections E.9-E.16). CHU, T.L.; MUSICKI, Z.; KOHUT, P.; et al. Brookhaven National Laboratory. June 1994. 502pp. 9408150234. BNL-NUREG-52399. 80556:202.

See NUREG/CR-6144, V02, P1A abstract.

NUREG/CR-6144 V02P4: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUT-DOWN OPERATIONS AT SURRY, UNIT 1. Analysis Of Core Damage Frequency From Internal Events During Mid-Loop Operations. Appendices F-H. CHU, T.L.; MUSICKI, Z.; KOHUT, P.; et al. Brookhaven National Laboratory. June 1994. 612pp. 9408150246. BNL-NUREG-52399. 80559:001.

See NUREG/CR-6144, V02, P1A abstract.

NUREG/CR-6144 V02P5: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUT-DOWN OPERATIONS AT SURRY, UNIT 1. Analysis Of Core Damage Frequency From Internal Events During Mid-Loop Operations. Appendices I. CHU, T.L.; MUSICKI, Z.; KOHUT, P.; et al. Brookhaven National Laboratory. June 1994. 313pp. 9408150284. BNL-NUREG-52399. 80558:001.

See NUREG/CR-6144, V02, P1A abstract.

NUREG/CR-6144 V03 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUT-DOWN OPERATIONS AT SURRY, UNIT 1. Analysis Of Core Damage Frequency From Internal Fires During Mid-Loop Operations. Main Report. MUSICKI, Z.; CHU, T.L. Brookhaven National Laboratory. HO, V.; et al. PLG, Inc. (formerly Pickard, Lowe & Garrick, Inc.). July 1994. 180pp. 9408180144. BNL-NUREG-52399. 80618:083.

See NUREG/CR-6144, V02, P1A abstract.

NUREG/CR-6144 V03 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUT-DOWN OPERATIONS AT SURRY, UNIT 1. Analysis Of Core Damage Frequency From Internal Fires During Mid-Loop Operations. Appendices. MUSICKI, Z.; CHU, T.L. Brookhaven National Laboratory. HO, V.; et al. PLG, Inc. (formerly Pickard, Lowe & Garrick, Inc.). July 1994. 400pp. 9408180149. BNL-NUREG-52399. 80616:001.

See NUREG/CR-6144, V02, P1A abstract.

NUREG/CR-6144 V04: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUT-DOWN OPERATIONS AT SURRY, UNIT 1. Analysis Of Core Damage Frequency From Internal Floods During Mid-Loop Operations. KOHUT, P. Brookhaven National Laboratory. July 1994. 200pp. 9408180160. BNL-NUREG-52399. 80614:060.

See NUREG/CR-6144, V02, P1A abstract.

NUREG/CR-6144 V05: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUT-DOWN OPERATIONS AT SURRY, UNIT 1. Analysis Of Core Damage Frequency From Seismic Events During Mid-Loop Operations. Main Report. BUDNITZ, R.J. Future Resources Associates, Inc. DAVIS, P.R. PRD Consulting. RAVINDRA, M.K.; et al. EQE, Inc. August 1994. 114pp. 9409230293. 81010:116.

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects being performed by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL). Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied. The objectives of the program are to assess the risks of severe accidents initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents

initiated during full power operation as assessed in NUREG-1150. The objective of this report is to document the approach utilized in the Surry plant and discuss the results obtained. A parallel report for the Grand Gulf plant is prepared by SNL.

NUREG/CR-6145: VERIFICATION AND VALIDATION OF THE SAPHIRE VERSION 4.0 PRA SOFTWARE PACKAGE. BOLANDER,T.W.; CALLEY,M.B.; CAPPS,E.L.; et al. Idaho National Engineering Laboratory. February 1994. 331pp. 9404010225. EGG-2713. 78722:289.

A verification and validation (V&V) process has been performed for the System Analysis Programs for Hands-on Integrated Reliability Evaluation (SAPHIRE). SAPHIRE is a set of four computer programs that the Nuclear Regulatory Commission (NRC) developed to perform probabilistic risk assessments (PRAs). These programs allow an analyst to create, quantify, and evaluate the risk associated with a facility or process being analyzed. The programs included in this set are Integrated Reliability and Risk Analysis System (IRRAS), System Analysis and Risk Assessment (SARA), Models and Results Database (MARD), and Fault Tree/Event Tree/Piping and Instrumentation Diagram (FEP) graphical editor. The V&V steps included a V&V plan to describe the process and criteria by which the V&V would be performed; a software requirements documentation review to determine the correctness, completeness, and traceability of the requirements; a user survey to determine the usefulness of the user documentation, identification and testing of vital and non-vital features, and documentation of the test results.

NUREG/CR-6146: LOCAL CONTROL STATIONS: HUMAN ENGINEERING ISSUES AND INSIGHTS. BROWN,W.S.; HIGGINS,J.C.; O'HARA,J.M. Brookhaven National Laboratory. September 1994. 74pp. 9411080056. BNL-NUREG-52400. 81648:146.

The objective of this research project was to evaluate current human engineering at local control stations (LCSs) in nuclear power plants, and to identify good human engineering practices relevant to the design of these operator interfaces. General literature and reports of operating experience were reviewed to determine the extent and type of human engineering deficiencies at LCSs in nuclear power plants. In-plant assessments were made of human engineering at single-function as well as multifunction LCSs. Besides confirming the existence of human engineering deficiencies at LCSs, the in-plant assessments provided information about the human engineering upgrades that have been made at nuclear power plants. Upgrades were typically the result of any of three influences - regulatory activity, broad industry initiatives such as INPO, and specific in-plant programs (e.g., activities related to training). It is concluded that the quality of LCSs is quite variable and might be improved if there were greater awareness of good practices and existing human engineering guidance relevant to these operator interfaces, which is available from a variety of sources. To make such human engineering guidance more readily accessible, guidelines were compiled from such sources and included in the report as an appendix.

NUREG/CR-6147 V01: CHARACTERIZATION OF CLASS A LOW-LEVEL RADIOACTIVE WASTE 1986-1990. Executive Summary. DEHMEL,J.-C.; LOOMIS,D.; MAURO,J.; et al. S. Cohen & Associates, Inc. January 1994. 92pp. 9403140342. 78456:001.

This report describes the physical, chemical, and radiological properties of Class A low-level radioactive waste using data contained in the Manifest Information Management System (MIMS). Other sources of information include reports prepared by the NRC, DOE, low-level waste Compacts and States, and trade industries. The database characterizes low-level waste shipped for disposal from 1986 to 1990. A computer program was developed to analyze the data, with the results summarized in tables, histograms, and cumulative distribution curves presenting radionuclide concentration distributions in Class A waste as a function of waste streams, waste generators, and by re-

gions. The report also provides information characterizing the methods and facilities used to treat and dispose of non-radioactive waste, including industrial, municipal, and hazardous waste regulated under Subparts C and D of RCRA. The information includes a list of disposal options, the geographical locations of such facilities, and a description of such processing and disposal facilities.

NUREG/CR-6147 V02: CHARACTERIZATION OF CLASS A LOW-LEVEL RADIOACTIVE WASTE 1986-1990. Main Report-Part A. DEHMEL,J.-C.; LOOMIS,D.; MAURO,J.; et al. S. Cohen & Associates, Inc. January 1994. 300pp. 9403140349. 78455:001. See NUREG/CR-6147,V01 abstract.

NUREG/CR-6147 V03: CHARACTERIZATION OF CLASS A LOW-LEVEL RADIOACTIVE WASTE 1986-1990. Main Report-Part B. DEHMEL,J.-C.; LOOMIS,D.; MAURO,J.; et al. S. Cohen & Associates, Inc. January 1994. 300pp. 9403140351. 78452:001. See NUREG/CR-6147,V01 abstract.

NUREG/CR-6147 V04: CHARACTERIZATION OF CLASS A LOW-LEVEL RADIOACTIVE WASTE 1986-1990. Appendices A-E. DEHMEL,J.-C.; LOOMIS,D.; MAURO,J.; et al. S. Cohen & Associates, Inc. January 1994. 300pp. 9403140353. 78475:001. See NUREG/CR-6147,V01 abstract.

NUREG/CR-6147 V05: CHARACTERIZATION OF CLASS A LOW-LEVEL RADIOACTIVE WASTE 1986-1990. Appendix F. DEHMEL,J.-C.; LOOMIS,D.; MAURO,J.; et al. S. Cohen & Associates, Inc. January 1994. 600pp. 9403140355. 78473:012. See NUREG/CR-6147,V01 abstract.

NUREG/CR-6147 V06: CHARACTERIZATION OF CLASS A LOW-LEVEL RADIOACTIVE WASTE 1986-1990. Appendices G-J. DEHMEL,J.-C.; LOOMIS,D.; MAURO,J.; et al. S. Cohen & Associates, Inc. January 1994. 500pp. 9403140360. 78471:114. See NUREG/CR-6147,V01 abstract.

NUREG/CR-6147 V07: CHARACTERIZATION OF CLASS A LOW-LEVEL RADIOACTIVE WASTE 1986-1990. Appendices K-P. DEHMEL,J.-C.; LOOMIS,D.; MAURO,J.; et al. S. Cohen & Associates, Inc. January 1994. 400pp. 9403140364. 78470:005. See NUREG/CR-6147,V01 abstract.

NUREG/CR-6149: APPLICATIONS OF FIBER OPTICS IN PHYSICAL PROTECTION. BUCKLE,T.H. Sandia National Laboratories. March 1994. 51pp. 9404080104. SAND93-2478. 78794:261.

The purpose of this report is to provide technical information useful for the development of fiber-optic communications and intrusion detection subsystems relevant to physical protection. There are major sections on fiber-optic technology and applications. Other topics include fiber-optic system components and systems engineering. This document also contains a glossary, a list of standards and specifications, and a list of fiber-optic vendors.

NUREG/CR-6151: FEASIBILITY OF DEVELOPING RISK-BASED RANKINGS OF PRESSURE BOUNDARY SYSTEMS FOR INSERVICE INSPECTION. VO,T.V.; SMITH,B.W.; SIMONEN,F.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1994. 86pp. 9410130086. PNL-8912. 81266:091.

Goals of the Nondestructive Evaluation Reliability Program sponsored by the NRC at PNL are to 1) assess current inspection techniques and requirements of all pressure boundary systems and components, 2) determine if improvements are needed, and 3) if necessary, develop recommendations for revising the applicable ASME Codes and regulatory requirements. In evaluating approaches that could be used to provide a technical basis for improved inservice inspection (ISI) plans, PNL has developed and applied a method that uses results of probabilistic risk assessment (PRA) to rank pressure boundary systems for ISI. In the PNL program, the feasibility of developing risk-based generic ISI requirements is being conducted in two phases. Phase I involves identifying and prioritizing the systems most relevant to plant safety. The results of these evaluations

will be consolidated into requirements for comprehensive piping system inspections that will be developed in Phase II. This report presents the Phase I results of evaluations for eight selected plant PRAs. Based on the results of this study, it appears that there are generic insights that can be extrapolated from the selected plants to specific classes of light water reactors.

NUREG/CR-6152: EXPERIMENTS TO INVESTIGATE DIRECT CONTAINMENT HEATING PHENOMENA WITH SCALED MODELS OF THE SURRY NUCLEAR POWER PLANT. BLANCHAT, T.K.; ALLEN, M.D.; PILCH, M.M.; et al. Sandia National Laboratories. June 1994. 206pp. 9407260019. SAND93-2519. 80344:001.

The Containment Technology Test Facility (CTTF) and the Surtsey Test Facility at Sandia National Laboratories are used to perform scaled experiments that simulate High Pressure Melt Ejection accidents in a nuclear power plant (NPP). These experiments are designed to investigate the effects of direct containment heating (DCH) phenomena on the containment load. High-temperature, chemically reactive melt (thermite) is ejected by high-pressure steam into a scale model of a reactor cavity. Debris is entrained by the steam blowdown into a containment model where specific phenomena, such as the effect of sub-compartment structures, prototypic air/steam/hydrogen atmospheres, and hydrogen generation and combustion, can be studied. Four Integral Effects Tests (IETs) have been performed with scale models of the Surry NPP to investigate DCH phenomena. The 1/6 scale Integral Effects Tests (IET-9, IET-10, and IET-11) were conducted in CTTF, which is a 1/6(th) scale model of the Surry reactor containment building (RCB). The 1/10(th) scale IET test (IET-12) was performed in the Surtsey vessel, which had been configured as a 1/10 scale Surry RCB. Scale models were constructed in each of the facilities of the Surry structures, including the reactor pressure vessel, reactor support skirt, control rod drive missile shield, biological shield wall, cavity, instrument tunnel, residual heat removal platform and heat exchangers, seal table room and seal table, operating deck, and crane wall. This report describes these experiments and gives the results.

NUREG/CR-6154 V01: EXPERIMENTAL RESULTS FROM CONTAINMENT PIPING BELLOWS SUBJECTED TO SEVERE ACCIDENT CONDITIONS. Results From Bellows Tested In "Like-New" Conditions. LAMBERT, L.D.; PARKS, M.B. Sandia National Laboratories. September 1994. 81pp. 9411080061. SAND94-1711. 81648:221.

Bellows are an integral part of the containment pressure boundary in nuclear power plants. They are used at piping penetrations to allow relative movement between piping and the containment wall, while minimizing the load imposed on the piping and wall. Piping bellows are primarily used in steel containments; however, they have received limited use in some concrete (reinforced and prestressed) containments. In a severe accident they may be subjected to pressure and temperature conditions that exceed the design values, along with a combination of axial and lateral deflections. A test program to determine the leak-tight capacity of containment penetration bellows is being conducted under the sponsorship of the U.S. Nuclear Regulatory Commission at Sandia National Laboratories. Several different bellows geometries, representative of actual containment bellows, have been subjected to extreme deflections along with pressure and temperature loads. The bellows geometries and loading conditions are described along with the testing apparatus and procedures. A total of thirteen bellows have been tested, all in the 'like-new' condition. (Additional tests are planned of bellows that have been subjected to corrosion.) The tests showed that bellows are capable of withstanding relatively large deformations, up to, or near, the point of full compression or elongation, before developing leakage. The test data is presented and discussed.

NUREG/CR-6156: SUMMARY OF COMMENTS RECEIVED FROM WORKSHOPS ON RADIOLOGICAL CRITERIA FOR DECOMMISSIONING. CAPLIN, J.; PAGE, G.; SMITH, D.; et al. Advanced Systems Technology, Inc. January 1994. 200pp. 9402150306. 78122:101.

The Nuclear Regulatory Commission (NRC) is conducting an enhanced participatory rulemaking to establish radiological criteria for site cleanup and decommissioning of NRC-licensed facilities. Open public meetings were held during 1993 in Chicago, IL, San Francisco, CA, Boston, MA, Dallas, TX, Philadelphia, PA, Atlanta, GA, and Washington, DC. Interested parties were invited to provide input on the rulemaking issues before the NRC staff develops a draft proposed rule. This report summarizes 3,635 comments categorized from transcripts of the seven workshops and 1,677 comments from 100 NRC docketed letters from individuals and organizations. No analysis or response to the comments is included. The comments reflect a broad spectrum of view-points on the issues related to radiological criteria for site cleanup and decommissioning. The NRC also held public meetings on the scope of the Generic Environmental Impact Statement (GEIS) during July 1993. The GEIS meetings were held in Washington, DC., San Francisco, CA, Oklahoma City, OK, and Cleveland, OH. Related comments from these meetings were reviewed and comments which differed substantially from those from the workshops are also summarized in the body of the report. A summary of the comments from the GEIS scoping meetings is included as an Appendix.

NUREG/CR-6157: SURVEY AND EVALUATION OF AGING RISK ASSESSMENT METHODS AND APPLICATIONS. SANZO, D.; KVAM, P. Los Alamos National Laboratory. APOSTOLAKIS, G.; et al. Advanced Systems Concepts Associates. November 1994. 180pp. 9412130107. LA-12715-MS. 82007:186.

The US Nuclear Regulatory Commission initiated the nuclear power plant aging research program about 6 years ago to gather information about nuclear power plant aging. Since then, this program has collected a significant amount of information, largely qualitative, on plant aging and its potential effects on plant safety. However, this body of knowledge has not yet been integrated into formalisms that can be used effectively and systematically to assess plant risk resulting from aging, although models for assessing the effect of increasing failure rates on core damage frequency have been proposed. This report surveys the work on the aging of systems, structures, and components (SSCs) of nuclear power plants, as well as associated data bases. We take a critical look at the need to revise probabilistic risk assessments (PRAs) so that they will include the contribution to risk from plant aging, the adequacy of existing methods for evaluating this contribution, and the adequacy of the data that have been used in these evaluation methods. We identify a preliminary framework for integrating the aging of SSCs into the PRA and include the identification of necessary data for such an integration.

NUREG/CR-6158: IMPLICATIONS FOR ACCIDENT MANAGEMENT OF ADDING WATER TO A DEGRADING REACTOR CORE. KUAN, P.; HANSON, D.J.; PAFFORD, D.J.; et al. EG&G Idaho, Inc. February 1994. 220pp. 9404010230. EGG-2644. 78723:256.

This report evaluates both the positive and negative consequences of adding water to a degraded reactor core during a severe accident. The evaluation discusses the earliest possible stage at which an accident can be terminated and how plant personnel can best respond to undesired results. Specifically discussed are (a) the potential for plant personnel to add water for a range of severe accidents, (b) the time available for plant personnel to act, (c) possible plant responses to water added during the various stages of core degradation, (d) plant instrumentation available to understand the core condition and (e) the expected response of the instrumentation during the various stages of severe accidents.

NUREG/CR-6160: SUMMARY OF IMPORTANT RESULTS AND SCDAP/RELAP5 ANALYSIS FOR OECD LOFT EXPERIMENT LP-FP-2. CORYELL, E.W.; AKERS, D.W.; ALLISON, C.M.; et al. EG&G Idaho, Inc. April 1994. 165pp. 9405310168. NEA-CSNI-R(94)3. 79591:001.

This report summarizes significant technical findings from the LP-FP-2 Experiment sponsored by the Organization of Economic Cooperation and Development (OECD). It was the second, and final, fission product experiment conducted in the Loss-of-Fluid Test (LOFT) facility at the Idaho National Engineering Laboratory. The overall technical objective of the test was to contribute to the understanding of fuel rod behavior, hydrogen generation, and fission product release, transport, and deposition during a V-sequence accident scenario that resulted in severe core damage. An 11 by 11 test bundle, comprised of 100 prepressurized fuel rods, 11 control rods, and 10 instrumented guide tubes, was surrounded by an insulating shroud and contained in a specially designed central fuel module, that was inserted into the LOFT reactor. The simulated transient was a V-sequence loss-of-coolant accident scenario featuring a pipe break in the low pressure injection system line attached to the hot leg of the LOFT broken loop piping. The transient was terminated by reflood of the reactor vessel when the outer wall shroud temperature reached 1517 K. With sustained fission power and heat from oxidation and metal-water reactions, elevated temperatures resulted in zircaloy melting, fuel liquefaction, material relocation, and the release of hydrogen, aerosols, and fission products. A description and evaluation of the major phenomena, based upon the response of on-line instrumentation, analysis of fission product data, postirradiation examination of the fuel bundle, and calculations using the SCDAP/RELAP5 computer code, are presented.

NUREG/CR-6161: BUCKLING EVALUATION OF SYSTEM 80+(TM) CONTAINMENT. GREIMANN, L.; FANOUS, F.; SAFAR, S.; et al. Iowa State Univ., Ames, IA. August 1994. 145pp. 9411080051. IS-5103. 81648:001.

The System 80+(TM) containment may be subjected to compressive forces which could cause it to become unstable. The stability of the containment shell under prescribed loading combinations was investigated with two analysis levels: axisymmetric and three dimensional. An axisymmetric shell model, including additional mass to account for penetrations and the spray header system, was analyzed using BOSOR4 and BOSOR5 finite difference codes. Loading combinations with pressure, temperature, self weight, and seismic satisfied the American Society of Mechanical Engineers (ASME) stress allowables. The buckling assessment was performed using the worst meridian assumption, including material nonlinearities and a sinusoidal axisymmetric imperfection. The minimum factor of safety for Service Level C was 2.35. A SSE seismic margin of 2.91 was calculated. The ABAQUS finite element code was selected for the three dimensional analysis and tested with classical and BOSOR solutions. The three dimensional model included the equipment hatch, two personnel airlocks, and additional mass for the spray system and small penetrations. Material nonlinearity and an axisymmetric sinusoidal imperfection were incorporated. A minimum factor of safety of 1.91 was predicted, which does not satisfy ASME Section NE3222.1 or Regulatory Guide 1.57. Code Case N-284 is satisfied. The analysis is conservative primarily because the SRSS 10% method provides a conservative estimate of modal coupling.

NUREG/CR-6162: EFFECTS OF PRIOR DUCTILE TEARING ON CLEAVAGE FRACTURE TOUGHNESS IN THE TRANSITION REGION. DODDS, R.H.; TANG, M. Illinois, Univ. of, Urbana, IL. ANDERSON, T.L. Texas A&M Univ., College Station, TX. June 1994. 45pp. 9407250325. UILU-ENG93-2014. 80329:232.

Previous work by the authors described a micromechanics fracture model to correct measured $J(c)$ -values for the mechanistic effects of large-scale yielding. This new work extends the model to also include the influence of ductile crack extension prior to cleavage. Ductile crack extensions of 10-15 X the initial

crack tip opening displacement at initiation are considered in plane-strain, finite element computations. The finite element results demonstrate a significant elevation in crack-tip constraint due to macroscopic "sharpening" of the extending tip relative to the blunt tip at the initiation of growth. However this effect is offset partially by the additional plastic deformation associated with the increased applied J required to grow the crack. The initial a/W ratio, tearing modulus, strain hardening exponent and specimen size interact in a complex manner to define the evolving near-tip conditions for cleavage fracture. The paper explores development of the new model, provides necessary graphs and procedures for its application and demonstrates the effects of the model on fracture data sets for two pressure vessel steels (A533B and A515).

NUREG/CR-6164: RELEASE OF RADIONUCLIDES AND CHELATING AGENTS FROM CEMENT-SOLIDIFIED DECONTAMINATION LOW-LEVEL RADIOACTIVE WASTE COLLECTED FROM THE PEACH BOTTOM ATOMIC POWER STATION UNIT 3. AKERS, D.W.; KRAFT, N.C.; MANDLER, J.W. EG&G Idaho, Inc. March 1994. 71pp. 9405050409. EGG-2722. 79165:130.

As part of a study being performed for the Nuclear Regulatory Commission (NRC), small-scale waste-form specimens were collected during a low oxidation-state transition-metal ion (LOMI)-nitric permanganate (NP)-LOMI solidification performed in October 1989 at the Peach Bottom Atomic Power Station Unit 3. The purpose of this program was to evaluate the performance of cement-solidified decontamination waste to meet the low-level waste stability requirements defined in the NRC's "Technical Position on Waste Form," Revision 1. The samples were acquired and tested because little data have been obtained on the physical stability of actual cement-solidified decontamination ion-exchange resin waste forms and on the leachability of radionuclides and chelating agents from those waste forms. The Peach Bottom waste-form specimens were subjected to compressive strength, immersion, and leach testing in accordance with the NRC's "Technical Position on Waste Form," Revision 1. Results of this study indicate that the specimens withstood the compression tests (>500 psi) before and after immersion testing and leaching, and that the leachability indexes for all radionuclides, including (14)C, (99)Tc, and (129)I, are well above the leachability index requirement of 6.0, required by the NRC's "Technical Position on Waste Form," Revision 1.

NUREG/CR-6166: RISK IMPACT OF TECHNICAL SPECIFICATIONS REQUIREMENTS DURING SHUTDOWN FOR BWRs. STAPLE, B.D.; KIRK, H.K. Sandia National Laboratories. YAKLE, J. Science Applications International Corp. (formerly Science Applications, Inc.). October 1994. 160pp. 9411040066. 81646:030.

This report presents an application of probabilistic models and risk based criteria for determining the risk impact of the Limiting Conditions of Operations (LCOs) in the Technical Specifications (TSs) of a boiling water reactor during shutdown. This analysis studied the risk impact of the current requirements of Allowed Outage Times (AOTs) and Surveillance Test Intervals (STIs) in eight Plant Operational States (POSS) which encompass power operations, shutdown, and refueling. This report also discusses insights concerning TS action statements.

NUREG/CR-6168: DIRECT CONTAINMENT HEATING INTEGRAL EFFECTS TESTS AT 1/40 SCALE IN ZION NUCLEAR POWER PLANT GEOMETRY. BINDER, J.L.; MCUMBER, L.M.; SPENCER, B.W. Argonne National Laboratory. September 1994. 112pp. 9410120216. ANL-94/18. 81254:150.

The results of Direct Containment Heating (DCH) integral experiments are presented. The experiments simulated a high pressure melt ejection in the Zion Nuclear Power Plant. Experiments were conducted in a 1/40 scale model of the Zion containment. The model included the vessel lower head, cavity and instrument tunnel, and the lower containment structures. The

melt ejections were driven by steam. There were two main objectives of these experiments. The first was to investigate the effect of scale on DCH phenomena. The IET test series addressed this by conducting counterpart integral tests in a 1/40 scale facility at Argonne National Laboratory and in a 1/10 scale facility at Sandia National Laboratories. Iron/alumina thermite with chromium was used as a core melt simulant in the IET test series. The second objective was to address potential experiment distortions introduced by the use of non-prototypic iron/alumina thermite. The second objective was met in the U series of tests which utilized a prototypic core melt. Corium experiments were conducted that were counterpart to the IET-1RR and IET-6 iron/alumina tests.

NUREG/CR-6169: RELAY TEST PROGRAM. Series II Tests. Integral Testing Of Relays And Circuit Breakers. BANDYOPADHYAY, K.; KUNKEL, C.; SHTEYNGART, S. Brookhaven National Laboratory. February 1994. 176pp. 9403140191. BNL-NUREG-52406. 78476:001.

This report presents the results of a relay test program conducted by Brookhaven National Laboratory (BNL) under the sponsorship of the U.S. Nuclear Regulatory Commission (NRC). The program is a continuation of an earlier test program the results of which were published in NUREG/CR-4867. The current program was carried out in two phases: electrical testing and vibration testing. The objective was primarily to focus on the electrical discontinuity or continuity of relays subject to electrical pulses and vibration loads. The electrical testing was conducted by KEMA-Powertest Company and the vibration testing was performed at Wyle Laboratories, Huntsville, Alabama. This report discusses the test procedures, presents the test data, includes an analysis of the data and provides recommendations regarding reliable relay testing.

NUREG/CR-6174 V1 DFC: REVISED ANALYSES OF DECOMMISSIONING FOR THE REFERENCE BOILING WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On The Financial Assurance Requirements Of The Decommissioning Rule And.... SMITH, R.I.; BIRSCHBACH, M.; KONZEK, G.J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1994. 150pp. 9411080088. 81664:108.

With the issuance of the Decommissioning Rule (June 27, 1988), nuclear power plant licensees are required to submit to the U.S. Nuclear Regulatory Commission (NRC) for review, decommissioning plans and cost estimates. This reevaluation study provides some of the needed bases documentation to the NRC staff that will assist them in assessing the adequacy of the licensee submittals. This report presents the results of a review and reevaluation of the PNL 1980 decommissioning study of the WNP-2 nuclear power plant for the DECON, SAFSTOR, and ENTOMB decommissioning alternatives. These alternatives now include an initial 5-7 year period during which the spent fuel is stored in the spent fuel pool, prior to beginning major disassembly or extended safe storage of the plant. This report also includes consideration of the NRC requirement that decommissioning activities leading to termination of the nuclear license be completed within 60 years of final reactor shutdown, consideration of packaging and disposal requirements for Greater-Than-Class C low-level waste, and reflects all costs in 1993 dollars. Sensitivity of the total license termination cost to the disposal costs at different low-level radioactive waste disposal sites, and to different depths of contaminated concrete surface removal with the facilities are also examined.

NUREG/CR-6174 V2 DFC: REVISED ANALYSES OF DECOMMISSIONING FOR THE REFERENCE BOILING WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On The Financial Assurance Requirements Of The Decommissioning Rule And.... SMITH, R.I.; BIRSCHBACH, M.; KONZEK, G.J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1994. 300pp. 9411080106. 81664:258.

See NUREG/CR-6174, V01, DFC abstract.

NUREG/CR-6176: REVIEW OF ENVIRONMENTAL EFFECTS ON FATIGUE CRACK GROWTH OF AUSTENITIC STAINLESS STEELS. SHACK, W.J.; KASSNER, T.F. Argonne National Laboratory. May 1994. 38pp. 9406200315. ANL-94/1. 79830:287.

Fatigue and environmentally assisted cracking of piping, pressure vessel cladding, and core components in light water reactors are potential concerns to the nuclear industry and regulatory agencies. The degradation processes include intergranular stress corrosion cracking of austenitic stainless steel (SS) piping in boiling water reactors (BWRs), and propagation of fatigue or stress corrosion cracks (which initiate in sensitized SS cladding) into low-alloy ferritic steels in BWR pressure vessels. Crack growth data for wrought and cast austenitic SSs in simulated BWR water, developed at Argonne National Laboratory under U.S. Nuclear Regulatory Commission sponsorship over the past 10 years, have been compiled into a data base along with similar data obtained from the open literature. The data were analyzed to develop corrosion-fatigue curves for austenitic SSs in aqueous environments corresponding to normal BWR water chemistries, for BWRs that add hydrogen to the feedwater, and for pressurized water reactor (PWR) primary-system-coolant chemistry. The corrosion-fatigue data and curves in water were compared with the air line in Section XI of the ASME Code.

NUREG/CR-6177: ASSESSMENT OF THERMAL EMBRITTLEMENT OF CAST STAINLESS STEELS. CHOPRA, O.K.; SHACK, W.J. Argonne National Laboratory. May 1994. 35pp. 9406200312. ANL-94/2. 79830:326.

A procedure and correlations are presented for assessing thermal embrittlement and predicting Charpy-impact energy and fracture toughness J-R curve of cast stainless steel components under light water reactor operating conditions from known material information. The "saturation" impact strength and fracture toughness of a specific cast stainless steel, i.e., the minimum value that would be achieved for the material after long-term service, is estimated from the chemical composition of the steel. Fracture properties as a function of time and temperature of reactor service are estimated from the kinetics of embrittlement which are also determined from chemical composition. A common "predicted lower-bound" J-R curve for cast stainless steels of unknown chemical composition is also defined for a given grade of steel, ferrite content, and temperature. Examples of estimating fracture toughness of cast stainless steel components during reactor service are presented.

NUREG/CR-6178: LABORATORY CHARACTERIZATION OF ROCK JOINTS. HSIUNG, S.M.; KANA, D.D.; AHOLA, M.P.; et al. Center for Nuclear Waste Regulatory Analyses. May 1994. 240pp. 9407260012. CNWRA 93-013. 80342:001.

A laboratory characterization of the Apache Leap tuff joints under cyclic pseudostatic and dynamic loads has been undertaken to obtain a better understanding of dynamic joint shear behavior and to generate a complete data set that can be used for validation of existing rock-joint models. Study has indicated that available methods for determining joint roughness coefficient (JRC) significantly underestimate the roughness coefficient of the Apache Leap tuff joints, that will lead to an underestimation of the joint shear strength. The results of the direct shear tests have indicated that both under cyclic pseudostatic and dynamic loadings the joint resistance upon reverse shearing is smaller than that of forward shearing and the joint dilation resulting from forward shearing recovers during reverse shearing. Within the range of variation of shearing velocity used in these tests, the shearing velocity effect on rock-joint behavior seems to be minor, and no noticeable effect on the peak joint shear strength and the joint shear strength for the reverse shearing is observed.

NUREG/CR-6180: HYDROGEN MIXING STUDIES (HMS): USER'S MANUAL. LAM, K.L.; WILSON, T.L. Los Alamos National Laboratory. TRAVIS, J.R. Science Applications International Corp. (formerly Science Applications, Inc.). December 1994. 130pp. 9412300171. LA-12741-M. 82159:052.

Hydrogen Mixing Studies (HMS) is a best-estimate analysis tool for predicting the transport, mixing, and combustion of hydrogen and other gases in nuclear reactor containments and other facilities. It can model geometrically complex facilities having multiple compartments and internal structures. The code can simulate the effects of steam condensation, heat transfer to walls and internal structures, chemical kinetics, and fluid turbulence. The gas mixture may consist of components included in a built-in library of 20 species. This manual describes how to use the code. It explains how to set up the model geometry, define walls and obstacles, and specify gas species and material properties. Definitions of initial and boundary conditions are also described. The manual also describes various physical model and numerical procedure options, as well as how to turn them on. The reader also learns how to specify different outputs, especially graphical display of solution variables. Finally sample problems are included to illustrate some applications of the code. An input deck that illustrates the minimum required data to run HMS is given at the end of this manual.

NUREG/CR-6181: A PILOT APPLICATION OF RISK-BASED METHODS TO ESTABLISH INSERVICE INSPECTION PRIORITIES FOR NUCLEAR COMPONENTS AT SURRY UNIT 1 NUCLEAR POWER STATION. VO, T.V.; GORE, B.F.; SIMONEN, F.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1994. 72pp. 9409090115. PNL-9020. 80817:001.

As part of the NDE Reliability Program sponsored by the NRC, PNL is developing a method that uses risk-based approaches to establish inservice inspection plans for nuclear power plant components. This method uses probabilistic risk assessment (PRA) results and failure Modes and Effects Analysis techniques to identify and prioritize the most risk-important systems and components for inspection. The Surry Nuclear Power Station Unit 1 was selected for pilot applications of this method. The specific systems addressed in this report are the reactor pressure vessel, reactor coolant, low-pressure injection, and auxiliary feedwater. The results provide a risk-based ranking of components within these systems and relate the target risk to target failure probability values for individual components. These results will be used to guide the development of improved inspection plans for nuclear power plants. To develop inspection plans, the acceptable level of risk from structural failure for important systems and components will be apportioned as a small fraction of the total PRA-estimated risk for core damage. This process will determine acceptable target risk and target failure probability values for individual components. Inspection requirements will be set at levels to assure that acceptable failure probabilities are maintained.

NUREG/CR-6182 V01: OFFSCALE: A PC INPUT PROCESSOR FOR THE SCALE CODE SYSTEM. Volume 1: The CSASIN Processor For The Criticality Sequences. BOWMAN, S.M. Oak Ridge National Laboratory. November 1994. 21pp. 9412070069. ORNL/TM-12663. 81956:148.

OFFSCALE is a suite of personal computer input processor programs developed at Oak Ridge National Laboratory to provide an easy-to-use interface for modules in the SCALE-4 code system. CSASIN (formerly known as OFFSCALE) is a program in the OFFSCALE suite that serves as a user-friendly interface for the Criticality Safety Analysis Sequences (CSAS) available in SCALE-4. It is designed to assist a SCALE-4 user in preparing an input file for execution of criticality safety problems. Output from CSASIN generates an input file that may be used to execute the CSAS control module in SCALE-4. CSASIN features a pulldown menu system that accesses sophisticated data entry screens. The program allows the user to quickly set up a CSAS

input file and perform data checking. This capability increases productivity and decreases the chance of user error.

NUREG/CR-6182 V02: OFFSCALE: A PC INPUT PROCESSOR FOR THE SCALE CODE SYSTEM. Volume 2: The ORIGNATE Processor for ORIGEN-S. BOWMAN, S.M. Oak Ridge National Laboratory. November 1994. 27pp. 9412070073. ORNL/TM-12663. 81956:171.

OFFSCALE is a suite of personal computer input processor programs developed at Oak Ridge National Laboratory to provide an easy-to-use interface for modules in the SCALE-4 code system. ORIGNATE is a program in the OFFSCALE suite that serves as a user-friendly interface for the ORIGEN-S isotopic generation and depletion code. It is designed to assist an ORIGEN-S user in preparing an input file for execution of light-water-reactor (LWR) fuel depletion and decay cases. ORIGNATE generates an input file that may be used to execute ORIGEN-S in SCALE-4. ORIGNATE features a pulldown menu system that accesses sophisticated data entry screens. The program allows the user to quickly set up an ORIGEN-S input file and perform error checking. This capability increases productivity and decreases the chance of user error.

NUREG/CR-6183: PEER REVIEW OF THE TMI-2 VESSEL INVESTIGATION PROJECT METALLURGICAL EXAMINATIONS. BOHL, R.W.; GAYDOS, R.G.; VANDER VOORT, G.; et al. Argonne National Laboratory. July 1994. 45pp. 9408220048. ANL-94/3. 80637:206.

Fifteen samples recovered from the lower head of the Three Mile Island (TMI) Unit 2 nuclear reactor pressure vessel were subjected to detailed metallurgical examinations by the Idaho National Engineering Laboratory, Argonne National Laboratory (ANL) and several of the European participants. These examinations determined that a portion of the lower head reached temperatures as high as 1100 degrees C during the accident and cooled from these temperatures at ≈ 10 -100 degrees C/min. The remainder of the lower head was found to have remained below 727 degrees C. A panel of three outside peer reviewers was formed to conduct an independent review of this metallurgical analyses. This review determined that the conclusions resulting from the INEL study were fundamentally correct. In particular, the panel reaffirmed that four lower head samples attained temperatures as high as 1100 degrees C, and perhaps as high as 1150-1200 degrees C in one case, during the accident. They concluded that these samples subsequently cooled at a rate of ≈ 50 -125 degrees C/min in the temperature range of 600-400 degrees C. The reviewers also agreed that the remainder of the lower head samples had not exceeded 727 degrees C during the accident and suggested several refinements and alternative procedures that could have been employed in the original analysis.

NUREG/CR-6185: TMI-2 INSTRUMENT NOZZLE EXAMINATIONS AT ARGONNE NATIONAL LABORATORY. February 1991 - June 1993. NEIMARK, L.A.; SHEARER, T.L.; PUROHIT, A.; et al. Argonne National Laboratory. March 1994. 158pp. 9405050325. ANL-94/5. 79162:155.

Six of the 14 instrument-penetration-tube nozzles removed from the lower head of TMI-2 were examined to identify damage mechanisms, provide insight to the fuel relocation scenario, and provide input data to the margin-to-failure analysis. Visual inspection, gamma scanning, metallography, microhardness measurements, and scanning electron microscopy were used to obtain the desired information. The results showed varying degrees of damage to the lower head nozzles, from $\approx 50\%$ melt-off to no damage at all to near-neighbor nozzles. The elevations of nozzle damage suggested that the lower elevations (near the lower head) were protected from molten fuel, apparently by an insulating layer of fuel debris. The pattern of nozzle damage was consistent with fuel movement toward the hot-spot location identified in the vessel wall. Evidence was found for the exist-

ence of a significant quantity of control assembly debris on the lower head before the massive relocation of fuel occurred.

NUREG/CR-6187: RESULTS OF MECHANICAL TESTS AND SUPPLEMENTARY MICROSTRUCTURAL EXAMINATIONS OF THE TMI-2 LOWER HEAD SAMPLES. DIERCKS, D.R.; NEIMARK, L.A. Argonne National Laboratory. March 1994. 143pp. 9405040214. ANL-94/8. 79104:205.

Metallographic examinations of 15 samples from the lower head of the TMI-2 pressure vessel confirmed that four samples attained temperatures as high as 1100 degrees C during the accident and cooled at ≈ 10 -100 degrees C/min. Portions of two adjacent samples, and possibly a third sample away from the hot spot, also exceeded 727 degrees C. Results from tensile tests conducted on this material at 600-1200 degrees C generally agreed well with literature data on A533, Grade B steel. The material from the hot spot exhibited higher strengths than the remaining material, reflecting the heat treatment received during the accident. Charpy V-notch impact tests similarly found significantly lower upper shelf energies and higher transition temperatures for the material from the hot spot. However, creep tests conducted at ≈ 600 -1200 degrees C revealed little difference between material at and away from the hot spot. Cracks were found in the stainless steel cladding of boat samples from the hot spot. The cracks appeared to be the result of hot-tearing, probably assisted by intergranular penetration of liquid Ag-Cd. Crack propagation into the A533 vessel steel was a maximum of ≈ 6 mm. Materials in the cracks suggest the presence of control-assembly debris on the lower head before the massive fuel flow arrived.

NUREG/CR-6188 V01: MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE. Annual Report For FY 1993. ROGERS, R.D.; HAMILTON, M.A.; VEEH, R.H.; et al. EG&G Idaho, Inc. April 1994. 63pp. 9405040133. EGG-2730. 79103:259.

The Nuclear Regulatory Commission stipulates that disposed low-level radioactive waste (LLW) be stabilized. Because of apparent ease of use and normal structural integrity, cement has been widely used as a binder to solidify LLW. However, the resulting waste forms are sometimes susceptible to failure due to the actions of waste constituents, stress, and environment. This report reviews laboratory efforts that are being developed to address the effects of microbiologically influenced chemical attack on cement-solidified LLW. Groups of microorganisms are being employed that are capable of metabolically converting organic and inorganic substrates into organic and mineral acids. Such acids aggressively react with cement and can ultimately lead to structural failure. Results on the application of mechanisms inherent in microbially influenced degradation of cement-based material are the focus of this report. Sufficient data-validated evidence of the potential for microbially influenced deterioration of cement-solidified LLW has been developed during the course of this study. These data support the continued development of appropriate tests necessary to determine the resistance of cement-solidified LLW to microbially induced degradation that could impact the stability of the waste form. They also justify the continued effort of enumeration of the conditions necessary to support the microbiological growth and population expansion.

NUREG/CR-6190 V01: PROTECTION AGAINST MALEVOLENT USE OF VEHICLES AT NUCLEAR POWER PLANTS. Vehicle Barrier System Siting Guidance For Blast Protection. NEBUDA, D.T. Army, Dept. of, Corps of Engineers. August 1994. 18pp. 9409200299. 80954:218.

This manual provides for determining the minimum safe standoff distance between vital safety related equipment and the design basis vehicle bomb threat adopted by the U.S. Nuclear Regulatory Commission. Vital safety related equipment should survive the design basis vehicle bomb attack when the minimum safe standoff distance is provided. Guidance is provided for exposed vital safety related equipment and for equipment housed within vital area barriers. The types of vital area barriers

addressed are 12-, 18-, 24-, and 30 inch-thick reinforced concrete slabs with reinforcing ratios of 0.2, 0.4, 0.6, 0.8, 1.0 percent.

NUREG/CR-6190 V01 R1: PROTECTION AGAINST MALEVOLENT USE OF VEHICLES AT NUCLEAR POWER PLANTS. Vehicle Barrier System Selection Guidance For Blast Protection. NEBUDA, D.T. Army, Dept. of, Corps of Engineers. December 1994. 20pp. 9501180290. 82341:256.

This manual provides for determining the minimum safe standoff distance between vital safety related equipment and the design basis vehicle bomb threat adopted by the U.S. Nuclear Regulatory Commission. Vital safety related equipment should survive the design basis vehicle bomb attack when the minimum safe standoff distance is provided. Guidance is provided for exposed vital safety related equipment and for equipment housed within vital area barriers.

NUREG/CR-6190 V02: PROTECTION AGAINST MALEVOLENT USE OF VEHICLES AT NUCLEAR POWER PLANTS. Vehicle Barrier System Siting Guidance For Blast Protection. NEBUDA, D.T. Army, Dept. of, Corps of Engineers. August 1994. 41pp. 9409200303. 80954:236.

See NUREG/CR-6190, V01 abstract.

NUREG/CR-6190 V02 R1: PROTECTION AGAINST MALEVOLENT USE OF VEHICLES AT NUCLEAR POWER PLANTS. Vehicle Barrier System Selection Guidance. NEBUDA, D.T. Army, Dept. of, Corps of Engineers. December 1994. 41pp. 9501180287. 82341:215.

This manual provides a simplified procedure for selecting land vehicle barriers that will stop the design basis vehicle threat adopted by the U. S. Nuclear Regulatory Commission. Proper selection and construction of vehicle barriers should prevent intrusion of the design basis vehicle. In addition, vital safety related equipment should survive a design basis vehicle bomb attack when vehicle barriers are properly selected, sited, and constructed. This manual addresses passive vehicle barriers, active vehicle barriers, and site design features that can be used to reduce vehicle impact velocity.

NUREG/CR-6193: PRIMARY SYSTEM FISSION PRODUCT RELEASE AND TRANSPORT. A State-Of-The-Art Report To The Committee On The Safety Of Nuclear Installations. WRIGHT, A.L. Oak Ridge National Laboratory. June 1994. 242pp. 9407010317. NEA/CSNI/R(94)2. 80031:001.

This report presents a summary of the status of research activities associated with fission product behavior (release and transport) under severe accident conditions within the primary systems of water-moderated and water-cooled nuclear reactors. For each of the areas of fission product release and fission product transport, the report summarizes relevant information on important phenomena, major experiments performed, relevant computer models and codes, comparisons of computer code calculations with experimental results, and general conclusions on the overall state of the art. Finally, the report provides an assessment of the overall importance and knowledge of primary system release and transport phenomena and presents major conclusions on the state of the art.

NUREG/CR-6194: METALLOGRAPHIC AND HARDNESS EXAMINATIONS OF TMI-2 LOWER PRESSURE VESSEL HEAD SAMPLES. KORTH, G.E. EG&G Idaho, Inc. March 1994. 124pp. 9404110355. TMI V(92)EG01. 78817:224.

Fifteen steel samples were removed from the lower pressure vessel head of the damaged TMI-2 nuclear reactor to assess the thermal threat to the head posed by 15 to 20 metric tons of molten core debris relocating there during the accident. Full sections of thirteen of the samples and partial sections of the other two samples underwent hardness and metallographic examinations at the Idaho National Engineering Laboratory. These examinations have shown that eleven of the fifteen samples did not exceed the ferrite-austenite transformation temperature of

727 degrees C during the accident. The remaining four samples did show evidence of having a much more severe thermal history. The samples from core grid positions F-10 and G-8 are believed to have experienced temperatures of 1,040 to 1,060 degrees C for about 30 minutes. Samples from positions E-8 and E-6 appear to have been subjected to 1,075 to 1,100 degrees C for approximately 30 minutes.

NUREG/CR-6195: EXAMINATION OF RELOCATED FUEL DEBRIS ADJACENT TO THE LOWER HEAD OF THE TMI-2 REACTOR VESSEL. AKERS, D.W.; JENSEN, S.M.; SCHUETZ, B.K. EG&G Idaho, Inc. March 1994. 143pp. 9405050363. TMI V(92)EG10. 79193:304.

As part of the Three Mile Island Unit 2 (TMI-2) Vessel Investigation Project, funded by the Organization for Economic Cooperation and Development, physical, metallurgical, and radiochemical examinations were performed on samples of previously molten material that had relocated to the lower plenum of the TMI-2 reactor during the accident of 28 March 1979. This report presents the results of those examinations and some limited analysis of these results as required for the interpretation of the data. Principal conclusions of the examinations are that the bulk lower head debris is homogeneous and composed primarily of (U,Zr)O₂. This molten material reached temperatures greater than 2,600 degrees C and probably reached the lower head as a liquid or slurry at temperatures below the peak temperature. A debris bed was formed, which was composed of particulate debris above a monolithic melt that solidified on the lower head.

NUREG/CR-6196: CALCULATIONS TO ESTIMATE THE MARGIN TO FAILURE IN THE TMI-2 VESSEL. STICKLER, L.A.; REMPE, J.L.; CHAVEZ, S.A.; et al. EG&G Idaho, Inc. March 1994. 300pp. 9404060281. TMI V(93)EG01. 78759:001.

As part of the OECD-sponsored Three Mile Island Unit 2 (TMI-2) Vessel Investigation Project (VIP), margin-to-failure calculations for mechanisms having the potential to threaten the integrity of the vessel were performed to improve understanding of events that occurred during the TMI-2 accident. Analyses considered four failure mechanisms: tube rupture, tube ejection, global vessel failure, and localized vessel failure. Computational input was based on data from the TMI-2 VIP examinations of the vessel steel samples, the instrument tube nozzles, and samples of the hard layer of debris found on the TMI-2 vessel lower head. Sensitivity studies were performed to investigate the uncertainties in key parameters for these analyses.

NUREG/CR-6197: TMI-2 VESSEL INVESTIGATION PROJECT INTEGRATION REPORT. WOLF, J.R.; REMPE, J.L.; STICKLER, L.A.; et al. EG&G Idaho, Inc. March 1994. 168pp. 9405050346. TMI V(93)EG10. 79164:142.

The Three Mile Island Unit 2 (TMI-2) Vessel Investigation Project (VIP) was an international effort that was sponsored by the Nuclear Energy Agency of the Organization for Economic Cooperation and Development. The primary objectives of the VIP were to extract and examine samples from the lower head and to evaluate the potential modes of failure and the margin of structural integrity that remained in the TMI-2 reactor vessel during the accident. This report presents a summary of the major findings and conclusions that were developed from research during the VIP. Results from the various elements of the project are integrated to form a cohesive understanding of the vessel's condition after the accident.

NUREG/CR-6198: TMI-2 INSTRUMENT NOZZLE EXAMINATIONS PERFORMED AT THE INEL. AKERS, D.W.; SCHUETZ, B.K. EG&G Idaho, Inc. March 1994. 130pp. 9404010249. EGG-2735. 78728:034.

As part of the Three Mile Island Unit 2 (TMI-2) Vessel Investigation Project, under the auspices of the Organization for Economic Cooperation and Development, examinations were performed at the Idaho National Engineering Laboratory on eight nozzles and one guide tube from the TMI-2 reactor vessel. This document describes the examination methodology, summarizes the examination results, and presents interpretations of the re-

sults as they relate to the damage to the reactor vessel and to the development of a core relocation scenario. Not all examinations originally proposed as part of this program were completed due to facility problems at the INEL. Consequently, only the results of completed aspects of the examination program are presented.

NUREG/CR-6200: UNCERTAINTY ANALYSIS OF SUPPRESSION POOL HEATING DURING AN ATWS IN A BWR-5 PLANT. An Application Of The CSAU Methodology Using The BNL Engineering Plant Analyzer. WULFF, W.; CHENG, H.S.; MALLIN, A.N.; et al. Brookhaven National Laboratory. March 1994. 62pp. 9405040189. BNL-NUREG-52412. 79104:144.

The uncertainty has been estimated of predicting the peak temperature in the suppression pool of a BWR power plant, which undergoes an NRC-postulated Anticipated Transient Without Scram (ATWS). The ATWS is initiated by recirculation pump trips and then leads to power and flow oscillations as they had occurred at the LaSalle-2 Power Station in March of 1988. After limit-cycle oscillations have been established, the turbines are tripped, but without MSIV closure, allowing steam discharge through the turbine bypass into the condenser. Postulated operator actions, namely to lower the reactor vessel pressure and the level elevation in the downcomer, are simulated by a robot model, which accounts for operator uncertainty. All balance of plant and control systems modeling uncertainties were part of the statistical uncertainty analysis, that was patterned after the Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology. The analysis showed that the predicted suppression pool peak temperature of 329.3K (133 degrees F) has a 95-percentile uncertainty of 14.4K (26 degrees F), and that the size of this uncertainty bracket is dominated by the experimental uncertainty of measuring Safety and Relief Valve mass flow rates under critical flow conditions. The analysis showed also, that the probability of exceeding the suppression pool temperature limit of 352.6K (175 degrees F) is most likely zero (it is estimated as $< 5 \cdot 10^{-4}$). The square root of the sum of the squares of all the computed peak pool temperatures is 350.7K (171.6 degrees F).

NUREG/CR-6201: COMPRESSION AND IMMERSION TESTS AND LEACHING OF RADIONUCLIDES, STABLE METALS, AND CHELATING AGENTS FROM CEMENT-SOLIDIFIED DECONTAMINATION WASTE COLLECTED FROM NUCLEAR POWER STATIONS. AKERS, D.W.; KRAFT, N.C.; MANDLER, J.W. EG&G Idaho, Inc. June 1994. 250pp. 9407070300. EGG-2736. 80113:304.

A study was performed for the Nuclear Regulatory Commission (NRC) to evaluate structural stability and leachability of radionuclides, stable metals, and chelating agents from cement-solidified decontamination ion-exchange resin wastes collected from seven commercial boiling water reactors and one pressurized water reactor. The decontamination methods used at the reactors were the Can-Decon, AP/Citrox, Dow NS-1, and LOMI processes. Samples of untreated resin waste and solidified waste forms were subjected to immersion and compressive strength testing. Some waste-form samples were leach-tested using simulated groundwaters and simulated seawater for comparison with the deionized water tests that are normally performed to assess waste-form leachability. This report presents the results of these tests and assesses the effects of the various decontamination methods, waste form formulations, leachant chemical compositions, and pH of the leachant on the structural stability and leachability of the waste forms. Results indicate that releases from intact and degraded waste forms are similar and that the behavior of some radionuclides such as (55)Fe, (60)Co, and (99)Tc were similar. In addition, the leachability indexes are greater than 6.0, which meets the requirement in the NRC's "Technical Position on Waste Form," Revision 1.

NUREG/CR-6203: VALIDATION STUDIES FOR ASSESSING UNSATURATED FLOW AND TRANSPORT THROUGH FRACTURED ROCK. BASSETT,R.L.; NEUMAN,S.P.; RASMUSSEN,T.C.; et al. Arizona, Univ. of, Tucson, AZ. August 1994. 219pp. 9409090105. 80816:111.

The objectives of this contract are to examine hypotheses and conceptual models concerning unsaturated flow and transport through heterogeneous fractured rock and to design and execute confirmatory field and laboratory experiments to test these hypotheses and conceptual models. Important new information is presented such as the application and evaluation of procedures for estimating hydraulic, pneumatic, and solute transport coefficients for a range of thermal regimes. A field heater experiment was designed that focused on identifying the suitability of existing monitoring equipment to obtain required data. A reliable method was developed for conducting and interpreting tests for air permeability using a straddle-packer arrangement. Detailed studies of fracture flow from Queen Creek into the Magma Copper Company ore haulage tunnel have been initiated. These studies will provide data on travel time for transport of water and solute in unsaturated tuff. The collection of rainfall, runoff, and infiltration data at two small watersheds at the Apache Leap Tuff Site enabled us to evaluate the quantity and rate of water infiltrating into the subsurface via either fractures or matrix. Characterization methods for hydraulic parameters relevant to high-level waste transport, including fracture apertures, transmissivity, matrix porosity, and fracture wetting front propagation velocities, were developed.

NUREG/CR-6204: QUESTIONS AND ANSWERS BASED ON REVISED 10 CFR PART 20. BORGES,T.; STAFFORD,R.S.; LU,P.Y.; et al. Oak Ridge National Laboratory. May 1994. 111pp. 9405310172. ORNL/TM-12690. 79590:226.

NUREG/CR-6204 is a collection of questions and answers that were originally issued in seven sets and which pertain to revised 10 CFR Part 20. The questions came from both outside and within the NRC. The answers were compiled and provided by NRC staff within the offices of Nuclear Reactor Regulation, Nuclear Material Safety and Safeguards, Nuclear Regulatory Research, the Office of State Programs, and the five regional offices. Although all of the questions and answers have been reviewed by attorneys in the NRC Office of the General Counsel, they do not constitute official legal interpretations relevant to revised 10 CFR Part 20. The questions and answers do, however, reflect NRC staff decisions and technical options on aspects of the revised 10 CFR Part 20 regulatory requirements. This NUREG is being made available to encourage communication among the public, industry, and NRC staff concerning the major revisions of the NRC's standards for protection against radiation.

NUREG/CR-6205: VALVE ACTUATOR MOTOR DEGRADATION. KUECK,J.D. Oak Ridge National Laboratory. December 1994. 41pp. 9501180329. ORNL-6796. 82347:191.

Valve actuator motor degradation and failure has been a significant, but little studied, problem in the nuclear industry. This study provides a discussion of the primary failure mode - thermal degradation - and reviews the basis for the solution of thermal degradation - thermal protection. The study also provides reviews of various industry data bases, discusses effects of other failure modes such as corrosion, and provides a review of other considerations the user should entertain when assessing thermal protection.

NUREG/CR-6206: TRANSPORT CALCULATIONS OF RADIATION EXPOSURE TO VESSEL SUPPORT STRUCTURES IN THE TROJAN REACTOR. ASGARIM.; WILLIAMS,M.L. Louisiana State Univ., Baton Rouge, LA. KAM,F.B.K.; et al. Oak Ridge National Laboratory. July 1994. 74pp. 9412020031. ORNL/TM-12693. 81941:001.

Comparisons of transport calculations of the dosimeter activities with the experimental measurements shows that the values obtained with ENDF/B-VI cross-section data overestimate the

measured results for high-energy-threshold reactions in the cavity by up to 41% and thermal reactions by up to a factor of 3.0. The transport calculations performed with the original SAILOR cross-section library (based on ENDF/B-IV data) overestimate measured threshold reactions by only 15% and the thermal reactions by about a factor of 2.50. These results are inconsistent with those obtained in earlier studies that compared transport calculations done with SAILOR vs ENDF/B-VI, which indicate that SAILOR tends to underestimate cavity activities for threshold reactions, while the ENDF/B-VI values usually agree better with experimental results. One factor that probably contributes to the rather large discrepancy between the computed and measured activities is the core power distribution used in the calculations. Because of unavailability of plant-specific data, a generic power distribution provided by Westinghouse was used. Since the calculated cavity flux levels appear to be over-estimated, the results estimated for the exposure to the support structure should be conservative.

NUREG/CR-6208: AN EMPIRICAL INVESTIGATION OF OPERATOR PERFORMANCE IN COGNITIVELY DEMANDING SIMULATED EMERGENCIES. ROTH,E.M.; MUMAW,R.J. Westinghouse Electric Corp. LEWIS,P.M. Human Factors Branch (880717-941217). July 1994. 143pp. 9407250208. 80328:005.

This report documents the results of an empirical study of nuclear power plant operator performance in cognitively demanding simulated emergencies. During emergencies operators follow highly prescriptive written procedures. The objectives of the study were to understand and document what role higher-level cognitive activities such as diagnosis, or more generally 'situation assessment', play in guiding operator performance, given that operators utilize procedures in responding to the events. The study examined crew performance in variants of two emergencies: (1) an Interfacing System Loss of Coolant Accident and (2) a Loss of Heat Sink scenario. Data on operator performance were collected using training simulators at two plant sites. Up to 11 crews from each plant participated in each of two simulated emergencies for a total of 38 cases. Crew performance was videotaped and partial transcripts were produced and analyzed. The results revealed a number of instances where higher-level cognitive activities such as situation assessment and response planning enabled crews to handle aspects of the situation that were not fully addressed by the procedures. This report documents these cases and discusses their implications for the development and evaluation of training and control room aids, as well as for human reliability analyses.

NUREG/CR-6209: MEMPHIS AREA REGIONAL SEISMIC NETWORK. Final Report, October 1986 - September 1992. CHIU,J.M.; JOHNSTON,A.C. Memphis State Univ., Memphis, TN. March 1994. 69pp. 9404110348. 78831:143.

The Memphis Area Regional Seismic Network (MARSN) has provided important southern coverage of the New Madrid seismic zone (NMSZ). The network helped to identify the Crittenden County fault zone located east of and parallel to the SW segment of the NMSZ. MARSN data has also been added into a database obtained by the PANDA experiment for a comprehensive seismological study of the NMSZ. Results demonstrate that earthquakes in the NMSZ are mainly confined within depths ranging from 3 to 15km. The SW, NW, and NE segments of the NMSZ are characterized by narrow vertical strike-slip faults. The central NMSZ, however, is very complicated. The northcentral section is characterized by a well-defined planar feature dipping ~31 degrees SW which shows dominantly normal faulting. The southcentral section shows a ~48 degrees SW dipping fault with dominantly reverse faulting. Although the E-W regional stress may play an important role in fault movements in the NMSZ, the complication in focal mechanisms suggests that other factors including postseismic relaxation by the 1811-1812 earthquakes, interactions between adjacent fault segments, or features such as the right-lateral strike-slip Crittenden County

fault cannot be overlooked in future tectonic studies of the NMSZ.

NUREG/CR-6211: INTEGRATED FUEL-COOLANT INTERACTION (IFCI 6.0) CODE. User's Manual. DAVIS, F.J.; YOUNG, M.F. Sandia National Laboratories. April 1994. 68pp. 9405310193. SAND94-0406. 79590:044.

The Integrated Fuel-Coolant Interaction (IFCI) computer code is being developed at Sandia National Laboratories to investigate the fuel-coolant interaction (FCI) problem at large scale using a two-dimensional, four-field hydrodynamic framework and physically based models. IFCI will be capable of treating all major FCI processes in an integrated manner. This document is a product of the effort to generate a stand-alone version of IFCI, IFCI 6.0. The User's Manual describes in detail the hydrodynamic method and physical models used in IFCI 6.0. Appendix A is an input manual, provided for the creation of working decks.

NUREG/CR-6212: VALUE OF PUBLIC HEALTH AND SAFETY ACTIONS AND RADIATION DOSE AVOIDED. BAUM, J.W. Brookhaven National Laboratory. May 1994. 54pp. 9407260025. BNL-NUREG-52413. 80342:241.

The values judged best to reflect the willingness of society to pay for the avoidance or reduction of risk were deduced from studies of costs of health care, transportation safety, consumer product safety, government agency actions, wage-risk compensation, consumer behavior (market) studies, and willingness-to-pay surveys. The results ranged from \$1,400,000 to \$2,700,000 per life saved. Applying the mean of these values (\$2,100,000) and the latest risk per unit dose coefficients used by the ICRP (1991), which take into account risks to the general public, including genetic effects and non-fatal cancers, yields a value of dose avoided of \$750 to \$1,500 per person-cSv for public exposures. The lower value applies if adjustments are made for years of life lost per fatality. A nominal value of \$1,000 per person-cSv seems appropriate in light of the many uncertainties involved in deducing these values. These values are consistent with values recommended by several European countries for individual doses in the region of 1 mSv/y (100 mrem/y). Below this dose rate, most countries have values a factor of 7 to 10 lower, based on the assumption that society is less concerned with fatality risks below about 10⁻⁴/y.

NUREG/CR-6213: HIGH-TEMPERATURE HYDROGEN-AIR-STEAM DETONATION EXPERIMENTS IN THE BNL SMALL-SCALE DEVELOPMENT APPARATUS. CICCARELLI, G.; GINSBURG, T.; BOCCIO, J.; et al. Brookhaven National Laboratory. August 1994. 99pp. 9409230300. BNL-NUREG-52414. 81011:001.

The Small-Scale Development Apparatus (SSDA) was constructed to provide a preliminary set of experimental data to characterize the effect of temperature on the ability of hydrogen-air-steam mixtures to undergo detonations and, equally important, to support design of the larger scale High-Temperature Combustion Facility (HTCF) by providing a test bed for solution of a number of high-temperature design and operational problems. The SSDA, the central element of which is a 10-cm inside diameter, 6.1-m long tubular test vessel designed to permit detonation experiments at temperatures up to 700K, was employed to study self-sustained detonations in gaseous mixtures of hydrogen, air, and steam at temperatures between 300K and 650K at a fixed initial pressure of 0.1 MPa. Detonation cell size measurements provide clear evidence that the effect of hydrogen-air gas mixture temperature, in the range 300K-650K, is to decrease cell size and, hence to increase the sensitivity of the mixture to undergo detonations. The effect of steam content, at any given temperature, is to increase the cell size and, thereby, to decrease the sensitivity of stoichiometric hydrogen-air mixtures. The hydrogen-air detonability limits for the 10-cm inside diameter SSDA test vessel, based upon the onset of single-head spin, decreased from 15 percent hydrogen at 300K down to between 9 and 10 percent hydrogen at 650K.

NUREG/CR-6216: EVALUATION OF ROCK JOINT MODELS AND COMPUTER CODE UDEC AGAINST EXPERIMENTAL RESULTS. HSIUNG, S.M.; GHOSH, A.; CHOWDHURY, A.H.; et al. Center for Nuclear Waste Regulatory Analyses. November 1994. 116pp. 9411280288. CNWRA 93-024. 81851:276.

The Mohr-Coulomb, Barton-Bandis, and Continuously-Yielding rock joint models and their numerical implementation in the UDEC code were evaluated for their ability to simulate joint behavior under cyclic pseudostatic and dynamic loading conditions. Some deficiencies of these joint models and their implementation in UDEC were identified. These deficiencies include that the rock joint models under evaluation may not be able to sufficiently predict the joint shear and dilation behavior during reverse joint shearing. Both joint forward and reverse shearing are important phenomena of a rock joint behavior. Reverse shearing can result from earthquake, thermal load, or both - all of which are expected to be experienced during the life of a high-level waste repository. These deficiencies could result in an overestimation of the stability of emplacement drifts and emplacement boreholes and prediction of incorrect near-field flow pattern (including preferential pathways for water and gas).

NUREG/CR-6218: A REVIEW OF THE TECHNICAL ISSUES OF AIR INGRESSION DURING SEVERE REACTOR ACCIDENTS. POWERS, D.A.; KMETYK, L.N.; SCHMIDT, R.C. Sandia National Laboratories. September 1994. 84pp. 9409270204. SAND94-0731. 81046:023.

Severe reactor accident scenarios involving air ingress into the reactor coolant system are described. Evidence from modern reactor accident analyses and from the accident at Three Mile Island shows residual fuel will be present in the core region when air ingress is possible. This residual fuel can interact with the air. Exploratory calculations with the MELCOR code of station blackout accidents during shutdown conditions and during operations are used to examine clad oxidation by air and ruthenium release from fuel in air. Extensive ruthenium release is predicted when air ingress rates exceed about 10 moles/s. Past studies of air interactions with irradiated reactor fuel are reviewed. Effects air ingress may have on fission product release, transport, deposition, and revaporization are discussed. Perhaps the most important effects of air ingress are expected to be enhanced release of ruthenium from the fuel and the formation of copious amounts of aerosol from uranium oxide vapors. Revaporization of iodine and tellurium retained in the reactor coolant system might be expected.

NUREG/CR-6221: THE VALLES NATURAL ANALOGUE PROJECT. STOCKMAN, H.W.; KRUMHANS, J.L.; HO, C.K.; et al. Sandia National Laboratories. December 1994. 126pp. 9412300204. SAND94-0650. 82162:197.

The contact between an obsidian flow and a steep-walled tuff canyon was examined as an analogue for a high-level waste repository. The analogue site is located in the Valles Caldera in New Mexico, where a massive obsidian flow filled a paleocanyon in the Battleship Rock tuff. The obsidian flow provided a heat source, analogous to waste panels or an igneous intrusion in a repository, and caused evaporation and migration of water. The tuff and obsidian samples were analyzed for major and trace elements and mineralogy by INAA, XRF, X-ray diffraction, and scanning electron microscopy and electron microprobe. Samples were also analyzed for D/H and (39)Ar/(40)Ar isotopic composition. Overall, the effects of the heating event seem to have been slight and limited to the tuff nearest the contact. There is some evidence of devitrification and migration of volatiles in the tuff within 10 meters of the contact, but variations in major and trace element chemistry are small and difficult to distinguish from the natural (pre-heating) variability of the rocks.

NUREG/CR-6223: REVIEW OF THE PROPOSED MATERIALS OF CONSTRUCTION FOR THE SBWR AND AP600 ADVANCED REACTORS. DIERCKS,D.R.; SHACK,W.J.; CHUNG,H.M.; et al. Argonne National Laboratory. May 1994. 113pp. 9406210280. ANL-94/13. 79836:215.

The General Electric Simplified Boiling Water Reactor (SBWR) and the Westinghouse Advanced Passive 600 MWe Reactor (AP600) have been reviewed in detail by Argonne National Laboratory. The objectives of these reviews were to (a) evaluate proposed advanced-reactor designs and the materials of construction for the safety systems, (b) identify all aging and environmentally related degradation mechanisms for the materials of construction, and (c) evaluate from the safety viewpoint the suitability of the proposed materials for the design application. The materials selected for both reactors were generally sound, and no major selection errors were found. It was apparent that considerable thought had been given to the materials selection process, making use of lessons learned from previous LWR experience. The review resulted in the suggestion of alternate and possibly better materials choices in a number of cases, and several potential problem areas have been cited. The review of the AP600 materials of construction was impaired by the fact that the materials designations given in the Standard Safety Analysis Report (SSAR) were often too vague to identify the specific alloy to be used. The SSAR for the SBWR generally gave more detailed materials information.

NUREG/CR-6224 DFC: PARAMETRIC STUDY OF THE POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS. Draft For Comment. ZIGLER,G.; BEATY,R.; BRIDEAU,J.; et al. Science & Engineering Associates, Inc. August 1994. 238pp. 9408310226. SEA93-554-06-A1. 80719:001.

This report documents a plant-specific study for a BWR 4 with Mark I containment, that evaluated LOCA-generated debris phenomena and the potential for losing recirculation cooling capability due to ECCS pump suction strainer clogging and loss of net positive suction head (NPSH) margin. The major elements of the study were: (1) acquisition of data from a reference BWR; (2) analysis of weld failure frequencies to estimate the LOCA frequency; (3) development of BWR debris generation and debris transport models; (4) modeling debris transport in the suppression pool; (5) estimation of strainer blockage frequency and loss of NPSH margin; and (6) estimation of core damage frequency attributable to a loss of ECCS following a large LOCA. A point estimate overall DEGB pipe break frequency (per Rx-year) of $1.6E-04$ was calculated with a corresponding overall DCCS loss of NPSH frequency (per Rx-year) of $1.5E-04$. The point estimate of core damage frequency (per Rx-year) due to blockage-related accident sequences for the reference plant ranged from $4.2E-06$ to $2.5E-05$. The results of this study show that severe strainer blockage and loss of NPSH can occur within the first 30 minutes of the LOCA if particulates are present in addition to the LOCA-generated debris. This study illustrates the competing effects of debris settling versus turbulence effects, which can take place in the suppression pool.

NUREG/CR-6226: EFFECT OF DYNAMIC STRAIN AGING ON THE STRENGTH AND TOUGHNESS OF NUCLEAR FERRITIC PIPING AT LWR TEMPERATURES. MARSCHALL,C.W.; MOHAN,R.; KRISHNASWAMY,P.; et al. Battelle Memorial Institute, Columbus Laboratories. October 1994. 102pp. 9411160005. BMI-2176. 81756:201.

This topical report is on the phenomenon of dynamic strain aging (DSA) in ferritic nuclear piping steels and its effect on fracture at LWR temperatures. The report was a deliverable from the U.S. NRC's program entitled "Short Cracks in Piping and Piping Welds". The objective of this work was to predict the occurrence of and evaluate the effects of ductile crack instabilities, which occur frequently in ferritic steel pipe fracture tests at 288 degrees C (550 degrees F), and are believed to be due to dynamic strain aging. Numerous laboratory tests and one numerical simulation of a C(T) test with crack instabilities were un-

dertaken. Virtually every U.S. nuclear pipe ferritic steel grade tested had pipe lengths that were susceptible to DSA and unstable crack jumps at 288 degrees C (550 degrees F). Data from specimens and pipe tests showed that the instabilities are random and the number of or precise locations of crack jumps cannot be predicted accurately, however the susceptibility of a steel to crack jumps occurring can be predicted. A screening criteria based on the ratio of the hardness at operating versus room temperature was developed to make these predictions. Since experiments at LWR temperatures on U.S. ferritic nuclear pipe steel have been shown that large crack jumps occurred in pipe and laboratory experiments, ferritic steels that are highly sensitive to crack jumps, should be used with caution. The DSA hardness screening criterion is relatively simple to implement and could be used in situ on existing plant piping systems, or as a mill quality control test when purchasing new materials. The data suggest that it may be possible to manufacture ferritic steel pipe that is less susceptible to DSA, which would be beneficial for future advanced light water reactor plants or replacement piping in existing plants.

NUREG/CR-6228: PRELIMINARY ASSESSMENT OF THE FRACTURE BEHAVIOR OF WELD MATERIAL IN FULL-THICKNESS CLAD BEAMS. KEENEY,J.A.; BASS,B.R.; MCAFEE,W.J.; et al. Oak Ridge National Laboratory. October 1994. 64pp. 9411040076. ORNL/TM-12735. 81646:193.

A testing program is described that utilizes full-thickness clad beam specimens to quantify fracture toughness for shallow cracks in material for which metallurgical conditions are prototypic of those found in reactor pressure vessels (RPVs). The beam specimens are fabricated from a section of an RPV wall (removed from a canceled nuclear plant) that includes weld, plate and clad material. Metallurgical factors potentially influencing fracture toughness for shallow cracks in the beam specimens include material gradients due to welding and cladding applications, as well as material inhomogeneities in welded regions due to reheating in multiple weld passes. Summaries of the testing program and analytical studies are provided. Fracture toughness data from the three clad beam specimens are compared with other shallow- and deep-crack uniaxial beam and cruciform data generated previously from A 533 Grade B plate material. Difficulties with interpreting lower-bound fracture toughness curves constructed from the shallow-crack data are essentially resolved by adopting a single normalizing temperature parameter, namely, the nil-ductility temperature (NDT). When normalized to NDT, the combined shallow-crack data base from the plate and weld material exhibit an elevated mean fracture toughness in the transition temperature region, accompanied by increased data scatter that tends toward the same lower bound associated with highly-constrained deep-crack data. Additional full-thickness clad beam testing is proposed to complete the investigation of shallow-crack fracture toughness behavior in prototypic weld and plate material. This report is designated as HSST Report No. 143.

NUREG/CR-6231: A COMPARISON OF THE RELATIVE IMPORTANCE OF COPPER PRECIPITATES AND POINT DEFECT CLUSTERS IN REACTOR PRESSURE VESSEL EMBRITTLEMENT. STOLLER,R.E. Oak Ridge National Laboratory. December 1994. 52pp. 9501180295. ORNL-6811. 82341:276.

Embrittlement of reactor pressure vessel (RPV) steels is believed to arise primarily from matrix hardening due to the formation of radiation-induced point defect clusters (PDC) and radiation-enhanced, copper-rich precipitates (CRP). A model has been developed to investigate the relative importance of PDC and CRP in RPV embrittlement. The model has been used to determine the influence of a range of irradiation and material parameters on the predicted change in yield strength. The results indicate that there are temperature and displacement rate regimes wherein either CRP or PDC can dominate the material's response to irradiation. Both interstitial and vacancy type defects contribute to the PDC component. The different de-

dependencies of the CRP and PDC on temperature and displacement rate indicate that simple data extrapolations could lead to poor predictions of RPV embrittlement. Fortunately, the yield strength changes predicted by the composite PDC/CRP model exhibit very little dependence on displacement rate below about 5×10^{-9} dpa/s. If this result is confirmed, concerns about accelerated displacement rates in power reactor surveillance programs should be minimized. The sensitivity of the model to microstructural parameters highlights the need for more detailed characterization of RPV steels.

NUREG/CR-6232: ASSESSING THE ENVIRONMENTAL AVAILABILITY OF URANIUM IN SOILS AND SEDIMENTS. AMONETTE, J.E.; HOLDREN, G.R.; KRUPA, K.M.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. June 1994. 116pp. 9406290309. PNL-9750. 80014:097.

This report reviews existing approaches to determine their potential for assessing the environmental availability (i.e., aqueous solubility) of uranium in bulk soils or sediments. Direct empirical approaches that involve extractions by aqueous solutions provide estimates of the solubility of operationally defined components. Alternately, indirect approaches based on advanced instrumental techniques can be used to identify specific forms of uranium; the contribution of each form to the solubility is then inferred from geochemical equilibrium and kinetic models. At present, the direct empirical approaches are more likely to provide useful estimates of environmentally available uranium at reasonable costs. For the long term, we recommend development of a flow-cell procedure that incorporates dissolution-rate information into the assessment of environmental availability. In the interim, we recommend development of a batch wet-chemical procedure based on a combination of standard and non-standard methods. Both procedures require laboratory testing and correlation with field data before being used for regulatory purposes. Lastly, we have tabulated literature data on the aqueous complexes of uranium and major uranium minerals, shown that the environmental availability of uranium cannot be predicted from thermodynamic solubility data alone, and compiled a list of laboratories capable of performing environmental availability determinations.

NUREG/CR-6233 V01: STABILITY OF CRACKED PIPE UNDER INERTIAL STRESSES. Subtask 1.1 Final Report. SCOTT, P.M.; WILSON, M.; OLSON, R.J.; et al. Battelle Memorial Institute, Columbus Laboratories. August 1994. 195pp. 9409260057. BML-2177. 81043:001.

This report presents the results of the pipe fracture experiments, analyses, and material characterization efforts performed within Subtask 1.1 of the IPIRG Program. The objective of Subtask 1.1 was to experimentally verify the analysis methodologies for circumferentially cracked pipe subjected primarily to inertial stresses. Eight cracked-pipe experiments were conducted on 6-inch nominal diameter TP304 and A106B pipe. The experimental procedure was developed using nonlinear time-history finite element analyses which included the nonlinear behavior due to the crack. The model did an excellent job of predicting the displacements, forces, and times to maximum moment. The comparison of the experimental loads to the predicted loads by the Net-Section-Collapse (NSC), Dimensionless Plastic-Zone Parameter, J-estimation schemes, R6, and ASME Section XI in-service flaw assessment criteria tended to underpredict the measured bending moments except for the NSC analysis of the A106B pipe. The effects of flaw geometry and loading history on toughness were evaluated by calculating the toughness from the pipe tests and comparing these results to C(T) values. These effects were found to be variable. The surface-crack geometry tended to increase the toughness (relative to C(T) results), whereas a negative load-ratio significantly decreased the TP304 stainless steel surface-cracked pipe apparent toughness. The inertial experiments tended to achieve complete failure within a few cycles after reaching maximum load in these relatively small diameter pipe experiments. Hence, a load-controlled

fracture mechanics analysis may be more appropriate than a displacement-controlled analysis for these tests.

NUREG/CR-6234: VALIDATION OF ANALYSIS METHODS FOR ASSESSING FLAWED PIPING SUBJECTED TO DYNAMIC LOADING. OLSON, R.J.; WOLTERMAN, R.L.; WILKOWSKI, G.M.; et al. Battelle Memorial Institute, Columbus Laboratories. August 1994. 68pp. 9409060227. ANL-94/22. 80785:028.

Argonne National Laboratory and Battelle have jointly conducted a research program for the USNRC to evaluate the ability of current engineering analysis methods and one state-of-the-art analysis method to predict the behavior of circumferentially surface-cracked pipe system water-hammer experiments. The experimental data used in the evaluation were from the HDR Test Group E31 series conducted by the Kernforschungszentrum Karlsruhe (KfK) in Germany. The incentive for this evaluation was that simplified engineering methods, as well as newer "state-of-the-art" fracture analysis methods, have been typically validated only with static experimental data. Hence these dynamic experiments were of high interest. High-rate dynamic loading can be classified as either repeating, e.g., seismic, or nonrepeating, e.g., water hammer. Development of experimental data and validation of cracked pipe analyses under seismic loading (repeating dynamic loads) are being pursued separately within the NRC's International Piping Integrity Research Group (IPIRG) program. This report described developmental and validation efforts to predict crack stability under water-hammer loading, as well as comparisons using currently used analysis procedures. Current fracture analysis methods use the elastic stress analysis loads decoupled from the fracture mechanics analysis, while state-of-the-art methods employ nonlinear cracked-pipe time-history finite element analyses. The results showed that the current decoupled methods were conservative in their predictions, whereas the cracked pipe finite element analyses were more accurate, yet slightly conservative. The nonlinear time-history cracked-pipe finite element analyses conducted in this program were also attractive in that they were done on a small Apollo DN5500 workstation, whereas other cracked-pipe dynamic analyses conducted in Europe on the same experiments required the use of a CRAY2 supercomputer, and were less accurate.

NUREG/CR-6236: SEISMIC INVESTIGATIONS OF THE HDR SAFETY PROGRAM. Summary Report. MALCHER, L.; SCHRAMMEL, D.; STEINHILBER, H.; et al. Germany, Federal Republic of. August 1994. 95pp. 9409060242. ANL-94/20. 80785:249.

The primary objective of the seismic investigations, performed at the HDR facility in Kahl/Main, FRG was to validate calculational methods for the seismic evaluation of nuclear-reactor systems, using experimental data from an actual nuclear plant. Using eccentric mass shaker excitation the HDR soil/structure system was tested to incipient failure, exhibiting highly nonlinear response and demonstrating that structures not seismically designed can sustain loads equivalent to a design basis earthquake (DBE). Load transmission from the structure to piping/equipment indicated significant response amplifications and shifts to higher frequencies, while the response of tanks/vessels depended mainly on their support conditions. The evaluation of various piping support configurations demonstrated that proper system design (for a given spectrum) rather than number of supports or system stiffness is important to limiting pipe stresses. Piping at loads exceeding the DBE eightfold still had significant margins and failure is improbable in spite of multiple support failures. The mean value for pipe damping, even under extreme loads, was found to be about 4%. Comparison of linear and nonlinear computational results with piping response measurements showed that predictions have a wide scatter and do not necessarily yield conservative responses underpredicting, in particular, peak support forces. For the soil/structure system the quality of the predictions did not depend so much on the

complexity of the modeling, but rather on whether the model captured the salient features and nonlinearities of the system.

NUREG/CR-6237: STATISTICAL ANALYSIS OF FATIGUE STRAIN-LIFE DATA FOR CARBON AND LOW-ALLOY STEELS. KEISLER, J.; CHOPRA, O.K.; SHACK, W.J. Argonne National Laboratory. August 1994. 52pp. 9409260059. ANL-94/21. 81043:196.

The existing fatigue strain vs. life (S-N) data, foreign and domestic, for carbon and low-alloy steels used in the construction of nuclear power plant components have been compiled and categorized according to material, loading, and environmental conditions. A statistical model has been developed for estimating the effects of the various test conditions on fatigue life. The results of a rigorous statistical analysis have been used to estimate the probability of initiating a fatigue crack. Data in the literature were reviewed to evaluate the effects of size, geometry, and surface finish of a component on its fatigue life. The fatigue S-N curves for components have been determined by applying design margins for size, geometry, and surface finish to crack initiation curves estimated from the model. The significance of the effect of environment on the current Code design curve and on the proposed interim design curves for carbon and low-alloy steels presented in NUREG/CR-5999 is discussed.

NUREG/CR-6241: TECHNICAL GUIDELINES FOR ASEISMIC DESIGN OF NUCLEAR POWER PLANTS. Translation Of JEAG 4601-1987. PARK, Y.J.; HOFMAYER, C.H. Brookhaven National Laboratory. June 1994. 962pp. 9407080256. BNL-NUREG-52422. 80129:001.

This document is a translation, in its entirety, of the Japan Electric Association (JEA) publication entitled "Technical Guidelines for Aseismic Design of Nuclear Power Plants - JEAG 4601-1987." This guideline describes in detail the aseismic design techniques used in Japan for nuclear power plants. It contains chapters dealing with: (a) the selection of earthquake ground motions for a site, (b) the investigation of foundation and bedrock conditions, (c) the evaluation of ground stability and the effects of ground movement on buried piping and structures, (d) the analysis and design of structures, and (e) the analysis and design of equipment and distribution systems (piping, electrical raceways, instrumentation, tubing and HVAC duct). The guideline also includes appendices which summarize data, information and references related to aseismic design technology.

NUREG/CR-6245: ASSESSMENT OF PRESSURIZED WATER REACTOR CONTROL ROD DRIVE MECHANISM NOZZLE CRACKING. SHAH, V.N.; WARE, A.G.; PORTER, A.M. EG&G Idaho, Inc. October 1994. 67pp. 9412300191. EGG-2715. 82159:255.

This report surveys the field experience related to cracking of pressurized water reactor (PWR) control rod drive mechanism nozzles (Alloy 600 material); evaluates design, fabrication, and operating conditions for the nozzles in U.S. PWRs; and evaluates the safety significance of nozzle cracking. Inspection at 78 overseas and one U.S. PWR has revealed mainly axial cracks in 101 nozzles. The cracking is caused by primary water stress corrosion cracking, which requires the simultaneous presence of high tensile stresses, high operating temperatures, and susceptible microstructure. CRDM nozzle cracking is not a short-term safety issue. An axial crack is not likely to grow above the vessel head to a critical length because the stresses are not high enough to support the growth away from the attachment weld. Primary coolant leaking through an axial crack could cause a short circumferential crack on the outside surface. However, this crack is not likely to propagate through the nozzle wall to cause rupture. Leakage of the primary coolant from a through-wall crack could cause boric acid corrosion of the vessel head and challenge the structural integrity of the head, but it is very unlikely that the accumulated deposits of boric acid crystals resulting from such leakage could remain undetected.

NUREG/CR-6249: UNIRRADIATED MATERIAL PROPERTIES OF MIDLAND WELD WF-70. MCCABE, D.E.; NANSTAD, R.K.; ISKANDER, S.K.; et al. Oak Ridge National Laboratory. October 1994. 93pp. 9411160151. ORNL/TM-12777. 81756:303.

Weld metal, designated WF-70, taken from the nozzle course and beltline welds of the Midland Reactor, Unit 1, has been given a preliminary evaluation using the conventional Charpy V-notch (CVN), drop-weight (DWT), and chemical analyses. There was a significant difference in copper content, nominally 0.25% versus 0.40%. Because the objective of this study was to evaluate the before-and-after irradiation properties, these are regarded as different materials. This report summarizes material characterization results and presents the results of fracture mechanics tests on the unirradiated material to establish baseline transition temperature and J-R curves. Tensile properties were also determined. Reference nil-ductility temperatures, RD(NDT), determined from CVN transition curves (RT(NDT)) method specific to low upper-shelf energy materials) varied from -20 to +37 degrees C (-4 to 99 degrees F) at various locations in the beltline weld. Reference temperatures using a fracture mechanics-based transition temperature model were -60 degrees C (-76 degrees F) for the beltline weld and -33 degrees C (-27 degrees F) for the nozzle weld. Tensile tests indicated the nozzle weld had higher strength than the beltline weld. J-R curves were developed at 21, 50, and 288 degrees C (70, 32, and 550 degrees F). The predicted J-R curves matched the experimental J-R curves reasonably well. Crack-arrest tests were conducted, but the specimens failed to develop American Society for Testing and Materials valid data in all but one test. More experimentation and test method development are recommended. Five experimental objectives to be accomplished from the testing of irradiated material were identified.

NUREG/CR-6250: SUMMARY OF COMMENTS RECEIVED ON STAFF DRAFT PROPOSED RULE ON RADIOLOGICAL CRITERIA FOR DECOMMISSIONING. CAPLIN, J.; PAGE, G.; SMITH, D.; et al. Advanced Systems Technology, Inc. August 1994. 123pp. 9409060247. 80786:001.

The Nuclear Regulatory Commission (NRC) is conducting an enhanced participatory rulemaking to establish radiological criteria for the decommissioning of NRC-licensed facilities. The NRC obtained comments on the scope, issues, and approaches through a series of workshops (57 FR 58727), Generic Environmental Impact Statement (GEIS) scoping meetings (58 FR 33570), a dedicated electronic bulletin board system (58 FR 37760), and written submissions. A summary of workshop and scope-meeting comments was published as NUREG/CR-6156. On February 2, 1994, the Commission published in the Federal Register (59 FR 4868) a notice that the NRC staff had prepared a "staff draft" proposed rule on radiological criteria for decommissioning. Copies of the staff draft were distributed to the Agreement States, participants in the earlier meetings, and other interested parties for comment. This report summarizes the comments identified from the 96 docketed letters received on the staff draft. No analysis or response is included in this report. The comments reflect a broad spectrum of viewpoints. Two subjects on which the commenters were in general agreement were (1) that the enhanced participatory rulemaking should proceed, and (2) that the forthcoming GEIS and guidance documents are needed for better understanding of the draft rule.

NUREG/CR-6252: LESSONS LEARNED FROM THE THREE MILE ISLAND-UNIT 2 ADVISORY PANEL. LACH, D.; BOLTON, P.; DURBIN, N.; et al. Battelle Seattle Research Center. August 1994. 53pp. 9409060253. PNL-9871. 80786:124.

In response to public concern about the cleanup of the Three Mile Island, Unit 2 (TMI-2) facility after an accident on March 28, 1979 involving a loss of reactor coolant and subsequent damage to the reactor fuel, twelve citizens were asked to serve on an independent Advisory Panel to consult with the Nuclear Regulatory Commission (NRC) on the decontamination and

cleanup of the facility. The panel met 78 times over a period of thirteen years, holding public meetings in the vicinity of TMI-2 and meeting regularly with NRC Commissioners in Washington, D.C. This report describes the results of a project designed to identify and describe the lessons learned from the Advisory Panel and place those lessons in the context of what we generally know about citizen advisory groups. A summary of the empirical literature on citizen advisory panels is followed by a brief history of the TMI-2 Advisory Panel. The body of the report contains the analysis of the lessons learned, preliminary conclusions about the effectiveness of the Panel, and implications for the NRC in the use of advisory panels. Data for the report include meeting transcripts and interviews with past and present Panel participants.

NUREG/CR-6254: SOUTHERN APPALACHIAN REGIONAL SEISMIC NETWORK. CHIU,S.C.C.; JOHNSTON,A.C.; CHIU,J.-M. Memphis State Univ., Memphis, TN. August 1994. 81pp. 9409230288. 81010:035.

Memphis State University has monitored the seismicity of the southern Appalachian area since late 1979 by means of the Southern Appalachian Regional Seismic Network (SARSN), which has provided good spatial coverage for earthquake location. Activity is more heavily concentrated in the Valley and Ridge province (VR) of eastern Tennessee than in the Blue Ridge (BR) or Piedmont (P). The majority of these events lie between the New York-Alabama and the Clinch/Ocoee lineaments, magnetic anomalies of deep-seated basement structures. Thus, SARSN has been able to define the first order characteristics of the Southern Appalachian seismic zone. The focal depths of the earthquakes are concentrated between 8 and 16 km, principally beneath the Appalachian overthrust. In cross-section the average seismicity is shallower beneath the BR and P provinces than the VR and North American craton. Results of focal mechanism studies for events that occurred between October 1986 and December 1991, indicate that the basement of the VR province is under horizontal, NE-SW compressive stress. Right-lateral strike-slip faulting on nearly N-S faults is preferred as it agrees with the trend of the regional magnetic anomaly pattern.

NUREG/CR-6255: DESIGN OF AN OPEN ARCHITECTURE SEISMIC MONITORING SYSTEM. GHALIB,H.A.; LEONARD,S.K.; KRAFT,G.D. ENSCO, Inc. September 1994. 125pp. 9410060315. DCS-94-024. 81223:235.

This document presents a top level design of an Open Architecture Seismic Monitoring (OASM) system intended to automatically monitor local and regional seismic activities. The system is designed to process single and three component data similar to those recorded by the U.S. National Seismic Network and local stations operated by academic and research institutions. The main components of the system are time series processing, event formation, event classification, and hazard assessment. These fully independent modules are complemented by operational and research databases. Signal detection, onset time estimation and signal characterization are functions of the time series module. Signal association and event location are performed in the event formation module. The seismic source is characterized in the event classification module. The seismic hazard assessment module estimates basic parameters for use in seismic risk analysis. The operational database acts as the central facility through which access to the seismic data and communication between the system's modules are accomplished. The function of the research database is to provide a seismic bulletin and segmented waveforms for events.

NUREG/CR-6258: THE LIQUEFACTION METHOD FOR ASSESSING PALEOSEISMICITY. TUTTLE,M.P. Lamont-Doherty Earth Observatory. December 1994. 47pp. 9501180164. 82348:262.

Paleoseismicity studies expand our knowledge of seismic activity into the prehistoric period and thereby can improve our understanding of the earthquake potential of various regions. Paleoseismology is proving especially useful in eastern North

American, where the recurrence interval of large earthquakes is longer than the historic record of earthquakes. Because surface traces of seismogenic faults have been difficult to identify in eastern North America, most paleoseismicity studies have employed features resulting from liquefaction. The goals of paleoliquefaction studies are to determine the recurrence intervals, magnitudes, and source areas of prehistoric earthquakes. To accomplish these goals, one must be able to identify earthquake-induced liquefaction features, determine their ages, and map their distribution. This report reviews (1) sediment deformation structures, (2) methods for dating liquefaction features, and (3) relationships between liquefaction and the magnitude and distance of causative earthquakes. Recent studies by the author in Quebec Province, Canada and in the New Madrid seismic zone of the central US provide the basis for this report.

NUREG/CR-6262: CLEAVAGE BEHAVIORS IN NUCLEAR VESSEL STEELS. IRWIN,G.R.; ZHANG,X.J. Maryland, Univ. of, College Park, MD. * Oak Ridge National Laboratory. November 1994. 27pp. 9412300223. ORNL SUB79777811. 82163:001.

Cleavage behaviors of nuclear vessel steels in the transition temperature range are reviewed. Viewpoints are presented to assist understanding of cleavage crack speed, cleavage initiation, cleavage arrest, and the sensitivity of fracture toughness to constraint and temperature. The importance of high local stress elevations by high strain rate is emphasized. This report is designated as HSST Report No. 149.

NUREG/CR-6267: AIR-WATER SIMULATION OF PHENOMENA OF CORIUM DISPERSION IN DIRECT CONTAINMENT HEATING. ISHII,M.; REVANKAR,S.T.; ZHANG,G.; et al. Purdue Univ., West Lafayette, IN. October 1994. 126pp. 9411160157. PU NE-93/1. 81757:031.

The degree of corium dispersion has not only the strongest parametric effect on the containment pressurization but also has the highest uncertainty in predicting it. In view of this, separate effect tests on the corium dispersion mechanisms in the reactor cavity and the subcompartment trapping mechanisms were carried out. Four major objectives of this corium dispersion study are: (1) to perform a detailed scaling study using the newly proposed step-by-step integral scaling method, then to evaluate existing models for entrainment, particle size and trapping, (2) to perform carefully designed simulation experiments using water-air and woods metal-air in a 1/10 linear scale model, (3) to develop reliable mechanistic models and correlations for corium dispersions, which can be used to predict corium jet disintegration, entrainment, drop size, liquid film carry over, and subcompartment trapping, and (4) to use the models to perform stand alone calculations for typical prototypic conditions. The results of the experiments that were conducted using air-water are presented.

NUREG/CR-6269: A PLAN FOR THE MODIFICATION AND ASSESSMENT OF TRAC-PF1/MOD2 FOR USE IN ANALYZING CANDU 3 TRANSIENT THERMAL-HYDRAULIC PHENOMENA. SIEBE,D.A.; BOYACK,B.E.; GIGUERE,P.T. Los Alamos National Laboratory. November 1994. 73pp. 9412130154. LA-12853-MS. 82008:048.

This report presents the results of the review and planning done for the United States Nuclear Regulatory Commission to identify the thermal-hydraulic phenomena that could occur in the CANDU 3 reactor design during transient conditions, plan modifications to the TRAC-PF1/MOD2 (TRAC) computer code needed to adequately predict CANDU 3 transient thermal-hydraulic phenomena, and identify an assessment program to verify the ability of TRAC, when modified, to predict these phenomena. This work builds on analyses and recommendations produced by the Idaho National Engineering Laboratory (INEL). To identify the thermal-hydraulic phenomena, a large-break loss-of-coolant accident simulation, performed as part of earlier work by INEL with an Atomic Energy of Canada, Limited (AECL) thermal-hydraulic computer code (CATHENA), was analyzed in

detail. Other accident scenarios were examined for additional phenomena. A group of Los Alamos National Laboratory reactor thermal-hydraulics experts ranked the phenomena to produce a preliminary phenomena identification and ranking table (PIRT). This PIRT provided an adequate foundation for planning a program of code modifications. A plan for code modifications was developed based on this PIRT and on information about the modeling methodologies for CANDU-specific phenomena used in AECL codes. AECL thermal-hydraulic test facilities and programs were reviewed and the information used in developing an assessment plan to ensure that TRAC-PF1/MOD2, when modified, will adequately predict CANDU 3 phenomena.

NUREG/CR-6270 DRF FC: ESTIMATING BOILING WATER REACTOR DECOMMISSIONING COSTS. A User's Manual For The BWR Cost Estimating Computer Program (CECP) Software. Draft Report For Comment. BIRSCHBACH, M. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1994. 188pp. 9412300195. PNL-10086. 82160:278.

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

NUREG/CR-6276: QUALITY MANAGEMENT IN REMOTE AFTERLOADING BRACHYTHERAPY. TORTORELLI, J.P.; SIMION, G.P.; KOZLOWSKI, S.D. EG&G Idaho, Inc. October 1994. 93pp. 9412130077. EGG-2746. 82006:211.

Over a dozen government and professional organizations in the United States and Europe have issued regulations and guidance concerning quality management in the practice of remote afterloading brachytherapy. Information from the publications of these organizations was collected and collated for this report. This report provides the brachytherapy licensee access to a broad field of quality management information in a single, topically organized document.

NUREG/CR-6278: SURVEY OF INDUSTRY METHODS FOR PRODUCING HIGHLY RELIABLE SOFTWARE. LAWRENCE, J.D.; PERSONS, W.L. Lawrence Livermore National Laboratory. November 1994. 47pp. 9412130117. UCRL-ID-117524. 82008:001.

The Nuclear Reactor Regulation Office of the U.S. Nuclear Regulatory Commission is charged with assessing the safety of new instrument and control designs for nuclear power plants which may use computer-based reactor protection systems. Lawrence Livermore National Laboratory has evaluated the latest techniques in software reliability for measurement, estimation, error detection, and prediction that can be used during the software life cycle as a means of risk assessment for reactor protection systems. One aspect of this task has been a survey of the software industry to collect information to help identify the design factors used to improve the reliability and safety of software. The intent was to discover what practices really work in industry and what design factors are used by industry to achieve highly reliable software. The results of the survey are documented in this report. Three companies participated in the survey: Computer Sciences Corporation, International Business Machines (Federal Systems Company), and TRW. Discussions were also held with NASA Software Engineering Lab, University of Maryland, CSC, and the AIAA Software Reliability Project.

NUREG/CR-6281: A SIMPLIFIED LEAK-BEFORE-BREAK EVALUATION PROCEDURES FOR AUSTENITIC AND FERRITIC STEEL PIPING. GAMBLE, R.M.; ZAHOR, A.; GHASSEMI, B. Novetech Corp. October 1994. 48pp. 9411280283. 81851:226.

A simplified procedure has been defined for computing the allowable circumferential throughwall crack length as a function of applied loads in piping. This procedure has been defined to enable leak-before-break (LBB) evaluations to be performed without complex and time consuming analyses. The development of the LBB evaluation procedure is similar to that now used in Section XI of the ASME Code for evaluation of part-throughwall flaws found in piping. The LBB evaluation procedure was bench marked using experimental data obtained from pipes having circumferential throughwall flaws. Comparisons of the experimental and predicted load carrying capacities indicate that the method has a conservative bias, such that for at least 97% of the experiments the experimental load is equal to or greater than 90% of the predicted load.

NUREG/CR-6288: GEOCHEMICAL INVESTIGATIONS RELATED TO THE YUCCA MOUNTAIN ENVIRONMENT AND POTENTIAL NUCLEAR WASTE REPOSITORY. MURPHY, W.M.; PABALAN, R.T. Center for Nuclear Waste Regulatory Analyses. November 1994. 200pp. 9412300218. 82161:098.

Research is reported which focused on experimental determinations and theoretical interpretations of fundamental thermodynamic and kinetic properties of minerals and reactions that characterize geochemical processes at the proposed nuclear waste repository site at Yucca Mountain and that could affect the capacity of the site to isolate nuclear waste. Cation exchange equilibria were determined for clinoptilolite and binary solutions of Na(+) with K(+), Ca(2+), and Sr(2+). Results were interpreted using a Margules solid solution model. Dissolution rate data for analcime can be rationalized by mechanisms which invoke either ultrareactive material or rate dependence on aqueous aluminum. Solubility determinations were interpreted to obtain equilibrium constants for dissolution reactions of analcime and of Na-clinoptilolite. Chemical analytical data indicate that natural groundwaters at Yucca Mountain appear to be at equilibrium with analcime. Employing principles of thermodynamics, reaction kinetics, mass transfer, an mass transport, computational models were developed using EQ3/6 to explore the evolution of the natural geochemical system, and the geochemical processes that are likely to occur under repository conditions.

NUREG/CR-6289: RECONCENTRATION OF RADIOACTIVE MATERIAL RELEASED TO SANITARY SEWERS IN ACCORDANCE WITH 10 CFR PART 20. AINSWORTH, C.C.; HILL, R.L.; CANTRELL, K.J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1994. 200pp. 9412300213. PNL-10193. 82161:294.

The U.S. Nuclear Regulatory Commission (NRC), in accordance with 10 CFR 20, and agreement states, in accordance with state regulations, regulates the discharge of radioactive materials into sanitary sewer systems. A one-year study was conducted by Pacific Northwest Laboratory (PNL) for the NRC to assess whether radioactive materials that are discharged to sanitary sewer systems undergo significant reconcentration within the wastewater treatment plants (WWTP) and to determine the physical and/or chemical process that may result in radionuclide reconcentration within the WWTPs. The study objectives were addressed by collecting information and data on wastewater treatment, relevant geochemical processes, and individual radionuclide behavior in WWTPs from the open literature, NRC reports, EPA surveys, and interviews with NRC licensees and staff of WWTPs that may be impacted by these discharges. Radionuclide mass balance and removal efficiencies were calculated for WWTPs at Oak Ridge, TN; and Erwin, TN, but were not shown to be reliable since the licensee release data generally underestimated the mass of radionuclide that was ultimately found in the sludge. This disparity may be

due, in part, to the fact that data available for use in this study were collected to address regulatory concerns and not to perform mass balance calculations. A limited modeling study showed some promise for predicting radionuclide behavior in WWTPs, however, the general applicability of using these empirical models remains uncertain. With the data and models currently available, it is not possible to quantitatively determine the physical and chemical processes that cause reconcentration or to calculate, a priori, reconcentration factors for specific WWTP unit processes or WWTPs in general.

NUREG/CR-6290: KEY ANALYSIS SYSTEM USER'S GUIDE. Version 2.0. MASSE, R.P. Interior, Dept. of, Geological Survey. November 1994. 146pp. 9412070007. 81956:001.

The KEY analysis system is a software program designed to process digital waveform data from the United States National Seismograph Network. The KEY system performs many data processing and scientific analysis functions. Detailed operating procedures for the KEY analysis system are provided in this User's guide.

NUREG/CR-6294: DESIGN FACTORS FOR SAFETY-CRITICAL SOFTWARE. LAWRENCE, J.D.; PRECKSHOT, G.G. Lawrence Livermore National Laboratory. December 1994. 22pp. 9412300220. 82162:323.

This report is the fourth of a series of reports prepared for the Nuclear Regulatory Commission Office of Nuclear Reactor Regulation, and provides the summary and conclusion for this task. It is widely believed in the software engineering community that almost anything can affect the ability of software to reliably perform its tasks, particularly when safety is at issue. While this statement is true, both in the abstract and in specific instances, it is not particularly helpful. It remains necessary for auditors and other reviewers to assure themselves and the public that safety-critical software has sufficiently low probability of failing in such a way as to cause death or injury to permit it to be used in safety-critical applications. Achieving this assurance is best done by using a well-planned, methodical approach. A possible approach is to concentrate on those attributes of the software and the development process (design factors) that are most influential in achieving dependable software. Seventy-four design factors are identified in this report, divided into nine categories. Seven categories relate to the development process, and one category relates to the products of that process. The remaining category contains negative factors whose presence should be regarded as cause for intense scrutiny of the development process. Seven of the design factors should be considered mandatory for any organization responsible for developing safety-critical software. An additional nine factors are considered essential to safety, but not as important as the first seven. The remaining design factors can provide additional important indications of the quality of the development effort and the software resulting from that effort.

NUREG/CR-6303: METHOD FOR PERFORMING DIVERSITY AND DEFENSE-IN-DEPTH ANALYSES OF REACTOR PROTECTION SYSTEMS. PRECKSHOT, G.G. Lawrence Livermore National Laboratory. December 1994. 45pp. 9501180332. UCRL-ID-119239. 82347:226.

The purpose of this NUREG is to describe a method for analyzing computer-based nuclear reactor protection systems that discovers design vulnerabilities to common-mode failure. The potential for common-mode failure has become an important issue as the software content of protection systems has increased. This potential was not present in earlier analog protection systems because it could usually be assumed that common-mode failure, if it did occur, was due to slow processes such as corrosion or premature wear-out. This assumption is no longer true for systems containing software. It is the purpose of the analysis method described here to determine points of a design for which credible common-mode failures are uncompensated either by diversity or defense-in-depth.

NUREG/GR-0008: VALIDATION OF SEISMIC PROBABILISTIC RISK ASSESSMENTS OF NUCLEAR POWER PLANTS. ELLINGWOOD, B. Johns Hopkins Univ., Baltimore, MD. January 1994. 134pp. 9403150302. 78501:001.

A seismic probabilistic risk assessment (PRA) of a nuclear plant requires identification and information regarding the seismic hazard at the plant site, dominant accident sequences leading to core damage, and structure and equipment fragilities. Uncertainties are associated with each of these ingredients of a PRA. The sensitivity of accident sequence probabilities and high-confidence, low probability of failure (HCLPF) plant fragilities to seismic hazard and fragility modeling assumptions was examined for three nuclear power plants. Mean accident sequence probabilities were found to be relatively insensitive (by a factor of two or less) to: uncertainty in the coefficient of variation (logarithmic standard deviation) describing inherent randomness in component fragility; truncation of lower tail of fragility; uncertainty in random (non-seismic) equipment failures (e.g., diesel generators); correlation between component capacities; and functional form of fragility family. Accident sequence probabilities, expressed in the form of a frequency distribution, are affected significantly by the seismic hazard modeling. When the fragility modeling and plant logic are effectively uncoupled from the seismic hazard analysis in a seismic margin study, the influence of the large uncertainty in the seismic hazard is eliminated. In seismic margin studies, uncertainties in fragility modeling assume a different significance than they do in seismic PRA studies. Gross design and construction errors and their impact on estimates of seismic risk are dealt with in approximation by postulating various error scenarios and their effect on component fragilities and recalculating the core damage probabilities. Plausible design/construction errors appear to have little effect on mean core damage probabilities.

NUREG/GR-0011: INFORMATION BIAS AND LIFETIME MORTALITY RISKS OF RADIATION-INDUCED CANCER. Low LET Radiation. PETERSON, L.E.; SCHULL, W.J.; DAVIS, B.R.; et al. Texas, Univ. of, Houston, TX. April 1994. 182pp. 9405310162. 79591:163.

Additive and multiplicative models of relative risk were used to measure the effect of cancer misclassification and DS86 random errors on lifetime risk projections in the Life Span Study (LSS) of Hiroshima and Nagasaki atomic bomb survivors. The true number of cancer deaths in each stratum of the cancer mortality cross-classification was estimated using sufficient statistics from the EM algorithm. Average survivor doses in the strata were corrected for DS86 random error ($\sigma=0.45$) by use of reduction factors. Poisson regression was used to model the corrected and uncorrected mortality rates with covariates for age at-time-of-bombing, age at-time-of-death and gender. Excess risks were in good agreement with risks in RERF Report 11 (Part 2) and the BEIR-V Report. Bias due to DS86 random error typically ranged from -15% to -30% for both sexes, and all sites and models. The total bias, including diagnostic misclassification, of excess risk of nonleukemia for exposure to 1 Sv from age 18 to 65 under the non-constant relative projection model was -37.1% for males and -23.3% for females. Total excess risks of leukemia under the relative projection model were biased -27.1% for males and -43.4% for females. Thus, nonleukemia risks for 1 Sv from ages 18 to 65 (DRREF=2) increased from 1.91%/Sv to 2.68%/Sv among males and from 3.23%/Sv to 4.02%/Sv among females. Leukemia excess risks increased from 0.87%/Sv to 1.10%/Sv among males and from 0.73%/Sv to 1.04%/Sv among females. Bias was dependent on the gender, site, correction method, exposure profile and projection model considered. Future studies that use LSS data for U.S. nuclear workers may be downwardly biased if lifetime risk projections are not adjusted for random and systematic errors.

NUREG/GR-0013: APPLICATIONS OF A NEW MAGNETIC MONITORING TECHNIQUE TO IN SITU EVALUATION OF FATIGUE DAMAGE IN FERROUS COMPONENTS. JILES,D.C.; BINDER,S.B.; GOVINDARAJU,M.; et al. Iowa State Univ., Ames, IA. June 1994. 41pp. 9407250286. 80328:195.

The work undertaken in this project consisted of research into the use of magnetic inspection methods for the estimation of fatigue life of nuclear pressure vessel steel. The rationale for this work was that the mechanical and magnetic properties of ferromagnetic materials are closely interrelated, and therefore the measurements of the magnetic properties could be used to monitor the evolution of fatigue damage in these specimens as they were subjected to cyclic loading. The results of the work have shown that it is possible to monitor the fatigue damage nondestructively by magnetic techniques. For example, in load-controlled high-cycle fatigue tests, it has been found that the plastic strain and coercivity accumulate logarithmically during the fatigue process. Thus a quantitative relationship between coercivity and the number of fatigue cycles could be established based on two empirical coefficients, which can be determined from the test conditions and material properties. Also it was found that prediction of the onset of fatigue failure in steels was possible under certain conditions. In strain-controlled low cycle fatigue, critical changes in Barkhausen emissions, coercivity and hysteresis loss occurred in the last ten to twenty percent of fatigue life.

NUREG/IA-0093: RELAP5/MOD3 ASSESSMENT FOR CALCULATION OF SAFETY AND RELIEF VALVE DISCHARGE PIPING HYDRODYNAMIC LOADS. STUBBE,E.J.; VANHOENACKER,L.; OTERO,R. TRACTEBEL. February 1994. 300pp. 9405040180. ICAP00159. 79105:001.

This report presents an assessment study for the use of the code RELAP5/MOD3/5M5 in the calculation of transient hydrodynamic loads on safety and relief discharge pipes. Its predecessor, RELAP5/MOD1, was found adequate for this kind of calculation by EPRI. The hydrodynamic loads are very important for the discharge piping design because of the fast opening of the valves and the presence of liquid in the upstream loop

seals. The code results are compared to experimental load measurements performed at the Combustion Engineering Laboratory in Windsor (U.S.A.). Those measurements were part of the PWR Valve Test Program undertaken by EPRI after the TMI-2 accident. This particular kind of transients challenges the applicability of the following code models: two-phase choked discharge; interphase drag in conditions with large density gradients; heat transfer to metallic structures in fast changing conditions; and two-phase flow at abrupt expansions.

NUREG/IA-0114: ASSESSMENT OF RELAP5/MOD3 WITH THE LOFT L9-1/L3-3 EXPERIMENT SIMULATING AN ANTICIPATED TRANSIENT WITH MULTIPLE FAILURES. BANG,Y.S.; SEUL,K.W.; KIM,H.J. Korea Institute of Nuclear Safety. February 1994. 112pp. 9404010254. ICAP00196. 78728:155.

The RELAP5/MOD3 5m5 code was assessed using the L9-1/L3-3 test carried out in the LOFT facility, a 1/60-scaled experimental reactor, simulating a loss of feedwater accident with multiple failures and the sequentially-induced small break loss-of-coolant accident. The code predictability was evaluated for the four separated sub-periods with respect to the system response; initial heatup phase, spray and PORV cycling phase, blowdown phase and recovery phase. Based on the comparisons of the results from the calculation with the experiment data, it is shown that the overall thermal-hydraulic behavior important to the scenario such as a heat removal between the primary side and the secondary side and a system depressurization was well-predicted and that the code could be applied to the full-scale nuclear power plant for an anticipated transient with multiple failures within a reasonable accuracy. The minor discrepancies between the prediction and the experiment were identified in reactor scram time, post-scram behavior in the initial heatup phase, excessive heatup rate in the cycling phase, insufficient energy convected out the PORV under the hot leg stratified condition in the saturated blowdown phase and void distribution in secondary side in the recovery phase. This may come from the code uncertainties in predicting the spray mass flow rate, the associated condensation in pressurizer and junction fluid density under stratified condition.

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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 sub-sections of these (e.g., divisions, branches) where appropriate. Each entry is followed by a NUREG number and title of the report(s). If further information is needed, refer to the main citation by NUREG number.

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