Regulatory and Technical Reports (Abstract Index Journal)

Annual Compilation for 1995

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Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

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Regulatory and Technical Reports (Abstract Index Journal)

Annual Compilation for 1995

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M. Sheehan, Project Manager

**Regulatory Publications Branch
Division of Freedom of Information and Publications Services
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001**

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PREFACE

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

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

Secondary Report Number Index
Personal Author Index
Subject Index
NRC Originating Organization Index (Staff Reports)
NRC Originating Organization Index (International Agreements)
NRC Contract Sponsor Index (Contractor 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. Kraftweek Union. August 1986. 223 pp. 8608270424. 37659:138.

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

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

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

Availability of NRC Publications

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

Superintendent of Documents
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NRC Report Codes

The NUREG designation, NUREG-XXXX, indicates that the document is a formal NRC staff-generated report. Contractor-prepared formal NRC reports carry the report code NUREG/CR-XXXX. This type of identification replaces contractor-established codes such as ORNL/NUREG/TM-XXX and TREE-NUREG-XXXX, as well as various other numbers that could not be correlated with NRC sponsorship 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 V19: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of December 31, 1994. (Gray Book I) HARTFIELD, R.A. Office of Information Resources Management (Post 890205). April 1995. 350pp. 9504260255. 83635:001.

The Nuclear Regulatory Commission's annual summary of licensed nuclear power reactor data is based primarily on the report of operating data submitted by licensees for each unit for the month of December because that report contains data for the month of December, the year to date (in this case calendar year 1994) 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 V18 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, October-December 1994. (White Book) * Office of Nuclear Reactor Regulation (Post 941001). February 1995. 200pp. 9503140385. 83069:031.

This periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations during the period from October through December 1994.

NUREG-0040 V19 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, January-March 1995. (White Book) * Office of Nuclear Reactor Regulation (Post 941001). May 1995. 119pp. 9506020502. 84149:041.

This periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations during the period from January through March 1995.

NUREG-0040 V19 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, April-June 1995. (White Book) * Office of Nuclear Reactor Regulation (Post 941001). August 1995. 181pp. 9509070144. 85398:116.

This periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations during the period from April through June 1995.

NUREG-0040 V19 N03: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, July-September 1995. (White Book) * Office of Nuclear Reactor Regulation (Post 941001). December 1995. 224pp. 9601290256. 86882:001.

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 1995.

NUREG-0090 V17 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. July-September 1994. * Office for Analysis & Evaluation of Operational Data, Director. January 1995. 35pp. 9504190361. 83575:183.

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 July 1 through September 30, 1994. This report addresses five abnormal occurrences (AOs) at NRC-licensed facilities. One involved a medical brachytherapy misadministration, two involved medical teletherapy misadministrations, one involved a medical sodium iodide misadministration, and one involved a medical sodium iodide event. One AO report submitted by an Agreement State is included. It involved the loss of management and procedural control of a radioactive source. (Due to publication schedule constraints, NRC was unable to include all of the AO information received from the Agreement States. Any Agreement State information that was not included in this report will be published in the next quarterly report. The report also contains updates of six AOs previously reported by NRC licensees and three AOs previously reported by Agreement State licensees. Two "Other Events of Interest" concerning nuclear power reactors are also reported. One involved the fracture of a frozen pipe at Dresden Unit 1 with a consequent release of water, and the other involved the possible deliberate exposure of a contract laborer to radiation at Quad Cities Nuclear Power Station.

NUREG-0090 V17 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. October-December 1994. * Office for Analysis & Evaluation of Operational Data, Director. May 1995. 39pp. 9511140069. 86174:317.

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 occurrences to be made to Congress. This report provides a description of those incidents and events that have been determined to be AOs during the period of October 1 through December 31, 1994. This report addresses four AOs at NRC-licensed facilities. These occurrences involved the following: a generic concern relating to core shroud cracking in boiling water reactors; recurring incidents of administering higher doses than procedurally allowed for diagnostic imaging at a single facility; one medical teletherapy misadministration and one medical brachytherapy misadministration. Agreement States submitted four AO reports. These four occurrences involved the following: one major contamination at a commercial facility; two medical brachytherapy misadministrations; and one medical teletherapy misadministration. The report

2 Main Citations and Abstracts

also contains updates of seven AOs previously reported by the Agreement States. Two "Other Events of Interest" are also being reported. These occurrences involved the operability of safety relief valves at a nuclear powerplant, and an error in the installation process of a Leksell Gamma Knife ** Teletherapy unit that resulted in an operational failure.

NUREG-0090 V18 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. January-March 1995. * Office for Analysis & Evaluation of Operational Data, Director. July 1995. 25pp. 9507250296. 84831:118.

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 occurrences to be made to Congress. This report provides a description of those incidents and events that have been determined to be AOs during the period of January 1 through March 31, 1995. This report addresses one AO at an NRC-licensed facility which involved a medical brachytherapy misadministration. The report also contains updates of one AO previously reported by an NRC licensee and three AOs previously reported by the Agreement States. No "Other Events of Interest" items are being reported.

NUREG-0304 V19 N03: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Third Quarter 1994, July-September. * Division of Freedom of Information & Publications Services (Post 940714). December 1994. 62pp. 9502080121. 82660:001.

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 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1994. * Division of Freedom of Information & Publications Services (Post 940714). March 1995. 133pp. 9504180345. 83566:063.

See NUREG-0304, V19, N03 abstract.

NUREG-0304 V20 N01: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1995, January-March. * Division of Freedom of Information & Publications Services (Post 940714). July 1995. 48pp. 9508160248. 85041:277.

See NUREG-0304, V19, N03 abstract.

NUREG-0304 V20 N02: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Second Quarter 1995, April-June. * Division of Freedom of Information & Publications Services (Post 940714). September 1995. 48pp. 9510270358. 86009:180.

See NUREG-0304, V19, N03 abstract.

NUREG-0325 R18: U.S. NUCLEAR REGULATORY COMMISSION ORGANIZATION CHARTS AND FUNCTIONAL STATEMENTS. July 23, 1995. * Ofc of Personnel (Post 870413). July 1995. 67pp. 9511060194. 86094:255.

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

NUREG-0383 V01 R18: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Report Of NRC Approved Packages. * Office of Nuclear Material Safety & Safeguards. October 1995. 619pp. 9511160279. 86235:001.

The purpose of this directory is to make available a convenient source of information on packagings which have been ap-

proved by the U.S. Nuclear Regulatory Commission. To assist in identifying packaging, an index by Model Number and corresponding Certificate of Compliance Number is included at the front of Volumes 1 and 2. An alphabetical listing by user name is included in the back of Volume 3 of approved QA programs. The reports include a listing of all users of each package design and approved QA programs prior to the publication date.

NUREG-0383 V02 R18: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Certificates Of Compliance. * Office of Nuclear Material Safety & Safeguards. October 1995. 572pp. 9511160271. 86231:001.

See NUREG-0383, V01, R18 abstract.

NUREG-0383 V03 R15: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Report Of NRC Approved Quality Assurance Programs for Radioactive Materials Packages. * Office of Nuclear Material Safety & Safeguards. October 1995. 155pp. 9511160266. 86230:001.

See NUREG-0383, V01, R18 abstract.

NUREG-0390 V09 N01: TOPICAL REPORT REVIEW STATUS. * Office of Nuclear Reactor Regulation (Post 941001). February 1995. 44pp. 9503140388. 83068:272.

This report provides industry with procedures for submitting topical reports, guidance on how the U.S. Nuclear Regulatory Commission (NRC) processes and responds to topical report submittals, and an accounting, with review schedules, of all topical reports currently accepted for review by the NRC. This report is published semiannually.

NUREG-0430 V14: LICENSED FUEL FACILITY STATUS REPORT. Inventory Difference Data. July 1, 1993 - June 30, 1994. (Gray Book II) JOY, D.R. Office of Nuclear Material Safety & Safeguards. March 1995. 18pp. 9505100290. 83902:267.

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: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority) * Associate Director for Advanced Reactors & License Renewal (ADAR) (Post 941001). April 1995. 366pp. 9505180344. 83965:118.

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 (WBN) Plant. 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 nonradiological 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 (SAMDA) and concluded that none of the SAMDA, beyond the three procedural changes that the applicant committed to

implement, would be cost-beneficial for further mitigating environmental impacts.

NUREG-0525 V02 R03: SAFEGUARDS SUMMARY EVENT LIST (SSEL). January 1, 1990 Through December 31, 1994. FADDEN, M.; YARDUMIAN, J. Operations Branch. July 1995. 140pp. 9511060198. 86103:096.

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 V16 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30, 1994. * Division of Freedom of Information & Publications Services (Post 940714). January 1995. 307pp. 9502080222. 82662:001.

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

NUREG-0540 V16 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. December 1-31, 1994. * Division of Freedom of Information & Publications Services (Post 940714). February 1995. 319pp. 9503150148. 83100:285.

See NUREG-0540,V16,N11 abstract.

NUREG-0540 V17 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31, 1995. * Division of Freedom of Information & Publications Services (Post 940714). March 1995. 336pp. 9504120107. 83475:001.

See NUREG-0540,V16,N11 abstract.

NUREG-0540 V17 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. February 1-28, 1995. * Division of Freedom of Information & Publications Services (Post 940714). April 1995. 337pp. 9504250369. 83629:001.

See NUREG-0540,V16,N11 abstract.

NUREG-0540 V17 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. March 1-31, 1995. * Division of Freedom of Information & Publications Services (Post 940714). May 1995. 360pp. 9506020497. 84148:024.

See NUREG-0540,V16,N11 abstract.

NUREG-0540 V17 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30, 1995. * Division of Freedom of Information & Publications Services (Post 940714). June 1995. 207pp. 9506280447. 84468:129.

See NUREG-0540,V16,N11 abstract.

NUREG-0540 V17 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. May 1-31, 1995. * Division of Freedom of Information & Publications Services (Post 940714). July 1995. 300pp. 9511140075. 86175:001.

See NUREG-0540,V16,N11 abstract.

NUREG-0540 V17 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. June 1-30, 1995. * Division of Freedom of Information & Publications Services (Post 940714). August 1995. 276pp. 9508160276. 85041:001.

See NUREG-0540,V16,N11 abstract.

NUREG-0540 V17 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. July 1-31, 1995. * Division of Freedom of Information & Publications Services (Post 940714). September 1995. 278pp. 9510030198. 85664:001.

See NUREG-0540,V16,N11 abstract.

NUREG-0540 V17 N08: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. August 1-31, 1995. * Division of Freedom of Information & Publications Services (Post 940714). October 1995. 431pp. 9511060215. 86093:130.

See NUREG-0540,V16,N11 abstract.

NUREG-0540 V17 N09: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. September 1-30, 1995. * Division of Freedom of Information & Publications Services (Post 940714). November 1995. 346pp. 9512050194. 86412:001.

See NUREG-0540,V16,N11 abstract.

NUREG-0540 V17 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. October 1-31, 1995. * Division of Freedom of Information & Publications Services (Post 940714). December 1995. 344pp. 9601030180. 86689:001.

See NUREG-0540,V16,N11 abstract.

NUREG-0700 R01 DFC: HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE. Draft Report For Comment. O'HARA, J.M.; BROWN, W.S.; STUBLER, W.F.; et al. Brookhaven National Laboratory. February 1995. 496pp. 9503270312. 83269:001.

NUREG-0700, Rev. 1, provides human factors engineering (HFE) guidance to the U.S. Nuclear Regulatory Commission staff for its: (1) review of the human system interface (HSI) design submittals prepared by licensees or applicants for a license or design certification of commercial nuclear power plants, and (2) performance of HSI reviews that could be undertaken as part of an inspection or other type of regulatory review involving HSI design or incidents involving human performance. It consists of two major parts. Part 1 describes those aspects of the HSI design review process that are important to the identification and resolution of human engineering discrepancies that could adversely affect plant safety. Guidance is provided that could be used by the staff to review an applicant's HSI design review process. Part 1 could also be used by the staff to guide the development of an HSI design review plan, e.g., as part of an inspection activity. Part 2, "Guidelines for Human Factors Engineering Reviews," provides detailed HFE guidelines for the assessment of HSI design implementations.

NUREG-0713 V15: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES, 1993. Twenty-Sixth Annual Report. RADDATZ, C.T. Division of Regulatory Applications (Post 941217). HAGEMEYER, D. Science Applications International Corp. (formerly Science Applications, Inc.). January 1995. 307pp. 9502080242. 82661:001.

This report summarizes the occupational radiation exposure information that has been reported to the NRC's Radiation Exposure Information Reporting System (REIRS) by nuclear power facilities and certain other categories of NRC licensees during the years 1969 through 1993. The bulk of the data presented in the report was obtained from annual radiation exposure reports submitted in accordance with the requirements of 10 CFR 20.407 and the technical specifications of nuclear power plants. Data on workers terminating their employment at certain NRC licensed facilities were obtained from reports submitted pursuant to 10 CFR 20.408. The 1993 annual reports submitted by about 360 licensees indicated that approximately 189,711 individuals were monitored, 169,872 of whom were monitored by nuclear power facilities. They incurred an average individual dose of 0.16 rem (cSv) and an average measurable dose of about 0.31 (cSv). Termination radiation exposure reports were analyzed to reveal that about 99,749 individuals completed their employment with one or more of the 360 covered licensees during 1993. Some 91,000 of these individuals terminated from

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power reactor facilities, and about 12,685 of them were considered to be transient workers who received an average dose of 0.49 rem (cSv).

NUREG-0725 R10: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL. * Division of Industrial & Medical Nuclear Safety (Post 870729). April 1995. 30pp. 9505180571. 83980:001.

This circular has been prepared to provide information on the shipment of irradiated reactor fuel (spent fuel) subject to regulation by the Nuclear Regulatory Commission (NRC), and to meet the requirements of Public Law 96-295. The report provides a brief description of NRC authority for certain aspects of transporting spent fuel. It provides descriptive statistics on spent fuel shipments regulated by the NRC from 1979 to 1994. It also lists detailed highway and railway segments used within each state from January 1, 1993, through December 31, 1994.

NUREG-0750 V40: NUCLEAR REGULATORY COMMISSION ISSUANCES. Opinions And Decisions Of The Nuclear Regulatory Commission With Selected Orders. July-December 1994. * Division of Freedom of Information & Publications Services (Post 940714). July 1995. 400pp. 9508300374. 85286:001.

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

NUREG-0750 V40 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-December 1994. * Division of Freedom of Information & Publications Services (Post 940714). March 1995. 40pp. 9504120101. 83474:001.

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 V40 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1994. Pages 169-318. * Division of Freedom of Information & Publications Services (Post 940714). February 1995. 149pp. 9503150152. 83107:001.

See NUREG-0750,V40 abstract.

NUREG-0750 V40 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1994. Pages 319-387. * Division of Freedom of Information & Publications Services (Post 940714). February 1995. 77pp. 9503150157. 83101:242.

See NUREG-0750,V40 abstract.

NUREG-0750 V41 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-March 1995. * Division of Freedom of Information & Publications Services (Post 940714). June 1995. 34pp. 9507060350. 84533:306.

See NUREG-0750,V40,I02 abstract.

NUREG-0750 V41 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-June 1995. * Division of Freedom of Information & Publications Services (Post 940714). September 1995. 65pp. 9510180192. 85849:001.

See NUREG-0750,V40,I02 abstract.

NUREG-0750 V41 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1995. Pages 1-69. * Division of Freedom of Information & Publications Services (Post 940714). March 1995. 75pp. 9503280211. 83302:178.

See NUREG-0750,V40 abstract.

NUREG-0750 V41 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1995. Pages 71-178. * Division of Freedom of Information & Publications Services (Post 940714). May 1995. 115pp. 9505230100. 84017:107.

See NUREG-0750,V40 abstract.

NUREG-0750 V41 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1995. Pages 179-243. * Division of Freedom of Information & Publications Services (Post 940714). May 1995. 72pp. 9506140062. 84297:130.

See NUREG-0750,V40 abstract.

NUREG-0750 V41 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1995. Pages 245-319. * Division of Freedom of Information & Publications Services (Post 940714). June 1995. 83pp. 9507120290. 84633:001.

See NUREG-0750,V40 abstract.

NUREG-0750 V41 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1995. Pages 321-380. * Division of Freedom of Information & Publications Services (Post 940714). July 1995. 68pp. 9507250293. 84831:050.

See NUREG-0750,V40 abstract.

NUREG-0750 V41 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1995. Pages 381-496. * Division of Freedom of Information & Publications Services (Post 940714). August 1995. 124pp. 9509080022. 85405:146.

See NUREG-0750,V40 abstract.

NUREG-0750 V42 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-September 1995. * Division of Freedom of Information & Publications Services (Post 940714). November 1995. 21pp. 9512080146. 86450:187.

See NUREG-0750,V40,I02 abstract.

NUREG-0750 V42 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1995. Pages 1-45. * Division of Freedom of Information & Publications Services (Post 940714). August 1995. 53pp. 9509080026. 85405:270.

See NUREG-0750,V40 abstract.

NUREG-0750 V42 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1995. Pages 47-97. * Division of Freedom of Information & Publications Services (Post 940714). October 1995. 58pp. 9511020372. 86091:056.

See NUREG-0750,V40 abstract.

NUREG-0750 V42 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1995. Pages 99-110. * Division of Freedom of Information & Publications Services (Post 940714). November 1995. 18pp. 9511270462. 86321:320.

See NUREG-0750,V40 abstract.

NUREG-0750 V42 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1995. Pages 111-180. * Division of Freedom of Information & Publications Services (Post 940714). December 1995. 194pp. 9601180208. 86806:161.

See NUREG-0750,V40 abstract.

NUREG-0837 V14 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. October-December 1994. STRUCKMEYER,R. Region 1 (Post 820201). March 1995. 329pp. 9503170312. 83147:001.

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 1994.

NUREG-0837 V15 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. January-March 1995. STRUCKMEYER,R. Region 1 (Post 820201). May 1995. 230pp. 9506140068. 84298:001.

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 1995.

NUREG-0837 V15 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. April-June 1995. STRUCKMEYER,R. Region 1 (Post 820201). August 1995. 230pp. 9508300331. 85287:060.

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 1995.

NUREG-0837 V15 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. July-September 1995. STRUCKMEYER, R. Region 1 (Post 820201). December 1995. 231pp. 9601290199. 86879:051.

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 1995.

NUREG-0847 S15: 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. June 1995. 189pp. 9507120275. 84634:001.

Supplement No. 15 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. 14 was issued, and (2) matters that the staff had under review when Supplement No. 14 was issued.

NUREG-0847 S16: 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. Office of Nuclear Reactor Regulation (Post 941001). September 1995. 109pp. 9510270354. 86009:069.

Supplement No. 16 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. 15 was issued, and (2) matters that the staff had under review when Supplement No. 15 was issued.

NUREG-0847 S17: 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. Office of Nuclear Reactor Regulation (Post 941001). October 1995. 70pp. 9511020395. 86090:277.

Supplement No. 17 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. 16 was issued, and (2) matters that the staff had under review when Supplement No. 16 was issued.

NUREG-0847 S18: 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. Office of Nuclear Reactor Regulation (Post 941001). October 1995. 174pp. 9511290117. 86354:001.

Supplement No. 18 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. 17 was issued, and (2) matters that the staff had under review when Supplement No. 17 was issued.

NUREG-0847 S19: 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. Office of Nuclear Reactor Regulation (Post 941001). November 1995. 96pp. 9512040337. 86385:257.

Supplement No. 19 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. 18 was issued, and (2) matters that the staff had under review when Supplement No. 18 was issued.

NUREG-0933 S18: A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT, R. Division of Engineering Technology (Post 941217). April 1995. 266pp. 9505190010. 83981:001.

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-0933 S19: A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT, R. Division of Engineering Technology (Post 941217). December 1995. 414pp. 9601290268. 86878:001.

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 V13 N03: NRC REGULATORY AGENDA. Semiannual Report, July-December 1994. * Division of Freedom of Information & Publications Services (Post 940714). February 1995. 56pp. 9503140450. 83071:106.

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 semiannually.

NUREG-0936 V14 N01: NRC REGULATORY AGENDA. Semiannual Report, January-June 1995. * Division of Freedom of Information & Publications Services (Post 940714). September 1995. 56pp. 9510030203. 85663:193.

See NUREG-0936, V13, N03 abstract.

6 Main Citations and Abstracts

NUREG-0940 V13 N4 P1: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED REACTOR LICENSEES. Quarterly Progress Report, October-December 1994. * Ofc of Enforcement (Post 870413). February 1995. 360pp. 9503140428. 83071:171.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (October - December 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 V13 N4 P2: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MEDICAL LICENSEES. Quarterly Progress Report, October-December 1994. * Ofc of Enforcement (Post 870413). February 1995. 300pp. 9503140434. 83077:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (October - December 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 V13 N4 P3: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MATERIAL LICENSEES (NON-MEDICAL). Quarterly Progress Report, October-December 1994. * Ofc of Enforcement (Post 870413). February 1995. 360pp. 9503140438. 83075:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (October - December 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-0940 V14 N1 P1: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, REACTOR LICENSEES. Quarterly Progress Report, January-March 1995. * Ofc of Enforcement (Post 870413). May 1995. 360pp. 9506160009. 84333:017.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January - March 1995) 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 V14 N1 P2: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, MEDICAL LICENSEES. Quarterly Progress Report, January-March 1995. * Ofc of Enforcement (Post 870413). May 1995. 300pp. 9506140141. 84301:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January - March 1995) 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 V14 N1 P3: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, MATERIAL LICENSEES (NON-MEDICAL). Quarterly Progress Report, January-March 1995. * Ofc of Enforcement (Post 870413). May 1995. 439pp. 9506160045. 84331:306.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January - March 1995) 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-0940 V14 N2 P1: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, INDIVIDUAL ACTIONS. Quarterly Progress Report, April-June 1995. * Ofc of Enforcement (Post 870413). August 1995. 226pp. 9509080019. 85404:138.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1995) and includes copies of Orders sent by the Nuclear Regulatory Commission to individuals 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. The Commission believes this information may be useful to licensees in making employment decisions.

NUREG-0940 V14 N2 P2: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, REACTOR LICENSEES. Quarterly Progress Report, April-June 1995. * Ofc of Enforcement (Post 870413). August 1995. 189pp. 9509080016. 85405:016.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1995) 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 V14 N2 P3: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MATERIAL LICENSEES. Quarterly Progress Report, April-June 1995. * Ofc of Enforcement (Post 870413). August 1995. 223pp. 9509130141. 85432:111.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1995) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to material licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

NUREG-0980 V01 N03: NUCLEAR REGULATORY LEGISLATION. 103D Congress. * Office of the General Counsel (Post 860701). August 1995. 542pp. 9511090181. 86131:114.

This document is a compilation of nuclear regulatory legislation and other relevant material through the 103d Congress, 2d Session. This compilation has been prepared for use as a resource document, which the NRC intends to update at the end

of every Congress. The contents of NUREG-0980 include The Atomic Energy Act of 1954, as amended; Energy Reorganization Act of 1974, as amended; Uranium Mill Tailings Radiation Control Act of 1978; Low-Level Radioactive Waste Policy Act; Nuclear Waste Policy Act of 1982; and NRC Authorization and Appropriations Acts. Other materials included are statutes and treaties on export licensing, nuclear non-proliferation, and environmental protection.

NUREG-0980 V02 N03: NUCLEAR REGULATORY LEGISLATION. 103d Congress. * Office of the General Counsel (Post 860701). August 1995. 469pp. 9511090195. 86130:001. See NUREG-0980,V01,N03 abstract.

NUREG-1065 R02: ACCEPTABLE STANDARD FORMAT AND CONTENT FOR THE FUNDAMENTAL NUCLEAR MATERIAL CONTROL (FNMC) PLAN REQUIRED FOR LOW-ENRICHED URANIUM FACILITIES. JOY,D.R. Division of Fuel Cycle Safety & Safeguards (Post 930207). December 1995. 62pp. 9601290207. 86892:210.

This report documents a standard format suggested by the NRC for use in preparing fundamental nuclear material control (FNMC) plans as required by the Low Enriched Uranium Reform Amendments (10CFR 74.31). This report also describes the necessary contents of a comprehensive plan and provides example acceptance criteria which are intended to communicate acceptable means of achieving the performance capabilities of the Reform Amendments. By using the suggested format, the licensee or applicant will minimize administrative problems associated with the submittal, review and approval of the FNMC plan. Preparation of the plan in accordance with this format will assist the NRC in evaluating the plan and in standardizing the review and licensing process. However, conformance with this guidance is not required by the NRC. A license applicant who employs a format that provides a equal level of completeness and detail may use their own format. This document is also intended for providing guidance to licensees when making revisions to their FNMC plan.

NUREG-1100 V11: BUDGET ESTIMATES. Fiscal Years 1996-1997. * Division of Budget & Analysis (Post 890205). February 1995. 201pp. 9504180341. 83565:223.

This report contains the fiscal year budget justification to Congress. The budget provides estimates for salaries and expenses and for the Office of the Inspector General for fiscal years 1996 and 1997.

NUREG-1122 R01: KNOWLEDGE AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS: PRESSURIZED WATER REACTORS. * Office of Nuclear Reactor Regulation (Post 941001). August 1995. 500pp. 9603280103.

This document provides the basis for the development of content-valid licensing examinations for reactor operators and senior reactor operators. The examinations developed using the PWR catalog will cover those topics listed under Title 10, "Code of Federal Regulations," Part 55. The PWR catalog contains approximately 5100 knowledge and ability (K/A) statements for reactor operators and senior reactor operators. The catalog is organized into six major sections: Catalog Organization; Generic Knowledge and Abilities; Plant Systems; Emergency and Abnormal Plant Evolutions; Components and Theory.

NUREG-1123 R01: KNOWLEDGE AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS: BOILING WATER REACTORS. * Office of Nuclear Reactor Regulation (Post 941001). August 1995. 412pp. 9509250374. 85579:001.

This document provides the basis for the development of content-valid licensing examinations for reactor operators and senior reactor operators. The examinations developed using the BWR catalog will cover those topics listed under Title 10, Code of Federal Regulations, Part 55. The BWR catalog contains approximately 7,000 knowledge and ability (K/A) statements for reactor operators and senior reactor operators. Each K/A statement has been rated for its importance to safe operation of the

plant in a manner ensuring personnel and public health and safety. The BWR K/A catalog is organized into six major sections: Organization of the Catalog; Plant Wide Generic Knowledge and Abilities; Plant Systems Grouped by Safety Functions; Emergency and Abnormal Plant Evolutions; Components; and Theory.

NUREG-1125 V16: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS. 1994 Annual. * ACRS - Advisory Committee on Reactor Safeguards. April 1995. 115pp. 9505190009. 83980:082.

This compilation contains 30 ACRS reports submitted to the Commission, or to the Executive Director for Operations, during calendar year 1994. 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 V11: U.S. NUCLEAR REGULATORY COMMISSION 1994 ANNUAL REPORT. * Office of Administration, Director (Post 940714). June 1995. 300pp. 9507170098. 84671:001.

This report covers the major activities, events, decisions, and planning that took place during Fiscal Year 1994 within the U.S. Nuclear Regulatory Commission (NRC) or involving the NRC.

NUREG-1266 V09: NRC SAFETY RESEARCH IN SUPPORT OF REGULATION - FY 1994. * Office of Nuclear Regulatory Research (Post 941217). June 1995. 100pp. 9507200206. 84750:001.

This report, the tenth 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 1994. The goal of the Office of Nuclear Regulatory Research (RES) is to ensure the availability of sound technical bases for timely rulemaking and related decisions in support of NRC regulatory/licensing/inspection activities. RES also has responsibilities related to the resolution of generic safety issues and to the review of licensee submittals regarding individual plant examinations. It is the responsibility of RES to conduct the NRC's rulemaking process, including the issuance of regulatory guides and rules that govern NRC licensed activities.

NUREG-1272 V08 N02: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA. 1993 Annual Report - Nuclear Materials. * Office for Analysis & Evaluation of Operational Data, Director. May 1995. 122pp. 9511140066. 86176:001.

The annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) is devoted to the activities performed during 1993. The report is published in two separate 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 non-reactor applications, such as personnel overexposures and medical misadministrations. Note that the subtitle of No. 2 has been changed from "Nonreactors" to "Nuclear Materials." Both reports also contain a discussion of the Incident Investigation Team program and summarize both the Incident Investigation Team and Augmented Inspection Team reports. Each volume contains a list of the AEOD reports issued for 1981-1993.

8 Main Citations and Abstracts

NUREG-1275 V11: OPERATING EXPERIENCE FEEDBACK REPORT - TURBINE-GENERATOR OVERSPEED PROTECTION SYSTEM. Commercial Power Reactors. ORNSTEIN, H.L. Division of Safety Programs (Post 870413). April 1995. 117pp. 9505180319. 83965:001.

The report presents the results of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) review of operating experience of main turbine-generator overspeed and overspeed protection systems. AEOD's study provides insight into the shortcomings in the design, operation, maintenance, testing, and human factors associated with turbine overspeed protection systems. It includes an indepth examination of the turbine overspeed event that occurred on November 9, 1991, at the Salem Unit 2 Nuclear Power Plant. It also provides information concerning actions taken by other utilities and the turbine manufacturers as a result of the Salem overspeed event. AEOD's study reviewed operating procedures and plant practices. It noted differences between turbine manufacturer designs and recommendations for operations, maintenance, and testing, and also identified significant variations in the manner that individual plants maintain and test their turbine overspeed protection systems.

NUREG-1280 R01: STANDARD FORMAT AND CONTENT ACCEPTANCE CRITERIA FOR THE MATERIAL CONTROL AND ACCOUNTING (MC&A) REFORM AMENDMENT. * Division of Fuel Cycle Safety & Safeguards (Post 930207). April 1995. 119pp. 9505180530. 83968:001.

In 1987 the NRC revised the material control and accounting requirements for NRC licensees authorized to possess and use a formula quantity (i.e., 5 formula kilograms or more) of strategic special nuclear material. Those revisions issued as 10CFR 47.51-59 require timely monitoring of in-process inventory and discrete items to detect anomalies potentially indicative of material losses. Timely detection and enhanced loss localization capabilities are beneficial to alarm resolution and also for material recovery in the event of an actual loss. NUREG-1280 was issued in 1987 to present criteria that could be used by applicants, licensees, and NRC license reviewers in the initial preparation and subsequent review of fundamental nuclear material control (FNMC) plans submitted in response to the Reform Amendment. This document is also intended for both licensees and license reviewers with respect to FNMC plan revisions. General performance objectives, system capabilities, process monitoring, item monitoring, alarm resolution, quality assurance, and accounting are addressed. This revision to NUREG-1280 is an expansion of the initial edition, which clarifies and expands upon several topics and addresses issues identified under Reform Amendment implementation experience.

NUREG-1307 R05: REPORT ON WASTE BURIAL CHARGES. Escalation Of Decommissioning Waste Disposal Costs At Low-Level Waste Burial Facilities. * Division of Regulatory Applications (Post 941217). August 1995. 59pp. 9509070078. 85404:079.

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 R01: LICENSE APPLICATION REVIEW PLAN FOR A GEOLOGIC REPOSITORY FOR SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE. Draft Review Plan. * Division of Waste Management (NMSS 940403). December 1995. 166pp. 9601290272. 86877:168.

The License Application Review Plan (LARP) provides guidance to the NRC staff for its review of the U.S. Department of Energy's (DOE's) license application to construct a mined geologic repository for the disposal of spent nuclear fuel and other high-level radioactive waste at Yucca Mountain. The LARP, Revision 0 was issued September 1994. Revision 0 represented the staff's initial efforts in developing the LARP and was comprised of both completed and outlined individual review plans. The LARP was and continues to be, however, a work in progress. This draft revision, designated Revision 1, represents the staff's latest efforts in the development of the LARP. Appendix D provides a status of the development of the individual review plans. Future revisions to the LARP, however, are uncertain due to a changing high-level waste program and associated budget constraints.

NUREG-1350 V07: NUCLEAR REGULATORY COMMISSION INFORMATION DIGEST. 1995 Edition. STADLER, L. Division of Budget & Analysis (Post 890205). March 1995. 67pp. 9505230149. 84017:040.

The Nuclear Regulatory Commission Information Digest (digest) provides a summary of information about the U.S. Nuclear Regulatory Commission (NRC), NRC's regulatory responsibilities, the activities NRC licenses, and general information on domestic and worldwide nuclear energy. The digest, published annually, is a compilation of nuclear- and NRC-related data and is designed to provide a quick reference to major facts about the agency and the industry it regulates. In general, the data cover 1975 through 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-1363 V06: ATOMIC SAFETY AND LICENSING BOARD PANEL BIENNIAL REPORT. Fiscal Years 1993 - 1994. * Atomic Safety & Licensing Board Panel. August 1995. 54pp. 9509070103. 85398:297.

In Fiscal Year 1993, the Atomic Safety and Licensing Board Panel ("the Panel") handled 30 proceedings. In Fiscal Year 1994, the Panel handled 36 proceedings. The cases addressed issues in the construction, operation, and maintenance of commercial nuclear power reactors and other activities requiring a license from the Nuclear Regulatory Commission. This report sets out the Panel's caseload during the year and summarizes, highlights, and analyzes how the wide-ranging issues raised in those proceedings were addressed by the Panel's judges and licensing boards.

NUREG-1415 V07 N02: OFFICE OF THE INSPECTOR GENERAL. Semiannual Report To Congress, October 1, 1994 - March 31, 1995. BARCHI, T.; NORTON, L.; GRODIN, M.; et al. Office of the Inspector General (Post 890417). June 1995. 50pp. 9506220103. 84394:337.

The Inspector General is required by the Inspector General Act of 1978, as amended, to prepare a "Semiannual Report" to the U.S. Congress that summarizes program activities. The 6-month reporting period ends March 31 and September 30. The Inspector General's report is submitted to the Chairman of the NRC not later than April 30 and October 31, respectively. The Chairman comments on the Inspector General'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 V08 N01: OFFICE OF THE INSPECTOR GENERAL. Semiannual Report To Congress, April 1, 1995 - September 30, 1995. BARCHI, T.; WATKINS, B.; GRODIN, M.; et al. Office of the Inspector General (Post 890417). December 1995. 44pp. 9601290161. 86882:226.

The Inspector General Act of 1978, as amended, requires that Inspectors General submit a "Semiannual Report to Congress" summarizing program activities. The Inspector General's report is submitted to the Chairman of the NRC not later than April 30, and October 31 for each reporting period. The Chairman comments on the report and prepares the NRC's Semiannual Report to Congress as required by the Act. The Chairman then submits the agency's report and the OIG's report no later than November 30 and May 31, respectively.

NUREG-1423 V05: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON NUCLEAR WASTE. July 1993 - June 1995. * Advisory Committee on Nuclear Waste. August 1995. 75pp. 9509130173. 85433:144.

This compilation contains 13 reports issued by the Advisory Committee on Nuclear Waste (ACNW) during the sixth and seventh years of its operation. The reports were submitted to the Chairman and Commissioners of the U.S. Nuclear Regulatory Commission. All reports prepared by the Committee have been made available to the public through the NRC Public Document Room and the U.S. Library of Congress.

NUREG-1430 V01 R01: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Specifications. * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 326pp. 9506220126. 84395:027.

This report documents the results of the combined effort of the NRC and the industry to produce improved Standard Technical Specifications (STS), Revision 1 for Babcock & Wilcox Plants. The changes reflected in Revision 1 resulted from the experience gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. This NUREG is the result of extensive public technical meetings and discussions between the Nuclear Regulatory Commission (NRC) staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, NSSS vendors, and the Nuclear Energy Institute (NEI). The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993. The improved STS will be used as the basis for individual nuclear power plant licensees to develop improved plant-specific technical specifications. This report contains three volumes. Volume 1 contains the Specifications for all chapters and sections of the improved STS. Volume 2 contains the Bases for Chapters 2.0 and 3.0, and Sections 3.1 - 3.3 of the improved STS. Volume 3 contains the Bases for Sections 3.4 - 3.9 of the improved STS.

NUREG-1430 V02 R01: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Bases (Sections 2.0 - 3.3). * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 300pp. 9506220131. 84396:001.

See NUREG-1430, V01, R01 abstract.

NUREG-1430 V03 R01: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Bases (Sections 3.4 - 3.9). * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 350pp. 9506220136. 84396:302.

See NUREG-1430, V01, R01 abstract.

NUREG-1431 V01 R01: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Specifications. * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 350pp. 9506280476. 84456:001.

This report documents the results of the combined effort of the NRC and the industry to produce improved Standard Technical Specifications (STS), Revision 1 for Westinghouse Plants. The changes reflected in Revision 1 resulted from the experi-

ence gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. This report is the result of extensive public technical meetings and discussions between the Nuclear Regulatory Commission (NRC) staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, NSSS vendors, and the Nuclear Energy Institute (NEI). The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993. The improved STS will be used as the basis for individual nuclear power plant licensees to develop improved plant-specific technical specifications. This report contains three volumes. Volume 1 contains the Specifications for all chapters and sections of the improved STS. Volume 2 contains the Bases for Chapters 2.0 and 3.0, and Sections 3.1 - 3.3 of the improved STS. Volume 3 contains the Bases for Sections 3.4 - 3.9 of the improved STS.

NUREG-1431 V02 R01: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Bases (Sections 2.0 - 3.3). * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 350pp. 9506280489. 84458:001.

See NUREG-1431, V01, R01 abstract.

NUREG-1431 V03 R01: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Bases (Sections 3.4 - 3.9). * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 500pp. 9506280506. 84459:001.

See NUREG-1431, V01, R01 abstract.

NUREG-1432 V01 R01: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS. Specifications. * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 442pp. 9506220142. 84397:326.

This report documents the results of the combined effort of the NRC and the industry to produce improved Standard Technical Specifications (STS), Revision 1 for Combustion Engineering Plants. The changes reflected in Revision 1 resulted from the experience gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. This report is the result of extensive public technical meetings and discussions between the Nuclear Regulatory Commission (NRC) staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, NSSS vendors, and the Nuclear Energy Institute (NEI). The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993. The improved STS will be used as the basis for individual nuclear power plant licensees to develop improved plant-specific technical specifications. This report contains three volumes. Volume 1 contains the Specifications for all chapters and sections of the improved STS. Volume 2 contains the Bases for Chapters 2.0 and 3.0, and Sections 3.1 - 3.3 of the improved STS. Volume 3 contains the Bases for Sections 3.4 - 3.9 of the improved STS.

NUREG-1432 V02 R01: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS. Bases (Sections 2.0 - 3.3). * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 450pp. 9506220171. 84399:040.

See NUREG-1432, V01, R01 abstract.

NUREG-1432 V03 R01: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS. Bases (Sections 3.4 - 3.9). * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 602pp. 9506220172. 84400:126.

See NUREG-1432, V01, R01 abstract.

NUREG-1433 V01 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS BWR/4. Specifications. * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 388pp. 9506290155. 84461:001.

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This report documents the results of the combined effort of the NRC and the industry to produce improved Standard Technical Specifications (STS), Revision 1 for General Electric BWR/4 Plants. The changes reflected in Revision 1 resulted from the experience gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. This report is the result of extensive public technical meetings and discussions between the Nuclear Regulatory Commission (NRC) staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, NSSS vendors, and the Nuclear Energy Institute (NEI). The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993. The improved STS will be used as the basis for individual nuclear power plant licensees to develop improved plant-specific technical specifications. This report contains three volumes. Volume 1 contains the Specifications for all chapters and sections of the improved STS. Volume 2 contains the Bases for Chapters 2.0 and 3.0, and Sections 3.1 - 3.3 of the improved STS. Volume 3 contains the Bases for Sections 3.4 - 3.10 of the improved STS.

NUREG-1433 V02 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS BWR/4.Bases (Sections 2.0 - 3.3). * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 338pp. 9506290159. 84462:025.
See NUREG-1433,V01,R01 abstract.

NUREG-1433 V03 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS BWR/4.Bases (Sections 3.4 - 3.10). * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 422pp. 9506290164. 84463:001.
See NUREG-1433,V01,R01 abstract.

NUREG-1434 V01 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6.Specifications. * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 412pp. 9506290222. 84466:001.

This report documents the results of the combined effort of the NRC and the industry to produce improved Standard Technical Specifications (STS), Revision 1 for General Electric BWR/6 Plants. The changes reflected in Revision 1 resulted from the experience gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. This report is the result of extensive public technical meetings and discussions between the Nuclear Regulatory Commission (NRC) staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, NSSS vendors, and the Nuclear Energy Institute (NEI). The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993. The improved STS will be used as the basis for individual nuclear power plant licensees to develop improved plant-specific technical specifications. This report contains three volumes. Volume 1 contains the Specifications for all chapters and sections of the improved STS. Volume 2 contains the Bases for Chapters 2.0 and 3.0, and Sections 3.1 - 3.3 of the improved STS. Volume 3 contains the Bases for Sections 3.4 - 3.10 of the improved STS.

NUREG-1434 V02 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6.Bases (Sections 2.0 - 3.3). * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 356pp. 9506290228. 84464:059.
See NUREG-1434,V01,R01 abstract.

NUREG-1434 V03 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6.Bases 3.4 - 3.10. * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 444pp. 9506290230. 84467:049.
See NUREG-1434,V01,R01 abstract.

NUREG-1435 S04: STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS. TMI Action Plan Requirements, Unresolved Safety Issues, Generic Safety Issues, Other Multiplant Action Issues. * Office of Nuclear Reactor Regulation (Post 941001). December 1994. 157pp. 9502080207. 82686:001.

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 requirements 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). Supplement 3, published in December, 1993, provided updated information as of September 30, 1993. This annual NUREG report provides updated information on TMI, USI, GSI and other MPAs as of September 30, 1994. The data contained in these NUREG reports are a product of the NRC's Safety Issues Management system (SIMS) data base, which is maintained by the Program 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 NUREG report is to serve as a follow-on to NUREG-0933, "A Prioritization of Generic Safety Issues," which tracks safety issues up until requirements are approved for imposition at licensed plants or until the NRC issues a request for action by licensees.

NUREG-1444 S01: SITE DECOMMISSIONING MANAGEMENT PLAN. FAUVER,D.N.; WEBER,M.F.; JOHNSON,T.C.; et al. Division of Waste Management (NMSS 940403). November 1995. 149pp. 9512150121. 86535:001.

The Nuclear Regulatory Commission (NRC) staff has identified 51 sites contaminated with radioactive material that require special attention to ensure timely decommissioning. While none of these sites represent an immediate threat to public health and safety, they have contamination that exceeds existing NRC criteria for unrestricted use. All of these sites require some degree of remediation, and several involve regulatory issues that must be addressed by the Commission before they can be released for unrestricted use and the applicable licenses terminated. This report contains the NRC staff's strategy for addressing the technical, legal, and policy issues affecting the timely decommissioning of the 51 sites and describes the status of decommissioning activities at the sites. This is supplement number one to NUREG-1444, which was published in October 1993.

NUREG-1464: NRC ITERATIVE PERFORMANCE ASSESSMENT PHASE 2. Development Of Capabilities For Review Of A Performance Assessment For A High-Level Waste Repository. WESCOTT,R.G.; LEE,M.P. Office of Nuclear Material Safety & Safeguards. MCCARTIN,T.J.; et al. Office of Nuclear Regulatory Research (Post 941217). October 1995. 535pp. 9511160289. 86233:001.

In order to better review a potential license application to construct and operate a geologic repository for spent nuclear fuel and high-level radioactive waste (HLW), the Nuclear Regulatory Commission staff (and its contractor) has expanded and

improved its capability to conduct performance assessments. This report documents the demonstration of the second phase of this capability. The demonstration made use of the scenario selection procedure developed by Sandia National Laboratories to provide a set of scenarios, with corresponding probabilities, for use in the consequence analysis of a potential HLW disposal site in unsaturated tuff. Models of release of radionuclides from the waste form and transport in ground water, air and by direct pathways provided preliminary estimates of releases to the accessible environment for a 10,000 year period. The input values of parameters necessary for the consequence models were sampled numerous times using Latin Hypercube Sampling from assumed probability distributions. The results from the consequence models were then used to generate Complementary Cumulative Distribution Functions (CCDFs) for either release to the accessible environment or effective dose equivalents to a target population. CCDFs were calculated for probabilistically significant combinations (scenarios) of four disruptive events; drilling, pluvial climate, seismicity and magmatism. Sensitivity and uncertainty analyses of the calculated releases and effective dose equivalents were also used to determine the importance of the parameters. Because of the preliminary nature of the analysis and the lack of an adequate data base, the results and conclusions presented in this report should be carefully interpreted. They should not be misconstrued to represent the actual performance of the proposed Yucca Mountain repository nor serve as an endorsement of the methods used.

NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS. SOFFER, L.; BURSON, S.B.; FERRELL, C.M.; et al. Division of Systems Technology (Post 941217). February 1995. 40pp. 9503140449. 83071:066.

In 1962 the U.S. Atomic Energy Commission published TID-14844, "Calculation of Distance Factors for Power and Test Reactors" which specified a release of fission products from the core to the reactor containment for a postulated accident involving "substantial meltdown of the core". This "source term", the basis for the NRC's Regulatory Guides 1.3 and 1.4, has been used to determine compliance with the NRC's reactor site criteria, 10 CFR Part 100 and to evaluate other important plant performance requirements. During the past 30 years substantial additional information on fission product releases has been developed based on significant severe accident research. This document utilizes this research by providing more realistic estimates of the "source term" release into containment, in terms of timing, nuclide types, quantities and chemical form, given a severe core-melt accident. This revised "source term" is to be applied to the design of future light water reactors (LWRs). Current LWR licensees may voluntarily propose applications based upon it.

NUREG-1470 V04: FINANCIAL STATEMENT FOR FISCAL YEAR 1994. * Office of the Controller (Post 890205). March 1995. 84pp. 9505180277. 83967:148.

The Chief Financial Officer's 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 annual financial statement was previously included in the Chief Financial Officer's Annual Report. This report is the fourth annual report for the NRC and includes an overview of the NRC, the audited principal financial statements and audit reports for fiscal year 1994, and supplemental financial and management information.

NUREG-1478 R01: NON-POWER REACTOR OPERATOR LICENSING EXAMINER STANDARDS. * Office of Nuclear Reactor Regulation (Post 941001). June 1995. 172pp. 9507060351. 84533:100.

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-1482: GUIDELINES FOR INSERVICE TESTING AT NUCLEAR POWER PLANTS. CAMPBELL, P.L. Office of Nuclear Reactor Regulation, Director (Post 870411). April 1995. 328pp. 9505030476. 83746:022.

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

NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Final Report. DEY, M. Division of Regulatory Applications (Post 941217). SKOBLAR, L.; CHEN, P.; et al. S. Cohen & Associates, Inc. September 1995. 300pp. 9510200161. 85903:001.

The Nuclear Regulatory Commission (NRC) is implementing an initiative to eliminate requirements that are marginal to safety and yet impose a significant regulatory burden on licensees. The containment leak-testing requirements for power reactors have been identified as one area where performance-based requirements could replace the current prescriptive requirements with only a marginal impact on safety. This technical support document (TSD) provides the technical bases for the NRC's rulemaking to revise leak-testing requirements for nuclear power reactors in 10 CFR Part 50, Appendix J. This report identifies alternatives to current containment testing requirements which would meet the NRC's Safety Goals and achieve greater efficiency in the use of resources. Changes in the allowable leak rate for containment and the testing frequencies for both integrated and local leak rate tests are evaluated in terms of both risk and cost impacts. The feasibility of applying statistically-based sampling techniques to local leak-rate testing, and the use of on-line monitoring systems to continuously monitor containment integrity are also evaluated.

NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Draft Report For Comment. DEY, M. Division of Regulatory Applications (Post 941217). SKOBLAR, L.; CHEN, P.; et al. S. Cohen & Associates, Inc. January 1995. 260pp. 9503150142. 83100:001.

Nuclear Regulatory Commission (NRC) is implementing an initiative to eliminate requirements that are marginal to safety and yet impose a significant regulatory burden on licensees. The containment leak-testing requirements for power reactors have been identified as one area where performance-based requirements could replace the current prescriptive requirements with only a marginal impact on safety. This technical support document (TSD) provides the technical bases for the NRC's rulemaking to revise leak-testing requirements for nuclear power re-

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actors in 10 CFR Part 50, Appendix J. This report identifies alternatives to current containment testing requirements which would meet the NRC's Safety Goals and achieve greater efficiency in the use of resources. Changes in the allowable leak rate for containment and the testing frequencies for both integrated and local leak rate tests are evaluated in terms of both risk and cost impacts. The feasibility of applying statistically-based sampling techniques to local leak-rate testing, and the use of on-line monitoring systems to continuously monitor containment integrity are also evaluated.

NUREG-1505 DRFT FC: A NONPARAMETRIC STATISTICAL METHODOLOGY FOR THE DESIGN AND ANALYSIS OF FINAL STATUS DECOMMISSIONING SURVEYS. Draft Report For Comment. GOGOLAK, C.V. Energy, Dept. of, Environmental Measurements Laboratory. HUFFERT, A.M.; POWERS, G.E. Division of Regulatory Applications (Post 941217). August 1995. 216pp. 9509200281. 85541:080.

This report describes a nonparametric statistical methodology for the design and analysis of final status decommissioning surveys in support of the proposed rulemaking on decommissioning. The techniques described are alternatives to the existing parametric statistical methodology in NRC draft NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination." Proposed nonparametric statistical methods for testing compliance with decommissioning criteria are provided for radionuclides which occur in natural background and for those that do not occur in natural background. The tests considered applicable are the Wilcoxon Signed Ranks test, Sign test, and Quantile test for the analysis of a single data set, and the Wilcoxon Rank Sum test and a Quantile test for comparing two independent data sets. An Elevated Measurement Comparison is also described to deal with any unusually high observations. This report contains information on the Data Quality Objectives process as it relates to the planning and analysis of final site surveys. The proposed process includes methods for determining the number of samples needed to obtain statistically valid comparisons with decommissioning criteria and the methods for conducting the statistical tests with the resulting sample data.

NUREG-1506 DRFT FC: MEASUREMENT METHODS FOR RADIOLOGICAL SURVEYS IN SUPPORT OF NEW DECOMMISSIONING CRITERIA. Draft Report For Comment. HUFFERT, A.M. Division of Regulatory Applications (Post 941217). MILLER, K.M. Energy, Dept. of, Environmental Measurements Laboratory. August 1995. 78pp. 9509200363. 85541:001.

This report describes proposed methodologies for measuring low-level radiation and radioactivity that could be used in conducting surveys associated with decommissioning of licensed NRC facilities. Guidance on survey planning within the context of the Data Quality Objective approach and on specific instrumentation for measurements of gross and nuclide-specific radiation and radioactivity is given. Scanning, direct measurements, and sampling are discussed in terms of the application to particular measurement locations. The basic survey meter techniques that are commonly used at present are outlined and more detailed information is given on the capabilities and application of in situ spectrometric techniques for providing high sensitivity for individual photon-emitting radionuclides. The use of various techniques in concert is recommended, as the different measurements, taken collectively, serve as a quality control check. The methodologies described provide the means to measure residual radionuclides at concentrations corresponding to the proposed decommissioning criteria which are in the range of 3 to 15 mrem per year for unrestricted release of a facility.

NUREG-1507 DRFT FC: MINIMUM DETECTABLE CONCENTRA-

TIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS. Draft Report For Comment. HUFFERT, A.M. Division of Regulatory Applications (Post 941217). ABELQUIST, E.W. Oak Ridge Associated Universities. BROWN, W.S. Brookhaven National Laboratory. August 1995. 160pp. 9509200367. 85540:156.

This report describes and quantitatively evaluates the effects of various factors on the detection sensitivity of commercially available portable field instruments being used to conduct radiological surveys in support of decommissioning. NRC is currently involved in a rulemaking effort to establish residual contamination criteria for release of facilities for restricted or unrestricted use. In support of that rulemaking, NRC has prepared a draft Generic Environmental Impact Statement (GEIS), consistent with the National Environmental Policy Act. The effects of this new rulemaking on the overall cost of decommissioning are among the many factors considered in the GEIS. The overall cost includes the costs of decontamination, waste disposal, and radiological surveys to demonstrate compliance with the applicable guidelines. An important factor affecting the costs of such surveys is the minimum detectable concentrations (MDCs) of field survey instruments in relation to the residual contamination guidelines. The purpose of this study was two-fold. First, the data were used to determine the validity of the theoretical MDCs used in the draft GEIS. Second, the results of the study, published herein, provide guidance to licensees for (a) selection and proper use of portable survey instruments and (b) understanding the field conditions and the extent to which the capabilities of those instruments can be limited. Such instruments as gas proportional, Geiger-Mueller, zinc sulfide, and sodium iodide detectors were evaluated.

NUREG-1514: GUIDANCE FOR A LARGE TABLETOP EXERCISE FOR A NUCLEAR POWER PLANT. WEINSTEIN, E.D.; BATES, E.F. Office for Analysis & Evaluation of Operational Data, Director. ADLER, M.V.; et al. Oak Ridge National Laboratory. March 1995. 78pp. 9504180482. 83570:007.

Tabletop exercises are held to discuss issues related to the response of organizations to an emergency event. This document describes in task format the planning, conduct, and reporting of lessons learned for a large interagency tabletop. A sample scenario, focus area, and discussion questions based on a simulated accident at a commercial nuclear power plant are provided.

NUREG-1516 DRFT FC: MANAGEMENT OF RADIOACTIVE MATERIAL SAFETY PROGRAMS AT MEDICAL FACILITIES. Draft Report For Comment. CAMPER, L.W.; SCHLUETER, J.; HENDERSON, P.; et al. Division of Industrial & Medical Nuclear Safety (Post 870729). January 1995. 178pp. 9502080136. 82663:152.

A Task Force originally composed of seven U.S. Nuclear Regulatory Commission and two Agreement State program staff developed the guidance contained in this report. The purpose of this report is to describe a systematic approach for effective management of radiation safety programs at medical facilities. This is accomplished by emphasizing the roles of institution executive management, radiation safety committee, and radiation safety officer. Various aspects of program management are discussed and include guidance on selecting the radiation safety officer, determining adequate resources for the program, the use of contractual services such as consultants and service companies, the conduct of audits, the roles of authorized users and supervised individuals, NRC's reporting and notification requirements, and a general description of how NRC's licensing, inspection, and enforcement programs work. Appendices provide detailed guidance on specific aspects of a radiation safety program and the glossary defines terms used throughout the report. The guidance contained herein does not represent new or proposed regulatory requirements and licensees will not be inspected against any portion of it. Additionally, regulatory com-

pliance with all applicable regulations is not assured by licensees who adopt any portion of, or apply the principles described in this report.

NUREG-1517: REPORT OF THE SOUTH TEXAS PROJECT ALLEGATIONS REVIEW TEAM. KOKAJKO, L.E.; SKAY, D.M.; WANG, H.-B.; et al. Office of Nuclear Reactor Regulation (Post 941001). March 1995. 222pp. 9504180323. 83565:001.

This report provides the results of the South Texas Project Allegations Review Team of the U.S. Nuclear Regulatory Commission. This team was formed to obtain and review allegations from individuals represented by three attorneys who had contacted Congressional staff members. The alleged were employed in various capacities at South Texas Project Electric Generating Station, licensed by Houston Lighting and Power Company, et al.; therefore, the allegations are confined to this site. The South Texas Project Allegations Review Team reviewed, referred, and dispositioned concerns related to discriminatory issues (harassment and intimidation), falsification of records and omission of information, and various technical issues. The team was able to substantiate certain technical issues of minor safety significance or regulatory concern at the South Texas Project facility, but it did not find widespread discriminatory practices such as harassment and intimidation.

NUREG-1519: SURFACE INTERACTIONS OF CESIUM AND BORIC ACID WITH STAINLESS STEEL. GROSSMAN, N. Division of Systems Technology (Post 941217). August 1995. 104pp. 9509130158. 85432:009.

In this report, the effects of cesium hydroxide and boric acid on oxidized stainless steel surfaces at high temperatures and near one atmosphere of pressure are investigated. This is the first experimental investigation of this chemical system. The experimental investigations were performed using a mass spectrometer and a mass electrobalance. Surfaces from the different experiments were examined using a scanning electron microscope to identify the presence of deposited species, and electron spectroscopy for chemical analysis to identify the species deposited on the surface. The analysis couples vaporization, deposition, and desorption of the compounds formed under conditions similar to what is expected during certain nuclear reactor accidents. Cesium deposits on an oxidized stainless steel surface at temperatures between 1000 and 1200 Kelvin and on stainless steel surfaces coated with boric oxide in the same temperature ranges. The mechanism for such deposition involved the chemical reaction between cesium and chromate. Some reevaporation in the cesium hydroxide-boric acid system was observed. Under the conditions given, boric acid will react with cesium hydroxide to form cesium metaborate. A model is proposed for this chemical reaction.

NUREG-1522: ASSESSMENT OF INSERVICE CONDITIONS OF SAFETY-RELATED NUCLEAR PLANT STRUCTURES. ASHAR, H.; BAGCHI, G. Office of Nuclear Reactor Regulation (Post 941001). June 1995. 116pp. 9508230244. 85127:001.

The report is a compilation from a number of sources of information related to the condition of structures and civil engineering features at operating nuclear power plants in the U.S. The most significant information came from the hands-on inspection of the six old plants (licensed prior to 1977) performed by the staff of the Civil Engineering and Geosciences Branch in the Division of Engineering of the Office of Nuclear Reactor Regulation. For the containment structures, most of the information related to the degraded conditions came from the licensees as part of the Licensing Event Report System (10 CFR 50.73), or as part of the requirement under limiting condition of operation of the plant-specific Technical Specifications. Most of the information related to the degradation of other structures and civil engineering features was extracted from the industry survey, the reported incidents, and the plant visits. The report discusses the condition of the structures and civil engineering features at operating nuclear power plants and provides information that

would help detect, alleviate, and correct the degraded conditions of the structures and civil engineering features.

NUREG-1525: ASSESSMENT OF THE NRC ENFORCEMENT PROGRAM. LIEBERMAN, J. Ofc of Enforcement (Post 870413). COBLENTZ, L. Walnut Creek Field Ofc, R4 (Post 940404). April 1995. 200pp. 9505040381. 83784:001.

On May 13, 1994, the Nuclear Regulatory Commission's (NRC's) Executive Director for Operations established a Review Team to Assess the NRC Enforcement Program. The team evaluated the current system, and solicited comments from various NRC Offices, other Federal agencies, members of industry, and the public. This report presents the team's assessment. The report summarizes current processes and suggests certain changes. It proposes: (1) increased clarity, focus, and simplicity in the enforcement program; (2) retention of four severity levels of violations, with a clear focus on safety; (3) holding enforcement conferences only when needed, clarifying their status as predecisional, and making open conferences the norm; (4) a streamlined civil penalty assessment process, with fewer decisional points and limited outcomes, and the use of discretion where appropriate; and (5) implementation changes to increase efficiency. Recommendations are given in each area.

NUREG-1526: LESSONS LEARNED FROM EARLY IMPLEMENTATION OF THE MAINTENANCE RULE AT NINE NUCLEAR POWER PLANTS. PETRONE, C.D.; CORREIA, R.P.; BLACK, S.C. Office of Nuclear Reactor Regulation (Post 941001). June 1995. 44pp. 9507060341. 84532:001.

This report summarizes the lessons learned from the nine pilot site visits that were performed to review early implementation of the maintenance rule using the draft NRC Maintenance Inspection Procedure. Licensees followed NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." In general, the licensees were thorough in determining which structures, systems, and components (SSCs) were within the scope of the maintenance rule at each site. The use of an expert panel was an appropriate and practical method of determining which SSCs are risk significant. When setting goals, all licensees considered safety but many licensees did not consider operating experience throughout the industry. Although required to do so, licensees were not monitoring at the system or train level the performance or condition for some systems used in standby service but not significant to risk. Most licensees had not established adequate monitoring of structures under the rule. Licensees established reasonable plans for doing periodic evaluations, balancing unavailability and reliability, and assessing the effect of taking equipment out of service for maintenance. However, these plans were not evaluated because they had not been fully implemented at the time of the site visits.

NUREG-1527: NRC'S OBJECT-ORIENTED SIMULATOR INSTRUCTOR STATION. GRIFFIN, J.I.; GRIFFIN, J.P. Technical Training Center. June 1995. 107pp. 9506280427. 84469:073.

As part of a comprehensive simulator upgrade program, the simulator computer systems associated with the Nuclear Regulatory Commission's (NRC) nuclear power plant simulators were replaced. Because the original instructor stations for two of the simulators were dependent on the original computer equipment, it was necessary to develop and implement new instructor stations. This report describes the Macintosh-based Instructor Stations developed by NRC engineers for the General Electric (GE) and Babcock and Wilcox (B&W) simulators.

NUREG-1528: RECONSTITUTION OF THE MANUAL CHAPTER 2512 CONSTRUCTION INSPECTION PROGRAM FOR WATTS BAR UNIT 1. PERANICH, M.; FREDRICKSON, P.; JAUDON, J. Region 2 (Post 820201). September 1995. 683pp. 9510310364. 86016:001.

This document provides information on the concepts, performance, and results of the Inspection Manual Chapter (MC) 2512 Light-Water Reactor Inspection Reconstitution Program for

the construction phase of Watts Bar Nuclear Plant, Unit 1 (WBNP 1). The U.S. Nuclear Regulatory Commission (NRC), Region II, conducted the reconstitution as a follow-up to the initial MC 2512 Inspection Program, completed in 1985. Through this initial inspection program, the NRC identified several problems with the quality of construction at the facility, as well as weaknesses in the corrective actions taken by the licensee to resolve those problems. Subsequently allegations and employees concerns echoed these findings, raising questions regarding construction quality. The NRC decided in 1994 that a real-time correlation or "reconstitution" was needed to verify that the final construction-related plant inspections met the requirements of the MC 2512 Inspection Program. The MC 2512 Reconstitution Program successfully validated completion of the WBNP 1 construction inspection program. It also inspires confidence in the effectiveness of the licensee's corrective actions in resolving construction problems and enhancing the quality of plant construction. Successful reconstitution is integral to the overall reasonable assurance assessment of the readiness for WBNP 1 to be licensed as an operating facility.

NUREG-1530: REASSESSMENT OF NRC'S DOLLAR PER PERSON-REM CONVERSION FACTOR POLICY. * Division of Regulatory Applications (Post 941217). December 1995. 23pp. 9601290154. 86877:334.

The U.S. Nuclear Regulatory Commission (NRC) has completed a review and analysis of its dollar per person-rem conversion factor policy. As a result of this review, the NRC has decided to adopt a \$2000 per person-rem conversion factor, subject it to present worth considerations, and limit its scope solely to health effects. This is in contrast to the previous policy and staff practice of using an undiscounted \$1000 per person-rem conversion factor that served as a surrogate for all offsite consequences (health and offsite property). The policy shift has been incorporated in "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," NUREG/BR-0058, Revision 2, November 1995.

NUREG-1535: INGESTION OF PHOSPHORUS-32 AT MASSACHUSETTS INSTITUTE OF TECHNOLOGY, CAMBRIDGE, MASSACHUSETTS, IDENTIFIED ON AUGUST 19, 1995. * Incident Investigation Team. December 1995. 129pp. 9512260311. 86626:001.

On Monday October 16, 1995, the Massachusetts Institute of Technology (MIT, the licensee) notified the U.S. Nuclear Regulatory Commission (NRC) of an incident involving ingestion of phosphorus-32 by a researcher at the MIT Center for Cancer Research. The licensee informed the NRC that a researcher had reported the incident on August 19. The licensee initially estimated the intake as 500 microcuries (19 MBq) and the dose as 4000 millirem (40 mSv) to the individual. On October 12, the licensee informed the researcher that its final intake estimate was 579 microcuries (21 MBq), just under the 600 microcuries (22 MBq) which would represent an overexposure. On October 17, the NRC established an Incident Investigation Team to investigate the case. NRC also contracted with Lawrence Livermore National Laboratory and Oak Ridge Institute for Science and Education to do independent dose assessments of the urine sample data and the whole-body data. The Team concluded that the licensee's final intake and dose estimates were in accordance with accepted scientific references and NRC guidance. However, recognizing the uncertainties involved in the use of models to simulate human characteristics, the Team determined the intake would be better characterized as likely falling within a range of 500 to 750 microcuries (19-28 MBq). An NRC medical consultant concluded that no symptoms or acute effects should be observed from an intake of this level.

NUREG-1600: GENERAL STATEMENT OF POLICY AND PROCEDURE FOR NRC ENFORCEMENT ACTIONS. Enforcement Policy. * Ofc of Enforcement (Post 870413). July 1995. 35pp. 9507250299. 84831:142.

This document includes the U.S. Nuclear Regulatory Commission's (NRC's or Commission's) revised General Statement of Policy and Procedure for Enforcement Actions (Enforcement Policy) as it was published in the Federal Register on June 30, 1995 (60 FR 34381). This document also includes the notice announcing the removal of the Enforcement Policy from the Code of Federal Regulations (60 FR 34380; June 30, 1995). The Enforcement Policy is a general statement of policy explaining the NRC's policies and procedures in initiating enforcement actions, and of the presiding officers and the Commission in reviewing these actions. This policy statement is applicable to enforcement in matters involving the radiological health and safety of the public, including employees' health and safety, the common defense and security, and the environment. This statement of general policy and procedure is published as NUREG-1600 to provide widespread dissemination of the Commission's Enforcement Policy. However, this is a policy statement and not a regulation. The Commission may deviate from this statement of policy and procedure as appropriate under the circumstances of a particular case.

NUREG/CP-0140 V01: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING. Plenary Session, Advanced Instrumentation & Control Hardware & Software, Human Factors Research, IPE & PRA. MONTELEONE, S. Brookhaven National Laboratory. April 1995. 403pp. 9505230106. 84016:001.

This three-volume report contains papers presented at the Twenty-Second Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, during the week of October 24-26, 1994. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from Finland, France, Italy, Japan, Russia 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-0140 V02: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING. Severe Accident Research, Thermal Hydraulic Research For Advanced Passive LWRs, High-Burnup Fuel Behavior. MONTELEONE, S. Brookhaven National Laboratory. April 1995. 430pp. 9505230109. 84018:001.

See NUREG/CP-0140, V01 abstract.

NUREG/CP-0140 V03: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING. Primary Systems Integrity, Structural And Seismic Engineering, Aging Research, Products And Applications. MONTELEONE, S. Brookhaven National Laboratory. April 1995. 265pp. 9505230115. 84019:068.

See NUREG/CP-0140, V01 abstract.

NUREG/CP-0141: PROCEEDINGS OF THE 23RD DOE/NRC NUCLEAR AIR CLEANING CONFERENCE. Held In Buffalo, New York, July 25-28, 1994. FIRST, M.W. Harvard School of Public Health, Boston, MA. February 1995. 800pp. 9503140367. CONF-940738. 83072:114.

This report contains the papers presented at the 23rd DOE/NRC Nuclear Air Cleaning Conference and the associated discussions. Major topics are: (1) nuclear air cleaning codes, (2) nuclear waste, (3) filters and filtration, (4) effluent stack monitoring, (5) gas processing, (6) adsorption, (7) air treatment systems, (8) source terms and accident analysis, and (9) fuel re-processing.

NUREG/CP-0142 V01: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7). Sessions 1-5. BLOCK, R.C.; FEINER, F. American Nuclear Society. September 1995. 509pp. 9509110196. 85399:001.

This volume includes papers presented at the 7th International Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-7) September 10-15, 1995 at Saratoga Springs, N.Y. The following subjects are discussed: progress in analytical and experimental work on the fundamentals of nuclear thermal-hydraulics, the development of advanced mathematical and numerical methods, and the application of advancements in the field in the development of novel reactor concepts. Also combined issues of thermal-hydraulics and reactor/power-plant safety, core neutronics and/or radiation.

NUREG/CP-0142 V02: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7). Sessions 6-11. BLOCK, R.C.; FEINER, F. American Nuclear Society. September 1995. 806pp. 9509110202. 85402:001.

See NUREG/CP-0142, V01 abstract.

NUREG/CP-0142 V03: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7). Sessions 12-16. BLOCK, R.C.; FEINER, F. American Nuclear Society. September 1995. 916pp. 9509110205. 85386:001.

See NUREG/CP-0142, V01 abstract.

NUREG/CP-0142 V04: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7). Sessions 17-24. BLOCK, R.C.; FEINER, F. American Nuclear Society. September 1995. 845pp. 9509110210. 85396:001.

See NUREG/CP-0142, V01 abstract.

NUREG/CP-0143: PROCEEDINGS OF THE THIRD INTERNATIONAL WORKSHOP ON THE IMPLEMENTATION OF ALARA AT NUCLEAR POWER PLANTS. Held At Hauppauge, Long Island, New York. KHAN, T.A. Brookhaven National Laboratory. March 1995. 822pp. 9503270314. BNL-NUREG-52440. 83266:001.

This report contains the papers presented and the discussions that took place at the Third International Workshop on ALARA Implementation at Nuclear Power Plants, held in Hauppauge, Long Island, New York from May 8-11, 1994. The workshop brought together scientists, engineers, health physicists, regulators, managers and others who are involved with occupational dose control and ALARA issues. One-hundred and seventy five persons from eleven countries attended the workshop. The countries represented were: Canada, Finland, France, Germany, Japan, Korea, Mexico, the Netherlands, Spain, Sweden, the United Kingdom and the United States. The workshop was organized into twelve sessions and three panel discussions. The topics were as follows: Session 1, Controlling Radiation Fields; Session 2, Panel Discussion on Recent Recommendations on Dose Limitation; Session 3, Presentations and Panel Discussion on ALARA in New Reactors; Session 4, Pathways to ALARA; Session 5, Panel Discussion on Economics Versus Excellence; Session 6, Short Presentations on ALARA Implementation; Session 7A, PWR and CANDU Presentations; Session 7B, BWR and Gas-Cooled Presentations 1; Session 8A, PWR and CANDU Presentations; Session 8B, BWR and Gas-Cooled Presentations; Session 9, Decommissioning of Nuclear Power Plants; Session 10, Decontamination of Nuclear Power Plants, and Session 11, Robotics and Remote Handling. The workshop was sponsored jointly by the U.S. Nuclear Regulatory Commission and the Brookhaven National Laboratory's ALARA Center.

NUREG/CP-0144 V01: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Summary. TORTORELLI, J.P. Idaho National Engineering Laboratory. August 1995. 122pp. 9508230352. INEL-94/0111. 85118:013.

A workshop was held at the Idaho National Engineering Laboratory, August 16-18, 1994, on the topic of risk assessment on medical devices that use radioactive isotopes. Its purpose was to review past efforts to develop a risk assessment methodology to evaluate these devices and to develop a program plan and scoping document for future development. This report presents a summary of that workshop, related technical papers, presentation material, and a transcript of the workshop. Participants included experts in the fields of radiation oncology, medical physics, risk assessment, human-error analysis, and human factors. Staff from the U.S. Nuclear Regulatory Commission (NRC) associated with the regulation of medical uses of radioactive materials and with research into risk-assessment methods participated in the workshop. The workshop participants concurred in NRC's intended use of risk assessment as an important technology in the development of regulations for the medical use of radioactive material and encouraged the NRC to proceed rapidly with a pilot study. Specific recommendations are included in the executive summary and the body of this report.

NUREG/CP-0144 V02: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Supporting Documents. TORTORELLI, J.P. Idaho National Engineering Laboratory. August 1995. 330pp. 9508230354. INEL-94/0111. 85119:001.

See NUREG/CP-0144, V01 abstract.

NUREG/CP-0145: WORKSHOP ON DEVELOPING SAFE SOFTWARE. Held At Hotel Del Coronado, San Diego, CA, July 22-23, 1992. LAWRENCE, J.D. Lawrence Livermore National Laboratory. November 1994. 30pp. 9502080047. 82644:318.

The Workshop on Developing Safe Software was held July 22-23, 1992, at the Hotel del Coronado, San Diego, California. The purpose of the workshop was to have four world experts discuss among themselves software safety issues which are of interest to the U.S. Nuclear Regulatory Commission. These issues concern the development of software systems for use in nuclear power plant protection systems. The workshop comprised four sessions. Wednesday morning, July 22, consisted of presentations from each of the four panel members. On Wednesday afternoon, the panel members went through a list of possible software development techniques and commented on them. The Thursday morning, July 23, session consisted of an extended discussion among the panel members and the observers from the NRC. A final session on Thursday afternoon consisted of a discussion among the NRC observers as to what was learned from the workshop.

NUREG/CP-0146: PROCEEDINGS OF THE WORKSHOP ON GATE VALVE PRESSURE LOCKING AND THERMAL BINDING. BROWN, E.J. Division of Safety Programs (Post 870413). July 1995. 268pp. 9508090040. 84964:001.

The purpose of the Workshop on Gate Valve Pressure Locking and Thermal Binding was to discuss pressure locking and thermal binding issues that could lead to inoperable gate valves in both boiling water and pressurized water reactors. The goal was to foster exchange of information to develop the technical bases to understand the phenomena, identify the components that are susceptible, discuss actual events, discuss the safety significance, and illustrate known corrective actions that can prevent or limit the occurrence of pressure locking or thermal binding. The presentations were structured to cover U.S. Nuclear Regulatory Commission staff evaluation of operating experience and planned regulatory activity; industry discussions of specific events, including foreign experience, and efforts to determine causes and alleviate the affects; and valve vendor ex-

perience and recommended corrective action. The discussions indicated that identifying valves susceptible to pressure locking and thermal binding was a complex process involving knowledge of components, systems, and plant operations. The corrective action options are varied and straightforward.

NUREG/CP-0147: PROCEEDINGS OF THE WORKSHOP ON THE ROLE OF NATURAL ANALOGS IN GEOLOGIC DISPOSAL OF HIGH-LEVEL NUCLEAR WASTE. Held In San Antonio, Texas, July 22-25, 1991. KOVACH, L.A. Division of Regulatory Applications (Post 941217). MURPHY, W.M. Southwest Research Institute. September 1995. 131pp. 9510310349. CNWRA 93-020. 86024:001.

A Workshop on the Role of Natural Analogs in Geologic Disposal of High-Level Nuclear Waste was held in San Antonio, Texas on July 22-25, 1991. The proceedings comprise seventeen papers submitted by participants at the workshop. A series of papers addresses the relation of natural analog studies to the regulation, performance assessment, and licensing of a geologic repository. Applications of reasoning by analogy are illustrated in papers on the role of natural analogs in studies of earthquakes, petroleum, and mineral exploration. A summary is provided of a recently completed, internationally coordinated natural analog study at Pogos de Caldas, Brazil. Papers also cover problems and applications of natural analog studies in four technical areas of nuclear waste management: waste form and waste package, near-field processes and environment, far-field processes and environment, and volcanism, and tectonics. Summaries of working group deliberations in these four technical areas provide reviews and proposals for natural analog applications.

NUREG/CP-0148: TRANSACTIONS OF THE TWENTY-THIRD WATER REACTOR SAFETY INFORMATION MEETING. MONTELEONE, S. Brookhaven National Laboratory. September 1995. 76pp. 9510060240. 85746:182.

This report contains summaries of papers on reactor safety research to be presented at the 23rd Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel, Bethesda, Maryland, October 23-25, 1995. The summaries briefly describe the program 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-0200 V1 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Control Modules. PARKS, C.V. Oak Ridge National Laboratory. April 1995. 781pp. 9506010500. ORNL/NUREG/CSD2. 84122:213.

SCALE-a modular code system for Standardized Computer Analyses Licensing Evaluation-has been developed by Oak Ridge National Laboratory at the request of the U.S. Nuclear Regulatory Commission. The SCALE system utilizes well-established computer codes and methods within standard analysis sequences that (1) allow an input format designed for the occasional user and/or novice, (2) automate the data processing and coupling between modules, and (3) provide accurate and reliable results. System development has been directed at problem-dependent cross-section processing and analysis of criticality safety, shielding, heat transfer, and depletion/decay problems. Since the initial release of SCALE in 1980, the code system has been heavily used for evaluation of nuclear fuel facility and package designs. This revision documents Version 4.2 of the system.

NUREG/CR-0200 V2P1R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Functional Modules F1-F8. PARKS, C.V. Oak Ridge National Laboratory. April 1995. 670pp. 9506010509. ORNL/NUREG/CSD2. 84124:266. See NUREG/CR-0200, V01, R04 abstract.

NUREG/CR-0200 V2P2R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Functional Modules F9-F16. PARKS, C.V. Oak Ridge National Laboratory. April 1995. 869pp. 9506010521. ORNL/NUREG/CSD2. 84128:273. See NUREG/CR-0200, V01, R04 abstract.

NUREG/CR-0200 V3 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Miscellaneous. PARKS, C.V. Oak Ridge National Laboratory. April 1995. 793pp. 9506010519. ORNL/NUREG/CSD2. 84126:208. See NUREG/CR-0200, V01, R04 abstract.

NUREG/CR-2850 V13: DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1991. BAKER, D.A. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1995. 192pp. 9505090280. PNL-4221. 83849:001.

Population and individual radiation dose commitments have been estimated from reported radionuclide releases from commercial power reactors operating during 1991. Fifty-year dose commitments for a one-year exposure from both liquid and atmospheric releases were calculated for four population groups (infant, child, teenager, 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 that 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 22 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 1.2 person-rem. The total population dose for all sites was estimated at 88 person-rem for the 130 million people considered at risk. The individual dose commitments estimated for all sites below the Appendix I design objectives.

NUREG/CR-2907 V13: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1992. TICHLER, J.; DOTY, K.; LUCADAMO, K. Brookhaven National Laboratory. August 1995. 350pp. 9509130181. BNL-NUREG-51581. 85431:001.

Releases of radioactive materials in airborne and liquid effluents from commercial light water reactors during 1992 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 1992 release data are summarized in tabular form. Data covering specific radionuclides are summarized.

NUREG/CR-2907 V14: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. TICHLER, J.; DOTY, K.; LUCADAMO, K. Brookhaven National Laboratory. December 1995. 351pp. 9601290279. BNL-NUREG-51581. 86880:001.

Releases of radioactive materials in airborne and liquid effluents from commercial light water reactors during 1993 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 1993 release data are summarized in tabular form. Data covering specific radionuclides are summarized.

NUREG/CR-3243: COMPARISON OF ASME CODE FATIGUE EVALUATION METHODS FOR NUCLEAR CLASS 1 PIPING WITH CLASS 2 OR 3 PIPING. RODABAUGH, E.C. E.C. Rodabaugh Associates, Inc. RODABAUGH, E.C. Oak Ridge National Laboratory. June 1983. 59pp. 9511160312. ORNL-SUB82222521. 86249:160.

The fatigue evaluation procedure used in the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, for Class 1 piping is different from the procedure used for Class 2 or 3 piping. The basis for each procedure is described and correlations between the two procedures are presented. Conditions under which either procedure, or both, may be unconservative are noted. Potential changes in the Class 2 or 3 piping procedure to explicitly cover all loadings are discussed. However, the report is intended to be informative and, while the contents of the report may guide future Code changes, specific recommendations are not given herein.

NUREG/CR-3469 V08: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA. SULLIVAN, S.G.; KHAN, T.A.; XIE, J.W. Brookhaven National Laboratory. May 1995. 78pp. 9506140072. BNL-NUREG-51708. 84297:202.

The ALARA Center at Brookhaven National Laboratory publishes a series of bibliographies of selected readings in radiation protection and ALARA in a continuing effort to collect and disseminate information on radiation dose reduction at nuclear power plants. This is volume 8 of the series. The abstracts in this bibliography were selected from proceedings of technical meetings and conference journals, research reports, and searches of the Energy Science and Technology database of the U.S. Department of Energy. The subject material of these abstracts relates to the many aspects of radiation protection and dose reduction, and ranges from use of robotics, to operational health physics, to water chemistry. Material on the design, planning, and management of nuclear power stations is included, as well as information on decommissioning and safe storage efforts. Volume 8 contains 232 abstracts, an author index, and a subject index. The author index is specific for this volume. The subject index is cumulative and lists all abstract numbers from volumes 1 to 8. The numbers in boldface indicate the abstracts in this volume; the numbers not in boldface represent abstracts in previous volumes.

NUREG/CR-4219 V10 N2: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM. Semiannual Progress Report For April-September 1993. PENNELL, W.E. Oak Ridge National Laboratory. May 1995. 159pp. 9506140082. ORNL/TM-9593. 84299:001.

The Heavy-Section Steel Technology (HSST) Program is conducted for the Nuclear Regulatory Commission by Oak Ridge National Laboratory (ORNL). The program focuses 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 characterizations and properties, (4) special technical assistance, (5) fracture analysis computer programs, (6) cleavage-crack initiation, (7) cladding evaluations, (8) pressurized-thermal-shock technology, (9) analysis methods validation, (10) fracture evaluation tests, (11) warm prestressing, and (12) biaxial loading effects on fracture toughness. The program tasks have been structured to emphasize 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 April - September 1993.

NUREG/CR-4219 V11 N1: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM. Semiannual Progress Report For October 1993 - March 1994. PENNELL, W.E. Oak Ridge National Laboratory. November 1995. 97pp. 9512260324. ORNL/TM-9593. 86627:001.

The Heavy-Section Steel Technology (HSST) Program is conducted for the U.S. Nuclear Regulatory Commission (NRC) by Oak Ridge National Laboratory (ORNL). The program focus is on the development and validation of technology for the assessment of fracture-prevention margins in commercial nuclear reactor pressure vessels. The HSST Program is organized in seven tasks: (1) program management, (2) constraint effects analytical development and validation, (3) evaluation of cladding effects, (4) ductile to cleavage fracture mode conversion, (5) fracture analysis methods development and applications, (6) material property data and test methods, and (7) integration of results into a state-of-the-art methodology. 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 with related research programs both in the United States and abroad. This report provides an overview of principal developments in each of the seven program tasks from October 1993 - March 1994.

NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING WELDS. Seventh Program Report March 1993 - December 1994. WILKOWSKI, G.M.; GHADIALI, N.; RUDLAND, D.; et al. Battelle Memorial Institute, Columbus Laboratories. April 1995. 74pp. 9505180445. BMI-2173. 83967:074.

This is the seventh progress report of the U.S. Nuclear Regulatory Commission's research program entitled "Short Cracks in Piping and Piping Welds." The program objective is to verify and improve fracture analyses for circumferentially cracked large-diameter nuclear piping with crack sizes typically used in leak-before-break (LBB) analyses and in-service flaw evaluations. All work in the eight technical tasks have been completed. Ten topical reports are scheduled to be published. Progress only during the reporting period, March 1993 - December 1994, not covered in the topical reports is presented in this report. Details about the following efforts are covered in this report: Improvements to the two computer programs NRCPIPE and NRCPIPES to assess the failure behavior of circumferential through-wall and surface-cracked pipe, respectively; Pipe material property database PIFRAC; Circumferentially cracked pipe database CIRCUMCK.WK1; An assessment of the proposed ASME Section III design stress rule changes on pipe flaw tolerance; and A pipe fracture experiment on a section of pipe removed from service degraded by microbiologically induced corrosion (MIC) which contained a girth weld crack. Progress in the other tasks is not repeated here as it has been covered in great detail in the topical reports.

NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, October 1993 - March 1994. CHUNG, H.M.; CHOPRA, O.K.; ERCK, R.A.; et al. Argonne National Laboratory. March 1995. 65pp. 9503280398. ANL-95/2. 83292:228.

This report summarizes work performed by Argonne National Laboratory (ANL) on fatigue and environmentally assisted cracking (EAC) in light water reactors during the six months from October 1993 to March 1994. Topics that have been investigated include fatigue of low-alloy steel used in piping, steam generators, and reactor pressure vessels; EAC of wrought and cast austenitic stainless steels (SSs); and radiation-induced segregation and irradiation-assisted stress corrosion cracking (IASCC) of Type 304 SS after accumulation of high fluence. Fatigue tests have been conducted on A302-Gr B low-alloy steel to verify whether the current predictions of modest decreases of fatigue life in simulated pressurized water reactor water are

valid for high-sulfur heats that show environmentally enhanced fatigue crack growth rates. Additional crack growth data were obtained on fracture-mechanics specimens of austenitic SSs to investigate threshold stress intensity factors for EAC in high-purity oxygenated water at 289 degrees C. Microchemical changes in high- and commercial-purity Type 304 SS specimens from control-blade absorber tubes and a control-blade sheath from operating boiling water reactors were studied by Auger electron spectroscopy and scanning electron microscopy to determine whether trace impurity elements may contribute to IASCC of solution-annealed materials.

NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, April-September 1994. CHOPRA, O.K.; CHUNG, H.M.; GAVENDA, D.J.; et al. Argonne National Laboratory. September 1995. 67pp. 9510030169. ANL-95/25. 85688:172.

This report summarizes work performed by Argonne National Laboratory (ANL) on fatigue and environmentally assisted cracking (EAC) in light water reactors from April to September 1994. Topics that have been investigated include (a) fatigue of carbon and low-alloy steel used in piping and reactor pressure vessels, (b) EAC of austenitic stainless steels (SSs) and Alloy 600, and (c) irradiation-assisted stress corrosion cracking (IASCC) of Type 304 SS. Fatigue tests have been conducted on A106-Gr B and A533-Gr B steels in oxygenated water to determine whether a slow strain rate applied during different portions of a tensile-loading cycle are equally effective in decreasing fatigue life. Crack growth data were obtained on fracture-mechanics specimens of SSs and Alloy 600 to investigate EAC in simulated boiling water reactor (BWR) and pressurized water reactor environments at 289 degrees C. The data were compared with predictions from crack growth correlations developed at ANL for SSs in water and from rates in air from Section XI of the ASME Code. Microchemical 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 to determine whether trace impurity elements may contribute to IASCC of these materials.

NUREG/CR-4674 V21: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1994 A STATUS REPORT. Main Report And Appendices A-H. BELLES, R.J.; CLETCHER, J.W.; COPINGER, D.A.; et al. Oak Ridge National Laboratory. December 1995. 270pp. 9601290218. ORNL/NOAC-232. 86881:001.

Nine operational events that affected ten commercial light-water reactors (LWRs) during 1994 that are considered to be precursors to potential severe core damage are described. All of these events had conditional probabilities of subsequent core damage greater than or equal to 1.0×10^{-6} . These events were identified by computer-screening the 1994 licensee event reports from commercial LWRs to identify those that could be potential 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 staff to ensure that 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 1984-1993 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 21 contains the main report and Appendices A-H; Volume 22 contains Appendix I.

NUREG/CR-4674 V22: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1994 A STATUS REPORT. Appendix I. BELLES, R.J.; CLETCHER, J.W.; COPINGER, D.A.; et al. Oak Ridge National Laboratory. December 1995. 531pp. 9601290235. ORNL/NOAC-232. 86876:001.
See NUREG/CR-4674, V21 abstract.

NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS. Progress Report Of 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). April 1995. 33pp. 9505030447. 83725:281.

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 (3) 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-5229 V07: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1994. Annual Report. MCCONNELL, J.W.; ROGERS, R.D.; JASTROW, J.D.; et al. Idaho National Engineering Laboratory. May 1995. 74pp. 9506140086. INEL-94/0278. 84298:232.

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. Compressive test results of 11-year-old cement and vinyl ester-styrene solidified waste forms are presented, which show effects of aging and self-irradiation. Results of the ninth 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-5462: AGING STUDY OF BOILING WATER REACTOR HIGH PRESSURE INJECTION SYSTEMS. CONLEY, D.A.; EDSON, J.L.; FINEMAN, C.F. Idaho National Engineering Laboratory. March 1995. 114pp. 9504120089. INEL-94/0090. 83474:051.

The purpose of high pressure injection systems is to maintain an adequate coolant level in reactor pressure vessels, so that the fuel cladding temperature does not exceed 1,200 degrees C (2,200 degrees F), and to permit plant shutdown during a variety of design basis loss-of-coolant accidents. This report presents the results of a study on aging performed for high pressure injection systems of boiling water reactor plants in the United States. The purpose of the study was to identify and evaluate the effects of aging and the effectiveness of testing and maintenance in detecting and mitigating aging degradation. Guidelines from the United States Nuclear Regulatory Commission's Nuclear Plant Aging Research Program were used in performing the aging study. Review and analysis of the failures reported in databases such as Nuclear Power Experience, Licensee Event Reports, and the Nuclear Plant Reliability Data System, along with plant-specific maintenance records databases, are included in this report to provide the information required to identify aging stressors, failure modes, and failure causes. Several probabilistic risk assessments were reviewed to identify risk-significant components in high pressure injection systems. Testing, maintenance, specific safety issues, and codes and standards are also discussed.

NUREG/CR-5535 V01: RELAP5/MOD3 CODE MANUAL. Code Structure, System Models, And Solution Methods. * Idaho National Engineering Laboratory. August 1995. 412pp. 9509150284. INEL-95/0174. 85495:001.

The RELAP5/MOD3 computer code has been developed for best-estimate simulation of light water reactor coolant systems during postulated accidents. The code models the coupled behavior of the reactor coolant system and the core for 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 simulating a variety of thermal-hydraulic systems. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater systems. RELAP5/MOD3 code documentation is divided into seven volumes: Volume I presents modeling theory and associated numerical solution schemes; Volume II details instructions for code application and input data preparation; Volume III presents the results of developmental assessment cases that demonstrate and verify the models used in the code; Volume IV discusses in detail RELAP5 models and correlations; Volume V presents guidelines that have evolved over the past several years through the use of the RELAP5 code; Volume VI discusses the numerical scheme used in RELAP5; and Volume VII presents a collection of independent assessment calculations.

NUREG/CR-5535 V02: RELAP5/MOD3 CODE MANUAL. User's Guide And Input Requirements. * Idaho National Engineering Laboratory. August 1995. 323pp. 9509150290. INEL-95/0174. 85496:049.

See NUREG/CR-5535,V01 abstract.

NUREG/CR-5535 V04: RELAP5/MOD3 CODE MANUAL. Models And Correlations. * Idaho National Engineering Laboratory. August 1995. 430pp. 9509150303. INEL-95/0174. 85498:001.

See NUREG/CR-5535,V01 abstract.

NUREG/CR-5535 V05 R1: RELAP5/MOD3 CODE MANUAL. User's Guideline. FLETCHER,C.D.; SCHULTZ,R.R. Idaho National Engineering Laboratory. August 1995. 293pp. 9509150307. INEL-95/0174. 85497:008.

See NUREG/CR-5535,V01 abstract.

NUREG/CR-5591 V03: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Progress Report For October 1991 - September 1992. CORWIN,W.R. Oak Ridge National Laboratory. February 1995. 60pp. 9503140442. ORNL/TM-11568. 83071:007.

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(Ic) curve shift in high-copper welds, (3) K(Ia) curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K(Ic) and K(Ia) 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 October 1991 to September 1992.

NUREG/CR-5591 V04 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Semiannual Progress Report For April-September 1993. CORWIN,W.R. Oak Ridge National Laboratory. March 1995. 48pp. 9504180477. ORNL/TM-11568. 83569:316.

The goal of the Heavy-Section Steel Irradiation Program is to provide a thorough, quantitative assessment of effects of neutron irradiation on 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 14 tasks: (1) program management, (2) fracture toughness (K(Ic)) curve shift in high-copper welds, (3) crack-arrest toughness (K(Ia)) curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K(Ic) and K(Ia) curve shifts in low upper-shelf welds, (6) annealing effects in low upper-shelf welds, (7) irradiation effects in a commercial low upper-shelf weld, (8) microstructural analysis of irradiation effects, (9) in-service aged material evaluations, (10) correlation monitor materials, (11) special technical assistance, (12) JPDR steel examination, (13) technical assistance for JCCCNRS Working Groups 3 and 12, and (14) additional requirements for materials. This report provides an overview of the activities within each of these tasks from April to September 1993.

NUREG/CR-5591 V05 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Semiannual Progress Report For September 1993 Through March 1994. CORWIN,W.R. Oak Ridge National Laboratory. April 1995. 71pp. 9505190052. ORNL/TM-11568. 83981:267.

The goal of the Heavy-Section Steel Irradiation Program is to provide a thorough, quantitative assessment of effects of neutron irradiation on 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 14 tasks: (1) program management, (2) fracture toughness (K(Ic)) curve shift in high-copper welds, (3) crack-arrest toughness (K(Ia)) curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K(Ic) and K(Ia) curve shifts in low upper-shelf welds, (6) annealing effects in low upper-shelf welds, (7) irradiation effects in a commercial low upper-shelf weld, (8) microstructural analysis of irradiation effects, (9) in-service aged material evaluations, (10) correlation monitor materials, (11) special technical assistance, (12) JPDR steel examination, (13) technical assistance for JCCCNRS Working Groups 3 and 12, and (14) additional requirements for materials. This report provides an overview of the activities within each of these tasks from September 1993 through March 1994.

NUREG/CR-5591 V05 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Progress Report For April 1994 Through September 1994. CORWIN, W.R. Oak Ridge National Laboratory. July 1995. 76pp. 9508160263. ORNL/TM-11568. 85042:092.

The goal of the Heavy-Section Steel Irradiation Program is to provide a thorough, quantitative assessment of effects of neutron irradiation on 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 14 tasks: (1) program management, (2) fracture toughness (K(Ic)) curve shift in high-copper welds, (3) crack-arrest toughness (K(Ia)) curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K(Ic) and K(Ia) curve shifts in low upper-shelf welds, (6) annealing effects in low upper-shelf welds, (7) irradiation effects in a commercial low upper-shelf weld, (8) microstructural analysis of irradiation effects, (9) in-service aged material evaluations, (10) correlation monitor materials, (11) special technical assistance, (12) JPDR steel examination, (13) technical assistance for JCCCNRS Working Groups 3 and 12, and (14) additional requirements for materials. This report provides an overview of the activities within each of these tasks from April 1994 Through September 1994.

NUREG/CR-5591 V06 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Semiannual Progress Report For October 1994 Through March 1995. CORWIN, W.R. Oak Ridge National Laboratory. October 1995. 82pp. 9511270455. ORNL/TM-11568. 86317:001.

The goal of the Heavy-Section Steel Irradiation Program is to provide a thorough, quantitative assessment of effects of neutron irradiation on 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 14 tasks: (1) program management, (2) fracture toughness (K(Ic)) curve shift in high-copper welds, (3) crack-arrest toughness (K(Ia)) curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K(Ic) and K(Ia) curve shifts in low upper-shelf welds, (6) annealing effects in low upper-shelf welds, (7) irradiation effects in a commercial low upper-shelf weld, (8) microstructural analysis of irradiation effects, (9) in-service aged material evaluations, (10) correlation monitor materials, (11) special technical assistance, (12) JPDR steel examination, (13) technical assistance for JCCCNRS Working Groups 3 and 12, and (14) additional requirements for materials. This report provides an overview of the activities within each of these tasks from October 1994 Through March 1995.

NUREG/CR-5657: AUTOCASK (AUTOMATIC GENERATION OF 3-D CASK MODELS). A Microcomputer Based System For Shipping Cask Design Review Analysis. GERHARD, M.A.; SOMMER, S.C. Lawrence Livermore National Laboratory. April 1995. 130pp. 9505180397. UCRL-ID-118766. 83967:231.

AUTOCASK (AUTOMATIC Generation of 3-D CASK Models) is a microcomputer-based system of computer programs and databases developed at the Lawrence Livermore National Laboratory for the structural analysis of shipping casks for radioactive material. Model specification is performed on the microcomputer, and the analyses are performed on the engineering workstation or mainframe computer. AUTOCASK is based on 80386-60486 compatible microcomputers. The system is composed of a series of menus, input programs, display programs, a mesh generation program, and archive programs. All data is entered through fill-in-the-blank input screens that contain descriptive data requests.

NUREG/CR-5758 V05: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY. Annual Summary Of Program Performance Reports CY 1994. WESTRA, C.; DURBIN, N.; FIELD, I.; et al. Battelle Seattle Research Center. August 1995. 83pp. 9508230332. PNL-10638. 85118:134.

This report summarizes data from semiannual reports on fitness-for-duty program performance submitted to the NRC by utilities for two reporting periods: Jan. 1 through June 30, 1994, and July 1 through Dec. 31, 1994. In 1994, licensees reported that they had conducted 163,247 tests for the presence of illegal drugs and alcohol. Of these tests, 1,372 (.84%) were confirmed positive. Positive test results varied by category of test and category of worker. The majority of positive test results (977) were obtained through pre-access testing. Of tests conducted on workers having access to the protected area, 233 were positive from random testing and 122 were positive from for-cause testing. Followup testing of workers who had previously tested positive resulted in 50 positive tests. For-cause testing resulted in the highest percentage of positive tests; about 16% of for-cause tests were positive. In comparison, 1.22% of pre-access tests and .28% of random tests were positive. Positive test rates also varied by category of worker. When all types of tests are combined (pre-access, random, for-cause, and followup), short-term contractor personnel had the highest positive test rate at 1.22%. Licensee employees and long-term contractors had lower combined positive test rates (.33% and .49% respectively). Of the substances tested, marijuana was responsible for the highest percentage of positive test results (52.79%), followed by cocaine (24.57%) and alcohol (17.93%).

NUREG/CR-5857: AGING OF TURBINE DRIVES FOR SAFETY-RELATED PUMPS IN NUCLEAR POWER PLANTS. COX, D.F. Oak Ridge National Laboratory. June 1995. 64pp. 9507120251. ORNL-6713. 84634:295.

This study examines the relationship between aging, and current industry practices in areas of maintenance, surveillance, and operation of steam turbine drives for safety related pumps. These pumps are located in the Auxiliary Feedwater (AFW) system for pressurized water reactor plants, and the Reactor Core Isolation Cooling and High Pressure Coolant Injection systems for Boiling Water Reactor facilities. Findings in a recent study on the AFW (NUREG/CR-5404) indicate that the turbine drive is the single largest contributor to AFW system degradation. However, examination of the data show that the turbine itself is a reliable piece of equipment with a good service record. Most of the problems documented are the result of problems with the turbine controls and the mechanical overspeed trip mechanism, which apparently stem from three major causes.

NUREG/CR-5884 V01: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And.... KONZEK, G.J.; SMITH, R.I.; BIRSCHBACH, M.C.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1995. 122pp. 9512260293. PNL-8742. 86610:226.

With the issuance of the final Decommissioning Rule (July 27, 1988), owners and operators of licensed nuclear power plants are required to prepare, and submit to the U.S. Nuclear Regulatory Commission (NRC) for review, decommissioning plans and cost estimates. The NRC staff is in need of bases documentation that will assist them in assessing the adequacy of the licensee submittals, from the viewpoint of both the planned actions, including occupational radiation exposure, and the probable costs. The purpose of this reevaluation study is to provide some of the needed bases documentation. This report contains the results of a review and reevaluation of the 1978 PNL decommissioning study of the Trojan nuclear power plant (NUREG/CR-0130), including all identifiable factors and cost assumptions which contribute significantly to the total cost of

decommissioning the nuclear power plant for the DECON, SAF-STOR, and ENTOMB decommissioning alternatives. These alternatives now include an initial 5-7 year period during which time the spent fuel is stored in the spent fuel pool, prior to beginning major disassembly or extended safe storage of the plant. Included for information (but not presently part of the license termination cost) is an estimate of the cost to demolish the decontaminated and clean structures on the site and to restore the site to a "green field" condition. This report also includes consideration of the NRC requirement that decontamination and 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 materials whose radionuclide concentrations exceed the limits for Class C low-level waste (i.e., Greater-Than-Class C), and reflects 1993 costs for labor, materials, transport, and disposal activities. Sensitivity of the total license termination cost to the disposal costs at different low-level radioactive waste disposal sites, and to different depths of contaminated concrete surface removal within the facilities is also examined.

NUREG/CR-5884 V02: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And.... KONZEK, G.J.; SMITH, R.I.; BIERSEBACH, M.C.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1995. 424pp. 9512260336. PNL-8742. 86609:001.

See NUREG/CR-5884, V01 abstract.

NUREG/CR-5927 V02: EVALUATION OF A PERFORMANCE ASSESSMENT METHODOLOGY FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES. Validation Needs. KOZAK, M.W.; OLAGUE, N.E. Sandia National Laboratories. February 1995. 43pp. 9503170314. SAND91-2801. 83151:304.

In this report, concepts on how validation fits into the scheme of developing confidence in performance assessments are introduced. A general framework for validation and confidence building in regulatory decision making is provided. It is found that traditional validation studies have a very limited role in developing site-specific confidence in performance assessments. Indeed, validation studies are shown to have a role only in the context that their results can narrow the scope of initial investigations that should be considered in a performance assessment. In addition, validation needs for performance assessment of low-level waste disposal facilities are discussed, and potential approaches to address those needs are suggested. These areas of topical research are ranked in order of importance based on relevance to a performance assessment and likelihood of success.

NUREG/CR-5944 V02: A CHARACTERIZATION OF CHECK VALVE DEGRADATION AND FAILURE EXPERIENCE IN THE NUCLEAR POWER INDUSTRY. 1991 Failures. MCELHANEY, K.L. Oak Ridge National Laboratory. July 1995. 87pp. 9508090036. ORNL-6734. 84963:003.

Review and characterization of check valve failures taken from the Institute of Nuclear Power Operation's Nuclear Plant Reliability Data System was performed in accordance with part two of a three-phase project sponsored by the U.S. Nuclear Regulatory Commission. The analysis was performed for check valve failures occurring in 1991 and is intended to update the previous analysis performed for the time period 1984-1990. To maintain consistency and for ease of trending, the 1991 analysis presents the same parameters and cross-correlations in essentially the same format as the 1984-1990 study. Additional data was obtained for the 1991 analysis, including information related to specific check valve type. This information is presented in a separate section of the report.

NUREG/CR-5954: EFFECT ON AGING ON PWR CHEMICAL AND VOLUME CONTROL SYSTEM. GROVE, E.J.; TRAVIS, R.J. Brookhaven National Laboratory. June 1995. 196pp. 9507060335. BNL-NUREG-52410. 84532:181.

The effect of aging on the PWR Chemical and Volume Control System (CVCS) has been evaluated. A detailed review of the NPRDS and LER databases for the 1988-1991 time period, together with a review of industry and NRC experience and research, indicate that age-related degradations and failures have occurred. These failures had significant effects on plant operation, including reactivity excursions, and pressurizer level transients. The majority of these component failures resulted in leakage of reactor coolant outside the containment. A representative plant of each PWR NSSS design (W, CE, and B&S) was visited to obtain specific information on system inspection, surveillance, monitoring, and inspection practices. The results of these visits indicate that adequate system maintenance and inspection is being performed. In some instances, the frequencies of inspection were increased in response to repeated failure events. A parametric study was performed to assess the effect of system aging on Core Damage Frequency (CDF). This study showed that as MOV operating failures increased the contribution of the High Pressure Injection to CDF also increased.

NUREG/CR-5973 R02: CODES AND STANDARDS AND OTHER GUIDANCE CITED IN REGULATORY DOCUMENTS. NICKOLAUS, J.R.; BOHLANDER, K.L. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1995. 525pp. 9509130146. PNL-8462. 85429:001.

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 2 adds more than 500 citations. This listing identifies the version of the code or standard cited in the regulatory document, the regulatory document, and the latest 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, Enforcement Manual, Generic Letters, Inspection Manual, Policy Statements, Regulatory Guides, Standard Technical Specifications, and the Standard Review Plan (NUREG-0800).

NUREG/CR-5975 R01: INCENTIVE REGULATION OF INVESTOR-OWNED NUCLEAR POWER PLANTS BY PUBLIC UTILITY REGULATORS. MCKINNEY, M.D.; SEELY, H.E.; MERRITT, C.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1995. 70pp. 9505180354. PNL-8466. 83967:001.

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

NUREG/CR-6002: RISK-BASED MAINTENANCE

MODELING. Prioritization Of Maintenance Importances And Quantification Of Maintenance Effectiveness. VESELY, W.E.; REZOS, J.T. Science Applications International Corp. (formerly Science Applications, Inc.). * Brookhaven National Laboratory. September 1995. 90pp. 9510030191. BNL-NUREG-52332. 85672:266.

This report describes approaches for prioritizing the risk importances of maintenances using a Probabilistic Risk Assessment (PRA). Approaches are then described for quantifying the reliability and risk effects of maintenance actions. Two different PRA importance measures, minimal cutset importances and risk reduction importances, are used to prioritize maintenances and the report shows that similar results are obtained if appropriate criteria are used. The justifications for the particular importance measures are also developed. The approaches which are developed for quantifying the reliability and risk effects of maintenance actions are extensions of the usual reliability models now used in PRAs. These extended models consider degraded states of the component and quantify the benefits of maintenance in correcting degradations and preventing failures. The negative effects of maintenance, including maintenance downtimes, are also included. These models are specific types of Markov models. This report analyzes a range of postulated values of input data in order to explore the potential usefulness of these models. The effects of maintenance are quantified to be large in specific cases.

NUREG/CR-6004: PROBABILISTIC PIPE FRACTURE EVALUATIONS FOR LEAK-RATE-DETECTION APPLICATIONS. RAHMAN, S.; GHADIALI, N.; PAUL, D.; et al. Battelle Memorial Institute, Columbus Laboratories. April 1995. 336pp. 9505090276. BML-2174. 83848:001.

Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," was published by the U.S. Nuclear Regulatory Commission (NRC) in May 1973, and provides guidance on leak detection methods and system requirements for Light Water Reactors. Additionally, leak detection limits are specified in plant Technical Specifications and are different for Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs). These leak detection limits are also used in leak-before-break evaluations performed in accordance with Draft Standard Review Plan, Section 3.6.3, "Leak Before Break Evaluation Procedures" where a margin of 10 on the leak detection limit is used in determining the crack size considered in subsequent fracture analyses. This study was requested by the NRC to: (1) evaluate the conditional failure probability for BWR and PWR piping for pipes that were leaking at the allowable leak detection limit, and (2) evaluate the margin of 10 to determine if it was unnecessarily large. A probabilistic approach was undertaken to conduct fracture evaluations of circumferentially cracked pipes for leak-rate-detection applications. Sixteen nuclear piping systems in BWR and PWR plants were analyzed to evaluate conditional failure probability and effects of crack-morphology variability on the current margins used in leak rate detection for leak-before-break.

NUREG/CR-6016: AGING AND SERVICE WEAR OF AIR-OPERATED VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS. COX, D.F.; MCELHANEY, K.L.; STAUNTON, R.H. Oak Ridge National Laboratory. May 1995. 62pp. 9506140092. ORNL/TM-6789. 84299:152.

Air-operated valves (AOVs) are used in a variety of safety-related applications at nuclear power plants. They are often used where rapid stroke times are required or precise control of the valve obturator is required. They can be designed to operate automatically upon loss of power, which is often desirable when selecting components for response to design basis conditions. The purpose of this report is to examine the reported failures of AOVs and determine whether there are identifiable trends in the failures related to predictable causes. This report examines the specific components that comprise a typical AOV, how those components fail, when they fail, and how such failures are dis-

covered. It also examines whether current testing frequencies and methods are effective in predicting such failures.

NUREG/CR-6017: FIRE MODELING OF THE HEISS DAMPF REAKTOR CONTAINMENT. NICOLETTE, V.F. Sandia National Laboratories. YANG, K.T. Notre Dame, Univ. of, Notre Dame, IN. September 1995. 115pp. 9510310360. SAND93-0528. 86018:001.

This report summarizes Sandia National Laboratories' participation in the fire modeling activities for the German Heiss Dampf Reaktor (HDR) containment building, under the sponsorship of the United States Nuclear Regulatory Commission. The purpose of this report is twofold: 1) to summarize Sandia's participation in the HDR fire modeling efforts and 2) to summarize the results of the international fire modeling community involved in modeling the HDR fire tests. Additional comments on the state of fire modeling and trends in the international fire modeling community are also included. It is noted that, although the trend internationally in fire modeling is toward the development of the more complex fire field models, each type of fire model has something to contribute to the understanding of fires in nuclear power plants.

NUREG/CR-6046: ALERTNESS, PERFORMANCE, AND OFF-DUTY SLEEP ON 8-HOUR AND 12-HOUR NIGHT SHIFTS IN A SIMULATED CONTINUOUS OPERATIONS CONTROL ROOM SETTING. BAKER, T.L. Institute for Circadian Physiology. April 1995. 157pp. 9505030466. 83745:228.

This study of simulated 8-hour and 12-hour work shifts compares alertness, speed, and accuracy of subjects assigned to the two shift conditions. Twenty male subjects, ages 20 to 45, were randomly assigned to either an 8-hour or 12-hour shift protocol, working in teams of two in a process control simulator. In addition to the simulator monitoring task, subjects completed a 15-minute performance assessment battery and a vigor and affect scale hourly. Throughout the simulated work shift, alertness was monitored by recording electroencephalogram (EEG), electrooculogram (EOG) and electromyogram (EMG) data. At 2-hour intervals, subjects' sleepiness-alertness level was assessed via standardized nap tests. During off-shift hours, subjects lived in attached, self-contained apartments. Subjects were restricted to bed for an 8-hour sleep period in total darkness one-hour after work shifts; EEG, EOG, AND EMG recordings were continued during sleep. During days off between blocks of shifts, subjects left the facility, but kept detailed logs of sleep-wake activities. No significant differences were found between subjects on 8-hour and 12-hour night shifts in performance on the simulator task, in length or quality of daytime sleep, or in physiological or subjective sleepiness-alertness on shift. Except for one subtest in the performance test battery, subjects on 12-hour shifts were slower, but more accurate, than those on 8-hour shifts. Several measures showed better alertness, mood, and off-duty sleep on evening shifts than on night shifts.

NUREG/CR-6054: ESTIMATING PRESSURIZED WATER REACTOR DECOMMISSIONING COSTS. A User's Manual For The PWR Cost Estimating Computer Program (CECP) Software. BIRSCHBACH, M.C. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1995. 174pp. 9512260287. PNL-8497. 86610:061.

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 user's manual and the accompanying Cost Estimating Computer Program (CECP) software provide a cost-calculating methodology to the NRC staff that will assist them in assessing the adequacy of the licensee submittals. The CECP, designed to be used on a personal computer, provides estimates for the cost of decommissioning PWR power stations to the point of license termination. Such cost estimates include component, piping, and equipment 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-6074 V04: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING. Technical Report On The Findings Of Task 4. Investigation Of Sealed Source for Paper Mill Digester. BENAC, D.J.; IDINGS, F.A. Southwest Research Institute. October 1995. 42pp. 9512040331. 04-4448-013. 86386:086.

This report covers the Task 4 activities for the Sealed Source and Device Safety testing program. SwRI was contracted to investigate a suspected leaking radioactive source that was installed in a gauge that was on a paper mill digester. The actual source that was leaking was not available, therefore, SwRI examined another source. SwRI concluded that the encapsulated source examined by SwRI was not leaking. However, the presence of Cs-137 on the interior and exterior of the outer encapsulation and handling tube suggests that contamination probably occurred when the source was first manufactured, then installed in the handling tube.

NUREG/CR-6074 V05: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING. Technical Report On The Findings Of Task 4. Investigation Of Failed Radioactive Stainless Steel Troxler Gauges. BENAC, D.J.; SCHICK, W.R. Southwest Research Institute. October 1995. 58pp. 9512040326. 04-4448-010. 86386:129.

This report covers the Task 4 activities for the Sealed Source and Device Safety testing program. SwRI was contracted to investigate failed radioactive stainless steel troxler gauges. SwRI's task was to determine the cause of failure of the rods and the extent of the problem. SwRI concluded that the broken rod failed in a brittle manner due to a hard zone in the heat affected zone.

NUREG/CR-6089: DETECTION OF PUMP DEGRADATION. GREENE, R.H.; CASADA, D.A. Oak Ridge National Laboratory. August 1995. 108pp. 9509130143. ORNL-6765. 85430:182.

This study examines the methods of detecting pump degradation that are employed in domestic and overseas nuclear facilities. This report evaluates criteria mandated by required pump testing at U.S. nuclear plants and compares them to state-of-the-art diagnostic programs and practices implemented by other major industries. Since the working condition of the pump driver is critical to pump operability, a review of new applications of motor diagnostics is also provided that highlights developments in this technology. Vibration spectral analysis is a powerful diagnostic tool for the pump analyst. The routine collection and analysis of spectral data is superior to other technologies in its ability to accurately detect numerous types and causes of pump degradation. Existing ASME testing criteria do not require the evaluation of pump vibration spectra but instead overall vibration amplitude. The mechanical information discernible from vibration amplitude is limited, and several pump failures in the nuclear power industry were not detected by vibration amplitude monitoring. Since spectral analysis provides pertinent information concerning the mechanical condition of rotating machinery, its incorporation into ASME testing criteria may merit a relaxation in the monthly-to-quarterly testing schedules that seek to verify pump operability.

NUREG/CR-6100: GATE VALVE AND MOTOR-OPERATOR RESEARCH FINDINGS. STEELE, R.; DEWALL, K.G.; WATKINS, J.C.; et al. Idaho National Engineering Laboratory. September 1995. 130pp. 9510060250. INEL-94/0156. 85745:001.

This report provides an update on the valve research being sponsored by the U.S. Nuclear Regulatory Commission (NRC) and conducted at the Idaho National Engineering Laboratory (INEL). The research addresses the need to provide assurance that motor-operated valves can perform their intended safety function, usually to open or close against specified (design basis) flow and pressure loads. This report describes several

important developments: (a) two methods for estimating or bounding the design basis stem factor (in rising-stem valves), using data from tests less severe than design basis tests; (b) a new correlation for evaluating the opening responses of gate valves and for predicting opening requirements; (c) an extrapolation method that uses the results of a best effort flow test to estimate the design basis closing requirements of a gate valve that exhibits atypical responses (peak force occurs before flow isolation); and (d) the extension of the original INEL closing correlation to include low-flow and low-pressure loads. The report also includes a general approach, presented in step-by-step format, for determining operating margins for rising-stem valves (gate valves and globe valves) as well as quarter-turn valves (ball valves and butterfly valves).

NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY. PILCH, M.M.; ALLEN, M.D. Sandia National Laboratories. SPENCER, B.W.; et al. Argonne National Laboratory. May 1995. 275pp. 9506010485. SAND93-2078. 84121:302.

This report uses the methodology and scenarios described in NUREG/CR-6075 and its supplement to address the direct containment heating (DCH) issue for the Surry nuclear power plant (NPP). Consistency of the initial distributions has been ensured by using insights from system-level codes, specifically SCDAP/RELAP5 and CONTAIN. The most useful insights are that the reactor coolant system (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/RELAP5 output was used as input to CONTAIN to assess the containment conditions at vessel breach. The load evaluations for Surry showed no intersections of the load distributions with the containment strength distribution, and thus the DCH issue for Surry can be resolved based on containment loads alone. However, the likelihood of high RCS pressures at vessel breach was evaluated for Surry for a limited number of sequences. The probability of RCS pressures greater than 1.38 MPa for all station blackout scenarios without power recovery or operator intervention was found to be low (0.077). This probability could have been factored into the containment failure probability for Surry if there had been intersections of the load and strength distributions.

NUREG/CR-6112: IMPACT OF REDUCED DOSE LIMITS ON NRC LICENSED ACTIVITIES. Major Issues In The Implementation Of ICRP/NCRP Dose Limit Recommendations. Final Report. MEINHOLD, C.B. Brookhaven National Laboratory. May 1995. 79pp. 9505230093. BNL/NUREG-52394. 84017:240.

This report summarizes information required to estimate, at least qualitatively, the potential impacts of reducing occupational dose limits below those given in 10 CFR 20 (Revised). The following overall conclusions were reached: (1) Although 10 mSv yr⁻¹ is a reasonable limit for many licensees, such a limit could be extraordinarily difficult to achieve and potentially destructive to the continued operation of some licensees, such as nuclear power, fuel fabrication, and medicine, (2) Twenty mSv yr⁻¹ as a limit is possible for some of these groups, but for others it would prove difficult, (3) Fifty mSv yr⁻¹ and age in 10s of mSv appear reasonable for all licensees, both in terms of the lifetime risk of cancer and severe genetic effects to the most highly exposed workers, and the practicality of operation. In some segments of the industry, this acceptability is based on the adoption of a "grandfather clause" for those people exceeding or close to exceeding the cumulative limit at this time.

NUREG/CR-6114 V02: AUXILIARY ANALYSES IN SUPPORT OF PERFORMANCE ASSESSMENT OF A HYPOTHETICAL LOW-LEVEL WASTE FACILITY. Two-Phase Flow And Contaminant Transport In Unsaturated Soils With Application To Low-Level Radioactive Waste Disposal. BINNING, P.; CELIA, M.A.; JOHNSON, J.C. Princeton Univ., Princeton, NJ. May 1995. 139pp. 9506140130. 84299:214.

A numerical model of multiphase air-water flow and contaminant transport in the unsaturated zone is presented. The multiphase flow equations are solved using the two-pressure, mixed form of the equations with a modified Picard linearization of the equations and a finite element spatial approximation. A volatile contaminant is assumed to be transported in either phase, or in both phases simultaneously. The contaminant partitions between phases with an equilibrium distribution given by Henry's Law or via kinetic mass transfer. The transport equations are solved using a Galerkin finite element method with reduced integration to lump the resultant matrices. The numerical model is applied to published experimental studies to examine the behavior of the air phase and associated contaminant movement under water infiltration. The model is also used to evaluate a hypothetical design for a low-level radioactive waste disposal facility. The model has been developed in both one and two dimensions; documentation and computer codes are available for the one-dimensional flow and transport model.

NUREG/CR-6116 V06: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Graphical Evaluation Module (GEM) Reference Manual. RUSSELL, K.D.; KVARFORDT, K.J.; HOFFMAN, C.L.; et al. Idaho National Engineering Laboratory. October 1995. 113pp. 9511270459. INEL-94/0039. 86344:149.

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 Graphical Evaluation Module (GEM) is a special application tool designed for evaluation of operational occurrences using the Accident Sequence Precursor (ASP) program methods. GEM provides the capability for an analyst to quickly and easily perform conditional core damage probability (CCDP) calculations. The analyst can then use the CCDP calculations to determine if the occurrence of an initiating event or a condition adversely impacts safety. It uses models and data developed in the SAPHIRE specially for the ASP program. GEM requires more data than that normally provided in SAPHIRE and will not perform properly with other models or data bases. This is the first release of GEM and the developers of GEM welcome user comments and feedback that will generate ideas for improvements to future versions. GEM is designated as version 5.0 to track GEM codes along with the other SAPHIRE codes as the GEM relies on the same, shared database structure.

NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Verification And Validation (V&V) Manual. JONES, J.L.; CALLEY, M.B.; CAPPS, E.L.; et al. Idaho National Engineering Laboratory. March 1995. 141pp. 9503270310. INEL-94/0039. 83270:133.

A verification and validation (V&V) process has been performed for the System Analysis Programs for Hands-on Integrated Reliability Evaluation (SAPHIRE) Version 5.0. SAPHIRE is a set of four computer programs that the Nuclear Regulatory Commission has developed for the performance of probabilistic risk assessments. These programs allow an analyst to perform many of the functions necessary 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 (MAR-D), and Fault Tree/Event Tree/Piping and Instrumentation Diagram (FEP) graphical editor. The intent of this program is V&V of successive versions of SAPHIRE. The SAPHIRE 5.0 V&V plan is based on the SAPHIRE 4.0 V&V plan with revisions to incorporate lessons learned from the previous effort. The SAPHIRE 5.0 vital and nonvital test procedures are based on the test procedures from SAPHIRE 4.0 with revisions to include the new SAPHIRE 5.0. The majority of the results from the testing was acceptable; however, some discrepancies between expected code

operation and actual code operation were identified. Modifications that have been made to SAPHIRE are identified.

NUREG/CR-6116 V10: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Data Loading Manual. VANHORN, R.L.; WOLFRAM, L.M.; FOWLER, R.D.; et al. Idaho National Engineering Laboratory. April 1995. 161pp. 9505180347. INEL-94/0039. 83966:120.

The Systems Analysis Programs for the Hands-on Integrated Reliability Evaluations (SAPHIRE) suite of programs can be used to organize and standardize in an electronic format information from probabilistic risk assessments or individual plant examinations. The Models and Results Database (MAR-D) program of the SAPHIRE suite serves as the repository for probabilistic risk assessment and individual plant examination data and information. This report demonstrates by examples the common electronic and manual methods used to load these types of data. It is not a stand alone document but references documents that contribute information relative to the data loading process. This document provides a more detailed discussion and instructions for using SAPHIRE 5.0 only when enough information on a specific topic is not provided by another available source.

NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS. Primer And User's Guides. Version 1.8.3 September 1994. SUMMERS, R.M.; COLE, R.K.; SMITH, R.C.; et al. Sandia National Laboratories. March 1995. 708pp. 9505040396. SAND93-2185. 83779:001.

MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants. MELCOR is being developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission as a second-generation plant risk assessment tool and the successor to the Source Term Code Package. A broad spectrum of severe accident phenomena in both boiling and pressurized water reactors is treated in MELCOR in a unified framework. These include: thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heatup, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport; and the impact of engineered safety features on thermal-hydraulic and radionuclide behavior. Current uses of MELCOR include estimation of severe accident source terms and their sensitivities and uncertainties in a variety of applications.

NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS. Reference Manuals. Version 1.8.3 September 1994. SUMMERS, R.M.; COLE, R.K.; SMITH, R.C. Sandia National Laboratories. March 1995. 885pp. 9505040402. SAND93-2185. 83781:001.

See NUREG/CR-6119, V01 abstract.

NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance. CALLAN, J.R.; KELLY, R.T.; QUINN, M.L.; et al. Pacific Science & Engineering Group, Inc. May 1995. 205pp. 9506140119. 84300:001.

Remote Afterloading Brachytherapy (RAB) is a medical process used in the treatment of cancer. RAB uses a computer-controlled device to remotely insert and remove radioactive sources close to a target (or tumor) in the body. Some RAB problems affecting the radiation dose to the patient have been reported and attributed to human error. To determine the root cause of human error in the RAB system, a human factors team visited 23 RAB treatment sites in the U.S. The team observed RAB treatment planning and delivery, interviewed RAB personnel, and performed walk-throughs, during which staff demonstrated the procedures and practices used in performing RAB tasks. Factors leading to human error in the RAB system were identified.

fied. The impact of those factors on the performance of RAB was then evaluated and prioritized in terms of safety significance. Finally, the project identified and evaluated alternative approaches for resolving the safety significant problems related to human error.

NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function And Task Analysis. CALLAN, J.R.; GWYNNE, J.W.; KELLY, R.T.; et al. Pacific Science & Engineering Group, Inc. May 1995. 143pp. 9506140126. 84300:207.

A human factors project on the use of nuclear by-product material to treat cancer using remotely operated afterloaders was undertaken by the Nuclear Regulatory Commission. The purpose of the project was to identify factors that contribute to human error in the system for remote afterloading brachytherapy (RAB). This report documents the findings from the first phase of the project, which involved an extensive function and task analysis of RAB. This analysis identified the functions and tasks in RAB, made preliminary estimates of the likelihood of human error in each task, and determined the skills needed to perform each RAB task. The findings of the function and task analysis served as the foundation for the remainder of the project, which evaluated four major aspects of the RAB system linked to human error: human-system interfaces; procedures and practices; training and qualifications of RAB staff; and organizational practices and policies. At its completion, the project identified and prioritized areas for recommended NRC and industry attention based on all of the evaluations and analyses.

NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Supporting Analyses Of Human-System Interfaces, Procedures And Practices, Training And Organizational Practices And Procedures. CALLAN, J.R.; KELLY, R.T.; QUINN, M.L.; et al. Pacific Science & Engineering Group, Inc. July 1995. 226pp. 9508090067. 84965:001.

A human factors project on the use of nuclear by-product material to treat cancer using remotely operated afterloaders was undertaken by the Nuclear Regulatory Commission. The purpose of the project was to identify factors that contribute to human error in the system for remote afterloading brachytherapy (RAB). This report documents the findings from the second, third, fourth, and fifth phases of the project, which involved detailed analyses of four major aspects of the RAB system linked to human error: human-system interfaces; procedures and practices; training practices and policies; and organizational practices and policies, respectively. Findings based on these analyses provided factual and conceptual support for the final phase of this project, which identified factors leading to human error in RAB. The impact of those factors on RAB performance was then evaluated and prioritized in terms of safety significance, and alternative approaches for resolving safety significant problems were identified and evaluated.

NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL. HELTON, J.C. Arizona State Univ., Tempe, AZ. JOHNSON, J.D.; ROLLSTIN, J.A.; et al. GRAM, Inc. January 1995. 96pp. 9502080054. SAND93-2370. 82644:222.

Uncertainty and sensitivity analysis techniques based on Latin hypercube sampling, partial correlation analysis and stepwise regression analysis are used in an investigation with the MACCS model of the chronic exposure pathways associated with a severe accident at a nuclear power station. The primary purpose of this study is to provide guidance on the variables to be considered in future review work to reduce the uncertainty in the important variables used in the calculation of reactor accident consequences. The effects of 75 imprecisely known input variables on the following reactor accident consequences are studied: crop growing season dose, crop long-term dose, water ingestion dose, milk growing season dose, long-term ground-

shine dose, long-term inhalation dose, total food pathways dose, total ingestion pathways dose, total long-term pathways dose, total latent cancer fatalities, area-dependent cost, crop disposal cost, milk disposal cost, population-dependent cost, total economic cost, condemnation area, condemnation population, crop disposal area and milk disposal area.

NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL. HELTON, J.C. Arizona State Univ., Tempe, AZ. JOHNSON, J.D. GRAM, Inc. MCKAY, M.D.; et al. Los Alamos National Laboratory. January 1995. 147pp. 9502080062. SAND93-2371. 82660:063.

Uncertainty and Sensitivity analysis techniques based on Latin hypercube sampling, partial correlation analysis and stepwise regression analysis are used in an investigation with the MACCS model of the early health effects associated with a severe accident at a nuclear power station. The primary purpose of this study is to provide guidance on the variables to be considered in future review work to reduce the uncertainty in the important variables used in the calculation of reactor accident consequences. The effects of 34 imprecisely known input variables on the following reactor accident consequences are studied: number of early fatalities, number of cases of prodromal vomiting, population dose within 10 mi of the reactor, population dose within 1000 mi of the reactor, individual early fatality probability within 1 mi of the reactor, and maximum early fatality distance.

NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL. HELTON, J.C. Arizona State Univ., Tempe, AZ. JOHNSON, J.D.; ROLLSTIN, J.A.; et al. GRAM, Inc. January 1995. 94pp. 9502080088. SAND93-2372. 82660:210.

Uncertainty and Sensitivity analysis techniques based on Latin hypercube sampling, partial correlation analysis and stepwise regression analysis are used in an investigation with the MACCS model of the food pathways associated with a severe accident at a nuclear power station. The primary purpose of this study is to provide guidance on the variables to be considered in future review work to reduce the uncertainty in the important variables used in the calculation of reactor accident consequences. The effects of 87 imprecisely known input variables on the following reactor accident consequences are studied: crop growing season dose, crop long-term dose, milk growing season dose, total food pathways dose, total ingestion pathways dose, total long-term pathways dose, area dependent cost, crop disposal cost, milk disposal cost, condemnation area, crop disposal area and milk disposal area.

NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS. SAMANTA, P.K.; KIM, I.S. Brookhaven National Laboratory. MANKAMO, T.; et al. Avaplan Oy (Finland). December 1994. 190pp. 9503010187. BNL-NUREG-52398. 82887:001.

Technical Specifications (TS) requirements for nuclear power plants define the Limiting Conditions for Operation and Surveillance Requirements to assure safety during operation. In general, these requirements are based on deterministic analysis and engineering judgments. Experiences with plant operation indicate that some elements of the requirements are unnecessarily restrictive, while a few may not be conducive to safety. Improving these requirements involves many considerations and is facilitated by the availability of plant-specific Probabilistic Safety Assessments and development of related methods for analyses. This handbook summarizes the risk- and reliability-based methods to improve TS requirements. The scope of the handbook includes reliability- and risk-based methods for evaluating allowed outage times, scheduled or preventive maintenances, action statements requiring shutdown where shutdown risk may be substantial, surveillance test intervals, and management of

plant configurations resulting from outages of systems, or components. For each topic, the handbook summarizes analytic methods with data needs, outlines the insights to be gained, lists additional references, and gives examples of evaluations.

NUREG/CR-6143 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Summary Of Results. CHU, T.L.; PRATT, W.T. Brookhaven National Laboratory. October 1995. 118pp. 9511270460. BNL-NUREG-52399. 86317:223.

This document contains a summarization of the results and insights from the Level 1 accident sequence analyses of internally initiated events, internally initiated fire and flood events, seismically initiated events, and the Level 2/3 risk analysis of internally initiated events (excluding fire and flood) for Grand Gulf, Unit 1 as it operates in the Low Power and Shutdown Plant Operational State 5 during a refueling outage. The report summarizes the Level 1 information contained in Volumes 2 - 5 and the Level 2/3 information contained in Volume 6 of NUREG/CR-6143.

NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Main Report And Appendices. BROWN, T.D.; KMETYK, L.N.; WHITEHEAD, D.W.; et al. Sandia National Laboratories. March 1995. 407pp. 9504050320. SAND93-2440. 83378:027.

The analysis documented in this volume of the report is the Level 2/3 analysis of the traditional internal events. Plant damage states, which define the configuration of the plant and its systems at the onset of core damage for the accident scenarios developed in the Level 1 analysis, were used to define the interface between the Level 1 and Level 2/3 analyses. In the Level 2/3 analysis, the possible progressions of the accident following the onset of core damage were delineated and the amount of radioactive material released to the environment was estimated. Based on the amount of radioactive material released to the environment, health effects to the general public were estimated. In addition to the offsite consequences, a scoping analysis of the potential doses and dose rates within the site were also estimated. The final product of the analysis was the integration of the accident frequencies with the consequences of the accidents to form an expression for aggregate risk.

NUREG/CR-6143 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Supporting MELCOR Calculations. KMETYK, L.N.; BROWN, T.D. Sandia National Laboratories. March 1995. 431pp. 9504100148. SAND93-2440. 83418:001.

The document contains the deterministic code calculations performed with the MELCOR Code that were used to support the development and quantification of the PRA models used in the 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 background for the work documented in this report is summarized including how deterministic codes are used in PRAs, why the MELCOR code is used, what the capabilities and features of MELCOR are and how the code has been used by others in the past. Brief descriptions of the Grand Gulf plant and its configuration during LP&S operation and of the MELCOR input model developed for the Grand Gulf plant in its LP&S configuration are given. The results of MELCOR analyses of various accident sequences for the plant operating state (POS) 5 configuration during refueling (approximately Cold Shutdown as defined by Grand Gulf Technical Specifications) are presented for accidents initiated at several different times after scram and shutdown including shortened thermal hydraulic and core damage calculations done in support of the Level 1 analysis and full plant analyses including

containment response and source terms supporting the Level 2 analysis.

NUREG/CR-6144 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Summary Of Results. CHU, T.L.; PRATT, W.T. Brookhaven National Laboratory. October 1995. 118pp. 9511270460. BNL-NUREG-52399. 86317:223.

This document contains a summarization of the results and insights from the Level 1 accident sequence analyses of internally initiated events, internally initiated fire and flood events, seismically initiated events, and the Level 2/3 risk analysis of internally initiated events (excluding fire and flood) for Surry, Unit 1. The analysis was confined to mid-loop operation, which can occur during three plant operational states (identified as POSs R6 and R10 during a refueling outage, and POS D6 during drained maintenance). The report summarizes the Level 1 information contained in Volumes 2-5 and the Level 2/3 information contained in Volume 6 of NUREG/CR-6144.

NUREG/CR-6144 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Evaluation Of Severe Accident Risk During Mid-Loop Operations. Main Report. JO, J.; LIN, C.C.; NEYMOTIN, L.; et al. Brookhaven National Laboratory. May 1995. 145pp. 9506220089. BNL-NUREG-52399. 84394:192.

This document contains the accident progression analysis of internally initiated events for Surry, Unit 1 as it operates in mid-loop operation during drained maintenance or 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 a full power analysis performed on Surry as a part of the Nureg-1150 study.

NUREG/CR-6144 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Evaluation Of Severe Accident Risk During Mid-Loop Operations. Appendices. JO, J.; LIN, C.C.; NEYMOTIN, L.; et al. Brookhaven National Laboratory. May 1995. 275pp. 9507120287. BNL-NUREG-52399. 84633:084.

See NUREG/CR-6144, V06, P1 abstract.

NUREG/CR-6150 V01: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. Interface Theory. ALLISON, C.M.; BERNA, G.A.; CORYELL, E.W.; et al. Idaho National Engineering Laboratory. June 1995. 70pp. 9507180412. EGG-2720. 84716:299.

The SCDAP/RELAP5 code has been developed for best estimate transient simulation of light water reactor coolant systems during a severe accident. The code models the coupled behavior of the reactor coolant system, the core, fission products released during a severe accident transient as well as large and small break loss-of-coolant accidents, 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. This volume describes the organization and manner of the interface between severe accident models which are resident in the SCDAP portion of the code and hydrodynamic models which are resident in the RELAP portion of the code. A description of the organization and structure of SCDAP/RELAP5 is presented. Additional information is provided regarding the manner in which models in one portion of the code impact other parts of the code, and models which are dependent on and derive information from other subcodes.

NUREG/CR-6150 V02: SCDAP/RELAP/MOD 3.1 CODE MANUAL. Damage Progression Model Theory. ALLISON, C.M.; BERNA, G.A.; CHENG, T.C.; et al. Idaho National Engineering Laboratory. June 1995. 190pp. 9507180418. EGG-2720. 84713:172.

The SCDAP/RELAP5 code has been developed for best estimate transient simulation of light water reactor coolant systems during a severe accident. The code models the coupled behavior of the reactor coolant system, the core, fission products released during a severe accident transient as well as large and small break loss-of-coolant accidents, 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. This volume contains detailed descriptions of the severe accident models and correlations. It provides the user with the underlying assumptions and simplifications used to generate and implement the basic equations into the code, so an intelligent assessment of the applicability and accuracy of the resulting calculation can be made.

NUREG/CR-6150 V03: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. User's Guide And Input Manual. ALLISON, C.M.; BERNA, G.A.; CORYELL, E.W.; et al. Idaho National Engineering Laboratory. June 1995. 352pp. 9507180429. EGG-2720. 84715:311.

The SCDAP/RELAP5 code has been developed for best estimate transient simulation of light-water-reactor coolant systems during a severe accident. The code models the coupled behavior of the reactor coolant system, core, fission products released during a severe accident transient as well as large and small break loss-of-coolant accidents, 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. This volume, Volume 3, provides guidelines to code users based upon lessons learned during the developmental assessment process. A description of problem control and the installation process is included. Appendix A contains the description of the input requirements.

NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis. ALLISON, C.M.; BERNA, G.A.; CHAMBERS, R.; et al. Idaho National Engineering Laboratory. June 1995. 674pp. 9507180441. EGG-2720. 84714:001.

The SCDAP/RELAP5 code has been developed for best estimate transient simulation of light-water-reactor coolant systems during a severe accident. The code models the coupled behavior of the reactor coolant system, core, fission products released during a severe accident transient as well as large and small break loss-of-coolant accidents, 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. This volume, Volume 4, describes the material properties correlations and computer subroutines (MATPRO) used by SCDAP/RELAP5. Formulation of the materials properties are generally semi-empirical in nature. A wide variety of material property subroutines are contained in this document. This document also contains descriptions of the reaction and solution rate models needed to analyze a reactor accident.

NUREG/CR-6150 V05: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. Developmental Assessment. ALLISON, C.M.; BERNA, G.A.; BOURDON, S.M.; et al. Idaho National Engineering Laboratory. June 1995. 349pp. 9507180448. EGG-2720. 84717:001.

The SCDAP/RELAP5 code has been developed for best estimate transient simulation of light water reactor coolant systems during a severe accident. The code models the coupled behavior of the reactor coolant system, the core, fission product released during a severe accident transient as well as large and small break loss of coolant accidents, 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. This volume contains detailed code-to-data calculations performed using SCDAP/RELAP5/MOD3.1, as well as comparison calculations performed with earlier code versions. The results of full plant calculations which include Surry, TMI-2, and Browns Ferry are described. The results of a nodalization study, which accounted for both axial and radial nodalization of the core, are also reported.

NUREG/CR-6154 V02: EXPERIMENTAL RESULTS FROM CONTAINMENT PIPING BELLOWS SUBJECTED TO SEVERE ACCIDENT CONDITIONS. Results From Bellows Tested In Corroded Conditions. LAMBERT, L.D.; PARKS, M.B. Sandia National Laboratories. October 1995. 55pp. 9511020400. SAND94-1711. 86091:001.

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 at Sandia National Laboratories under the sponsorship of the U.S. Nuclear Regulatory Commission. 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 nineteen bellows have been tested. Thirteen bellows were tested in "like-new" condition (results reported in Volume 1), and six were tested in a corroded condition. The tests showed that bellows in "like-new" condition are capable of withstanding relatively large deformations, up to, or near, the point of full compression or elongation, before developing leakage, while those in a corroded condition did not perform as well, depending on the amount of corrosion. The corroded bellows test program and results are presented in this report.

NUREG/CR-6159: USING MICRO SAINT TO PREDICT PERFORMANCE IN A NUCLEAR POWER PLANT CONTROL ROOM. A Test Of Validity And Feasibility. LAWLESS, M.T.; LAUGHERY, K.R. Micro Analysis & Design, Inc. PERSENSKY, J.J. Control, Instrumentation & Human Factors Branch (Post 941217). September 1995. 38pp. 9510030209. 85663:255.

The United States Nuclear Regulatory Commission (NRC) requires a technical basis for regulatory actions. In the area of human factors, one possible technical basis is human performance modeling technology including task network modeling. This study assessed the feasibility and validity of task network modeling to predict the performance of control room crews.

Task network models were built that matched the experimental conditions of a study on computerized procedures that was conducted at North Carolina State University. The data from the "paper procedures" conditions were used to calibrate the task network models. Then, the models were manipulated to reflect expected changes when computerized procedures were used. These models' predictions were then compared to the experimental data from the "computerized conditions" of the North Carolina State University study. Analyses indicated that the models predicted some subsets of the data well, but not all. Implications for the use of task network modeling are discussed.

NUREG/CR-6172: REVIEWING PSA-BASED ANALYSES TO MODIFY TECHNICAL SPECIFICATIONS AT NUCLEAR POWER PLANTS. SAMANTA,P.K.; MARTINEZ-GURIDI Brookhaven National Laboratory. VESELY,W.E. Science Applications International Corp. (formerly Science Applications, Inc.). December 1995. 62pp. 9601040277. BNL-NUREG-52428. 86730:170.

Changes to Technical Specifications (TSs) at nuclear power plants (NPPs) require review and approval by the United States Nuclear Regulatory Commission (USNRC). Currently, many requests for changes to TSs use analyses that are based on a plant's probabilistic safety assessment (PSA). This report presents an approach to reviewing such PSA-based submittals for changes to TSs. We discuss the basic objectives of reviewing a PSA-based submittal to modify NPP TSs; the methodology of reviewing a TS submittal, and the differing roles of a PSA review, a PSA Computer Code review, and a review of a TS submittal. To illustrate this approach, we discuss our review of changes to allowed outage time (AOT) and surveillance test interval (STI) in the TS for the South Texas Project Nuclear Generating Station. Based on this experience gained, a check-list of items is given for future reviewers; it can be used to verify that the submittal contains sufficient information, and also that the review has addressed the relevant issues. Finally, recommended steps in the review process and the expected findings of each step are discussed.

NUREG/CR-6173: A SUMMARY OF THE FIRE TESTING PROGRAM AT THE GERMAN HDR TEST FACILITY. NOWLEN,S.P. Sandia National Laboratories. November 1995. 57pp. 9511290120. SAND94-1795. 86353:298.

This report provides an overview of the fire safety experiments performed under the sponsorship of the German government in the containment building of the decommissioned pilot nuclear power plant known as HDR. This structure is a highly complex, multicompartment, multi-level building which has been used as the test bed for a wide range of nuclear power plant operation safety experiments. These experiments have included numerous fire tests. Test fire fuel sources have included gas burners, wood cribs, oil pools, nozzle release oil fires, and cables in cable trays. A wide range of ventilation conditions including full natural ventilation, full forced ventilation, and combined natural and forced ventilation have been evaluated. During most of the tests, the fire products mixed freely with the full containment volume. Macro-scale building circulation patterns which were very sensitive to such factors as ventilation configuration were observed and characterized. Testing also included the evaluation of selective area pressurization schemes as a means of smoke control for emergency access and evacuation stairwells.

NUREG/CR-6184: SEPARATE EFFECTS TESTING AND ANALYSES TO INVESTIGATE LINER TEARING OF THE 1:16-SCALE REINFORCED CONCRETE CONTAINMENT BUILDING. SPLETZER,B.L.; LAMBERT,L.D.; BERGMAN,V.L.; et al. Sandia National Laboratories. May 1995. 250pp. 9507200213. SAND92-1720. 84749:001.

The overpressurization of a 1:16-scale reinforced concrete containment building demonstrated that liner tearing is a plausible failure mode in such structures under severe accident conditions. A combined experimental and analytical program was developed to determine the important parameters that affect liner

tearing and to develop reasonably simple analytical methods for predicting when tearing will occur. Three sets of test specimens were designed to allow individual control over and investigation of the mechanisms believed to be important in causing failure of the liner plate. The series of tests investigated the effect on liner tearing produced by the anchorage system, the loading conditions, and the transition in thickness from the liner to the insert plate. Before testing, the specimens were analyzed using two- and three-dimensional finite element models. Based on the analysis, the failure mode and corresponding load conditions were predicted for each specimen. Test data and post-test examination of test specimens show mixed agreement with the analytical predictions with regard to failure mode and specimen response for most tests. Many similarities were also observed between the response of the liner in the 1:6-scale reinforced concrete containment model and the response of the test specimens. This work illustrates the fact that the failure mechanism of a reinforced concrete containment building can be greatly influenced by details of liner and anchorage system design. Further, it significantly increases the understanding of containment building response under severe conditions.

NUREG/CR-6188 V02: MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE. Annual Report For FY 1994. ROGERS,R.D.; HAMILTON,M.A.; VEEH,R.H.; et al. Idaho National Engineering Laboratory. August 1995. 66pp. 9509130162. INEL-95/0153. 85430:287.

The Nuclear Regulatory Commission stipulates in 10 CFR 61 that disposed low-level radioactive waste (LLW) be stabilized. To provide guidance to disposal vendors and nuclear station waste generators for implementing those requirements, the NRC developed Technical Position on Waste Form, Revision 1. That document details a specified set of recommended testing procedures and criteria, including several tests for determining the biodegradation properties of waste forms. Cement has been widely used to solidify LLW; however, the resulting waste forms are sometimes susceptible to failure due to the actions of waste constituents, stress, and environment. The purpose of this research program is to develop modified microbial degradation test procedures that will be more appropriate than the existing procedures for evaluating the effects of microbiologically influenced chemical attack on cement-solidified LLW. Groups of microorganisms indigenous to LLW disposal sites are being employed that can metabolically convert organic and inorganic substrates into organic and mineral acids. Such acids aggressively react with cement and can ultimately lead to structural failure. Results over the past year on the application of mechanisms inherent in microbially influenced degradation of cement-based material are the focus of this annual report. Data-validated evidence of the potential for microbially influenced deterioration of cement-solidified LLW and subsequent release of radionuclides has been developed during this study.

NUREG/CR-6191: SIZE AND DEFORMATION LIMITS TO MAINTAIN CONSTRAINT IN K(IC) AND J(C) TESTING OF BEND SPECIMENS. KOPPENHOEFER,K.; DODDS,R.H. Illinois, Univ. of, Urbana, IL. October 1995. 27pp. 9511270461. 86322:329.

The ASTM Standard Test Method for Plane-Strain Fracture Toughness of Metallic Materials (E399-90) restricts test specimen dimensions to insure the measurement of highly constrained fracture toughness values (K(IC)). These requirements insure small-scale yielding (SSY) conditions at fracture, and thereby the validity of linear elastic fracture mechanics. Recently, Dodds and Anderson have proposed a less restrictive size requirement for cleavage fracture toughness measured in terms of the J-integral (J(C)), as given by $a, b, B \geq 200 J(C) / \sigma_o$. The size requirement proposed by Dodds and Anderson increases the applicability of fracture toughness experiments by expanding the range of conditions over which fracture toughness data meeting SSY conditions can be reliably measured. This investigation compares the proposed size requirement with that of ASTM Standard Test Method E399 and, by comparison

with published experimental data for various alloys, provides validation of the new requirements.

NUREG/CR-6192: AGING AND SERVICE WEAR OF SPRING-LOADED PRESSURE RELIEF VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS. STAUNTON, R.H.; COX, D.F. Oak Ridge National Laboratory. March 1995. 86pp. 9504180472. ORNL-6791. 83568:226.

Spring-loaded pressure relief valves (PRVs) are used in some safety-related applications at nuclear power plants. In general, they are used in systems where, during accidents, pressures may rise to levels where pressure safety relief is required for protection of personnel, system piping, and components. This report documents a study of PRV aging and considers the severity and causes of service wear and how it is discovered and corrected in various systems, valve sizes, etc. Provided in this report are results of the examination of the recorded failures and identification of trends and relationships/correlations in the failures when all failure-related parameters are considered. Components that comprise a typical PRV, how those components fail, when they fail, and the current testing frequencies and methods are also presented in detail.

NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA. INGERSOLL, D.T.; WHITE, J.E.; WRIGHT, R.Q.; et al. Oak Ridge National Laboratory. January 1995. 172pp. 9502080096. ORNL-6795. 82650:146.

A new multigroup cross-section library based on ENDF/B-VI data has been produced and tested for light water reactor shielding and reactor pressure vessel dosimetry applications. The broad-group library, which is designated BUGLE-93, is intended to replace the aging BUGLE-80 and SAILOR libraries. The processing methodology is consistent with ANSI/ANS 6.1.2, since the ENDF data were first processed into a fine-group, pseudo-problem-independent format and then collapsed into the final broad-group format. The fine-group library, which is designated VITAMIN-B6, contains 120 nuclides. The BUGLE-93 47-neutron-group/20-gamma-ray-group library contains the same 120 nuclides processed as infinitely dilute and collapsed using a weighting spectrum typical of a concrete shield. Additionally, BUGLE-93 contains 105 nuclides processed with resonance self-shielding and weighted using spectra specific to BWR and PWR material compositions and reactor models. Several dosimetry response functions and kerma factors for all 120 nuclides are also included with the library. An extensive integral data testing effort was performed to qualify the new library. In general, results using the new data show significant improvements relative to earlier ENDF data.

NUREG/CR-6220: AN ASSESSMENT OF FIRE VULNERABILITY FOR AGED ELECTRICAL RELAYS. VIGIL, R.A. Science & Engineering Associates, Inc. NOWLEN, S.P. Sandia National Laboratories. March 1995. 38pp. 9504100121. SAND94-0769. 83419:069.

This report details testing to assess the impact of aging on the fire vulnerability of Agastat and General Electric relays. Both aged and unaged relays were tested. Aged relays were subjected to operational cycling under rated load and thermally aged for sixty days. All relays were exposed to one of three different fire temperature profiles in the Severe Combined Environments Test Chamber located at Sandia National Laboratories. The ability to operate properly in the given fire environment was monitored. Results for the aged and unaged relays were examined to determine the impact of aging on the relays' ability to sustain operation under the test conditions. Overall results indicated that the aged relays' performance was not significantly different from that of the unaged relays.

NUREG/CR-6224: PARAMETRIC STUDY OF THE POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS. ZIGLER, G.; BRIDEAU, J.; RAO, D.V.; et al. Science & Engineering Associates, Inc. October 1995. 465pp. 9512260318. SEA 93-554-06A1. 86607:001.

This report documents a plant-specific study for a BWR/4 with a Mark I containment that evaluated the potential for LOCA generated debris and the probability of losing long term recirculation capability due to ECCS pump suction strainer blockage. The major elements of this study were: (1) acquisition of detailed piping layouts and installed insulation details for a reference BWR; (2) analysis of plant specific piping weld failure probabilities to estimate the LOCA frequency; (3) development of an insulation and other debris generation and drywell transport models for the reference BWR; (4) modeling of debris transport in the suppression pool; (5) development of strainer blockage head loss models for estimating loss of NPSH margin; (6) estimation of core damage frequency attributable to loss of ECCS recirculation capability following a LOCA. Elements 2 through 5 were combined into a computer code, BLOCKAGE 2.3. A point estimate of overall DEGB pipe break frequency (per Rx-year) of $1.59E-04$ was calculated for the reference plant, with a corresponding overall ECCS loss of NPSH frequency (per Rx-year) of $1.58E-04$. The calculated point estimate of core damage frequency (per Rx-year) due to blockage related accident sequences for the reference BWR ranged from $4.2E-06$ to $2.5E-05$. The results of this study show that unacceptable strainer blockage and loss of NPSH margin can occur within the first few minutes after ECCS pumps achieve maximum flows when the ECCS strainers are exposed to LOCA generated fibrous debris in the presence of particulates (sludge, paint chips, concrete dust). Generic or unconditional extrapolation of these reference plant calculated results should not be undertaken.

NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES. Experiments And Analysis, March 1990 - December 1994. BRUST, F.W.; SCOTT, P.; RAHMAN, S.; et al. Battelle Memorial Institute, Columbus Laboratories. April 1995. 249pp. 9505040365. BMI-2179. 83784:199.

This topical report summarizes the work performed for the Nuclear Regulatory Commission's (NRC) research program entitled "Short Cracks in Piping and Piping Welds" that specifically focuses on pipes with short through-wall cracks. Previous NRC efforts, conducted under the Degraded Piping Program, focused on understanding the fracture behavior of larger cracks in piping and fundamental fracture mechanics developments necessary for this technology. This report gives details on: (i) material property determinations (ii) pipe fracture experiments, and (iii) development, modification, and validation of fracture analysis methods. The material property data required to analyze the experimental results are included. These data were also implemented into the NRC's PIRFAC database. Three pipe experiments with short through-wall cracks were conducted on large diameter pipe. Also, experiments were conducted on a large-diameter uncracked pipe and a pipe with a moderate-size through-wall crack. The analysis results reported here focus on simple predictive methods based on the J-Tearing theory as well as limit-load and ASME Section XI analyses. Some of these methods were improved for short-crack-length predictions. The accuracy of the various methods was determined by comparisons with experimental results from this and other programs.

NUREG/CR-6239 V01: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Main Report. STEVENSON, J.D. Stevenson & Associates. * Oak Ridge National Laboratory. November 1995. 107pp. 9512260280. ORNL SUB94SD4272. 86626:246.

Since 1982, there has been a major effort expended to evaluate the susceptibility of nuclear power plant equipment to failure

and significant damage during seismic events. This was done by making use of data on the performance of electrical and mechanical equipment in conventional power plants and other similar industrial facilities during strong motion earthquakes. This report is intended as an extension of the seismic experience data collection effort and a compilation of experience data specific to power plant piping and supports designed and constructed to U.S. power piping code requirements which have experienced strong motion earthquakes. Eight damaging (Richter Magnitude 7.7 to 5.5) California earthquakes and their effects on 8 power generating facilities in California were reviewed. All of these facilities were visited and evaluated. Seven fossil-fueled (dual use natural gas and oil) and one nuclear fueled plants consisting of a total of 36 individual boiler or reactor units were investigated. Peak horizontal ground accelerations that either had been recorded on site at these facilities or were considered applicable to these power plants on the basis of nearby recordings ranged between 0.20g and 0.51g with strong motion durations which varied from 3.5 to 15 seconds. Most U.S. nuclear power plants are designed for a safe shutdown earthquake peak ground acceleration equal to 0.20g or less with strong motion durations which vary from 10 to 15 seconds.

NUREG/CR-6239 V02: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Appendices. STEVENSON, J.D. Stevenson & Associates. * Oak Ridge National Laboratory. November 1995. 228pp. 9512260283. ORNLSUB94SD4272. 86608:102.

Volume 2 of the "Survey of Strong Motion Earthquake Effects on Thermal Power Plants in California with Emphasis on Piping Systems" contains Appendices which detail the detail design and seismic response of several power plants subjected to strong motion earthquakes. The particular plants considered include the Ormond Beach, Long Beach and Seal Beach, Burbank, El Centro, Glendale, Humboldt Bay, Kern Valley, Pasadena and Valley power plants. Included is a typical power plant piping specification and photographs of typical power plant piping specification and photographs of typical piping and support installations for the plants surveyed. Detailed piping support spacing data are also included.

NUREG/CR-6240: APPLICATION OF BOUNDING SPECTRA TO SEISMIC DESIGN OF PIPING BASED ON THE PERFORMANCE OF ABOVE GROUND PIPING IN POWER PLANTS SUBJECTED TO STRONG MOTION EARTHQUAKES. STEVENSON, J.D. Stevenson & Associates. * Oak Ridge National Laboratory. February 1995. 104pp. 9503140368. ORNLSUB94SD4271. 83069:312.

This report extends the potential application of Bounding Spectra evaluation procedures, developed as part of the A-46 Unresolved Safety Issue applicable to seismic verification of in-situ electrical and mechanical equipment, to in-situ safety related piping in nuclear power plants. The report presents a summary of earthquake experience data which define the behavior of typical U.S. power plant piping subject to strong motion earthquakes. The report defines those piping system caveats which would assure the seismic adequacy of the piping systems which meet those caveats and whose seismic demand are within the bounding spectra input. Based on the observed behavior of piping in strong motion earthquakes, the report distinguishes between the capabilities of the piping system to carry seismic loads as a function of the type of connection (i.e., threaded vs welded). This report also discusses in some detail the basic causes and mechanisms for earthquake damages and failures to power plant piping systems.

NUREG/CR-6242: CASKS (COMPUTER ANALYSIS OF STORAGE CASKS): A MICROCOMPUTER BASED ANALYSIS

SYSTEM FOR STORAGE CASK DESIGN REVIEW. User's Manual To Version 1b (Including Program Reference). CHEN, T.F.; GERHARD, M.A.; TRUMMER, D.J.; et al. Lawrence Livermore National Laboratory. February 1995. 200pp. 9503140405. UCRL-ID-117418. 83070:058.

CASKS (Computer Analysis of Storage CaskS) is a micro-computer-based system of computer programs and databases developed at the Lawrence Livermore National Laboratory (LLNL) for evaluating safety analysis reports on spent-fuel storage casks. The bulk of the complete program and this user's manual are based upon the SCANS (Shipping Cask Analysis System) program previously developed at LLNL. A number of enhancements and improvements were added to the original SCANS program to meet requirements unique to storage casks. CASKS is an easy-to-use system that calculates global response of storage casks to impact loads, pressure loads and thermal conditions. This provides reviewers with a tool for an independent check on analyses submitted by licensees. CASKS is based on microcomputers compatible with the IBM-PC family of computers. The system is composed of a series of menus, input programs, cask analysis programs, and output display programs. All data is entered through fill-in-the-blank input screens that contain descriptive data requests.

NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Main Report. HARPER, F.T. Sandia National Laboratories. GOOSSENS, L.H.J.; COOKE, R.M.; et al. Netherlands, Govt. of. January 1995. 106pp. 9503150202. EUR 15855EN. 83108:230.

The development of two new probabilistic accident consequence codes, MACCS and COSYMA, was completed in 1990. These codes estimate the consequences from the accidental releases of radiological material from hypothesized accidents at nuclear installations. In 1991, the U.S. Nuclear Regulatory Commission and the Commission of the European Communities began co-sponsoring a joint uncertainty analysis of the two codes. The ultimate objective of this joint effort was to systematically develop credible and traceable uncertainty distributions for the respective code input variables. Because of the magnitude and expense required to complete a full-scale consequence uncertainty analysis, a trial study was performed to evaluate the feasibility of such a joint study by initially limiting efforts to the dispersion and deposition code input variables. A formal expert judgment elicitation and evaluation process was identified as the best technology available for developing a library of uncertainty distributions for these consequence parameters. This report focuses on the methods used in and results of this trial study.

NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Appendices A And B. HARPER, F.T. Sandia National Laboratories. GOOSSENS, L.H.J.; COOKE, R.M.; et al. Netherlands, Govt. of. January 1995. 400pp. 9503150209. EUR 15855EN. 83109:001.

See NUREG/CR-6244, V01 abstract.

NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment. Appendices C, D, E, F, G, H. HARPER, F.T. Sandia National Laboratories. GOOSSENS, L.H.J.; COOKE, R.M.; et al. Netherlands, Govt. of. January 1995. 98pp. 9503140347. EUR 15855EN. 83069:214.

See NUREG/CR-6244, V01 abstract.

NUREG/CR-6251: STAINLESS STEEL SUBMERGED ARC WELD FUSION LINE TOUGHNESS. ROSENFELD, A.R.; HELD, P.R.; WILKOWSKI, G.M. Battelle Memorial Institute, Columbus Laboratories. April 1995. 88pp. 9505190023. BMI-2180. 83980:197.

This effort evaluated the fracture toughness of austenitic steel submerged-arc weld (SAW) fusion lines. The incentive was to

explain why cracks grow into the fusion line in many pipe tests conducted with cracks initially centered in SAWs. The concern was that the fusion line may have a lower toughness than the SAW. It was found that the fusion line, J(lc) was greater than the SAW toughness but much less than the base metal. Of greater importance may be that the crack growth resistance (J(D)-R) of the fusion line appeared to reach a steady-state value, while the SAW had a continually increasing J(D)-R curve. This explains why the cracks eventually turn to the fusion line in the pipe experiments. A method of incorporating these results would be to use the weld metal J-R curve up to the fusion-line steady-state J value. These results may be more important to LBB analyses than the ASME flaw evaluation procedures, since there is more crack growth with through-wall cracks in LBB analyses than for surface cracks in pipe flaw evaluations.

NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Program. Test Results For Fiscal Years 1986, 1987, 1988, And 1989. MCCONNELL, J.W.; ROGERS, R.D.; FINDLAY, M.W.; et al. Idaho National Engineering Laboratory. May 1995. 134pp. 9506020457. INEL-95/0073. 84147:001.

The Field Lysimeter Investigations: Low-Level Waste Data Base Development Program, funded by the U.S. Nuclear Regulatory Commission (NRC), 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 first 4 years 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 experimental equipment is described and results of waste form characterization using tests recommended by the NRC's "Technical Position on Waste Form" are presented. 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-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990, 1991, 1992, And 1993. MCCONNELL, J.W.; ROGERS, R.D. Idaho National Engineering Laboratory. JASTROW, J.D.; et al. Argonne National Laboratory. December 1995. 170pp. 9601290286. INEL-95/0073. 86893:016.

The Field Lysimeter Investigations: Low-Level Waste Data Base Development Program, funded by the U.S. Nuclear Regulatory Commission (NRC), 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 second 4 years 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-East in Illinois and at Oak Ridge National Laboratory. The experimental equipment is described and results of waste form characterization using tests recommended by the NRC's "Technical Position on Waste Form" are presented. 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-6257: CANDU 3 TRANSIENT ANALYSIS USING ATOMIC ENERGY OF CANADA LTD CODES. JEDD, J.L.; SHUMWAY, R.W. Idaho National Engineering Laboratory. EBERT, D.D. Reactor & Plant Systems Branch (Post 941217). January 1995. 88pp. 9503270305. INEL-95/0070. 83268:267.

A limited number of transient scenarios were calculated using a computer code suite and input modeling provided by the Atomic Energy of Canada Limited (AECL) for the CANDU 3 design. Emphasis was placed on a large-break loss-of-coolant accident with delays in actuation of the two independent shutdown systems (shutdown rods and liquid poison injection). Although an extremely unlikely scenario, it was studied because of the potential consequences that would result from a positive void coefficient of reactivity. Results indicate that a few seconds delay in shutdown would result in quickly reaching fuel or cladding melting temperatures before the emergency core cooling system would be activated. Only small changes in the timing and consequences of the scenario result when several parameters, of potential importance to the progression of the accident, are varied. Five calculations were also performed for loss-of-off-site-power scenarios. These calculations assume that the plant failed to enter the island mode, i.e., power to the main coolant pumps was not restored using on-site power generation.

NUREG/CR-6259: CONSTRAINT EFFECTS ON FRACTURE INITIATION LOADS IN HSST WIDE-PLATE TESTS. DODDS, R.H. Illinois, Univ. of, Urbana, IL. DODDS, R.H. Oak Ridge National Laboratory. December 1994. 42pp. 9502080251. UI-LUENG942009. 82660:304.

During the period 1984-1987, researchers of the Heavy-Section Steel Technology program at the Oak Ridge National Laboratory performed a unique series of fracture mechanics tests using exceptionally large, SE(T) specimens (a/W=0.2) fabricated from a reactor pressure vessel material, A533B Class 1 steel. This study re-examines fracture initiation loads in the wide-plate tests using two constraint assessment methodologies developed over the past five years: the J-Q toughness locus approach and the toughness scaling approach based on a local failure criterion for cleavage. Both approaches demonstrate a significant loss of constraint in the elastic-plastic fields ahead of the crack in the wide-plate specimens caused by the inherent negative T-stress of the shallow notch SE(T) configuration. Moreover, the 25mm wide machined notch required for specimen fabrication is shown to further reduce constraint by introducing a traction free surface very near the crack tip. Both of these factors combined to reduce near-tip stresses by 10% below those of the small-scale yielding, SSY (T=0), fields. This reduction places fracture results for the wide-plate specimens within the J-Q toughness locus defined by fracture toughness tests on the A533B material and within the constraint corrected J(c) values defined by the toughness scaling methodology.

NUREG/CR-6260: APPLICATION OF NUREG/CR-5999 INTERIM FATIGUE CURVES TO SELECTED NUCLEAR POWER PLANT COMPONENTS. WARE, A.G.; MORTON, D.K.; NITZEL, M.E. Idaho National Engineering Laboratory. March 1995. 200pp. 9503280383. INEL-95/0045. 83293:001.

Recent test data indicate that the effects of the light water reactor (LWR) environment could significantly reduce the fatigue resistance of materials used in the reactor coolant pressure boundary components of operating nuclear power plants. Argonne National Laboratory has developed interim fatigue curves based on test data simulating LWR conditions, and published them in NUREG/CR-5999. In order to assess the significance of these interim fatigue curves, fatigue evaluations of a sample of the components in the reactor coolant pressure boundary of LWRs were performed. The sample consists of components from facilities designed by each of the four U.S. nuclear steam supply system vendors. For each facility, six locations were studied, including two locations on the reactor pressure vessel. In addition, there are older vintage plants where components of the reactor coolant pressure boundary were designed to codes

that did not require an explicit fatigue analysis of the components. In order to assess the fatigue resistance of the older vintage plants, an evaluation was also conducted on selected components of three of these plants. This report discusses the insights gained from the application of the interim fatigue curves to components of seven operating nuclear power plants.

NUREG/CR-6261: A SUMMARY OF ORNL FISSION PRODUCT RELEASE TESTS WITH RECOMMENDED RELEASE RATES AND DIFFUSION COEFFICIENTS. LORENZ, R.A.; OSBORNE, M.F. Oak Ridge National Laboratory. July 1995. 83pp. 9509070074. ORNL/TM-12801. 85401:146.

Fission product release data from the ORNL HI test series and VI test series are summarized in this report and compared with release results from similar tests performed in France (the HEVA test series). The ORNL test results are also compared with four in-reactor tests, the SNL ST-1 and ST-2 tests, and the INEL SFD 1-3 and SFD 1-4 bundle tests. Test atmospheres range from steam to hydrogen, and the temperature range is 1675 to 2700 K.

NUREG/CR-6263 V01: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS. Candidate Guidelines, Technical Basis And Research Needs. Executive Summary. SETH, S.; BAIL, W.; CLEAVES, D.; et al. MITRE Corp. June 1995. 86pp. 9508020314. MTR 94W0000114. 84908:262.

The work documented in this report was performed in support of the U.S. Nuclear Regulatory Commission to examine the technical basis for candidate guidelines that could be considered in reviewing and evaluating high integrity computer software used in the safety systems of nuclear power plants. The framework for the work consisted of the following software development and assurance activities: requirements specification; design; coding; verification and validation, including static analysis and dynamic testing; safety analysis; operation and maintenance; configuration management; quality assurance; and planning and management. Each activity (framework element) was subdivided into technical areas (framework subelements). The report describes the development of approximately 200 candidate guidelines that span the entire range of software life-cycle activities; the assessment of the technical basis for those candidate guidelines; and the identification, categorization and prioritization of research needs for improving the technical basis. The report has two volumes: Volume 1, Executive Summary, includes an overview of the framework and of each framework element, the complete set of candidate guidelines, the results of the assessment of the technical basis for each candidate guideline, and a discussion of research needs that support the regulatory function; Volume 2 is the main report.

NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS. Candidate Guidelines, Technical Basis And Research Needs. Main Report. SETH, S.; BAIL, W.; CLEAVES, D.; et al. MITRE Corp. June 1995. 430pp. 9508020315. MTR 94W0000114. 84909:001.

See NUREG/CR-6263, V01 abstract.

NUREG/CR-6264 V01: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. An Assessment Of The J(M) Parameter. SHIH, C.F.; LIU, X.H. Brown Univ., Providence, RI. * Battelle Memorial Institute, Columbus Laboratories. February 1995. 35pp. 9503140423. BMI-2181. 83070:282.

Significant advances in elastic-plastic fracture became possible with the introduction of Rice's path independent J-integral which has two physical meanings. First, the J-integral is equivalent to the energy release rate associated with a virtual crack advance. Secondly, J can be regarded as the strength of the stress and strain singularity near a stationary crack tip. As a result of several experimental studies, the J-integral is generally accepted as a valid parameter to characterize a material's resistance to the onset of crack growth under large-scale yielding. Driven by simplicity and the practical benefits that could be derived from a geometry and size-independent material resistance curve for large amounts of crack growth, J(M), a modified J pa-

rameter was introduced. Initial results using J(M) were encouraging but subsequent studies did not support the earlier results. The present computational study presented in Volume 1 of this report investigates several forms of this parameter, how they are derived and the validity of these parameters for small and large amounts of crack growth. It is concluded that neither J nor J(M) (nor any single parameter) can satisfactorily capture the full range of near-tip fracture states. A discussion on the range of validity of J(M) is given in Volume 2.

NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions. SHIH, C.F.; XIA, L. Brown Univ., Providence, RI. HUTCHINSON, J.W.; et al. Harvard Univ., Cambridge, MA. February 1995. 50pp. 9503140430. BMI-2181. 83070:317.

In this report, Volume 2, Mode I crack initiation and growth under plane strain conditions in tough metals are computed using an elastic/plastic continuum model which accounts for void growth and coalescence ahead of the crack tip. The material parameters include the stress-strain properties, along with the parameters characterizing the spacing and volume fraction of voids in material elements lying in the plane of the crack. For a given set of these parameters and a specific specimen, or component, subject to a specific loading, relationships among load, load-line displacement and crack advance can be computed with no restrictions on the extent of plastic deformation. Similarly, there is no limit on crack advance, except that it must take place on the symmetry plane ahead of the initial crack. Suitably defined measures of crack tip loading intensity, such as those based on the J-integral, can also be computed, thereby directly generating crack growth resistance curves. In this report, the model is applied to five specimen geometries which are known to give rise to significantly different crack tip constraints and crack growth resistance behaviors. Computed results are compared with sets of experimental data for two tough steels for four of the specimen types. Details of the load, displacement and crack growth histories are accurately reproduced, even when extensive crack growth takes place under conditions of fully plastic yielding.

NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES. BARRIERE, M.T. Brookhaven National Laboratory. WREATHALL, J. John Wreathall & Co., Inc. COOPER, S.E.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). August 1995. 289pp. 9509200039. BNL-NUREG-52431. 85539:094.

Since the early 1970s, human reliability analysis (HRA) has been considered to be an integral part of probabilistic risk assessments (PRAs). Nuclear power plant (NPP) events, from Three Mile Island through the mid-1980s, showed the importance of human performance to NPP risk. Recent events demonstrate that human performance continues to be a dominant source of risk. In light of these observations, the current limitations of existing HRA approaches become apparent when the role of humans is examined explicitly in the context of real NPP events. The development of new or improved HRA methodologies to more realistically represent human performance is recognized by the Nuclear Regulatory Commission (NRC) as a necessary means to increase the utility of PRAs. To accomplish this objective, an Improved HRA Project, sponsored by the NRC's Office of Nuclear Regulatory Research (RES), was initiated in late February, 1992, at Brookhaven National Laboratory (BNL) to develop an improved method for HRA that more realistically assesses the human contribution to plant risk and can be fully integrated with PRA. This report describes the research efforts including the development of a multidisciplinary HRA framework, the characterization and representation of errors of commission, and an approach for addressing human dependen-

cies. The implications of the research and necessary requirements for further development also are discussed.

NUREG/CR-6266: ANALYSIS OF BORON DILUTION IN A FOUR-LOOP PWR. SUN, J.G.; SHA, W.T. Argonne National Laboratory. March 1995. 80pp. 9504180468. ANL-94/35. 83569:235.

Thermal mixing and boron dilution in a pressurized water reactor were analyzed with COMMIX codes. The reactor system was the four-loop Zion reactor. Two boron dilution scenarios were analyzed. In the first scenario, the plant is in cold shutdown and the reactor coolant system has just been filled after maintenance on the steam generators. To flush the air out of the steam generator tubes, a reactor coolant pump (RCP) is started, with the water in the pump suction line devoid of boron and at the same temperature as the coolant in the system. In the second scenario, the plant is at hot standby and the reactor coolant system has been heated to operating temperature after a long outage. It is assumed that an RCP is started, with the pump suction line filled with cold unborated water, forcing a slug of diluted coolant down the downcomer and subsequently through the reactor core. The subsequent transient thermal mixing and boron dilution that would occur in the reactor system is simulated for these two scenarios. The reactivity insertion rate and the total reactivity are evaluated and a sensitivity study is performed to assess the accuracy of the numerical modeling of the geometry of the reactor coolant system.

NUREG/CR-6273: BIAxIAL LOADING EFFECTS ON FRACTURE TOUGHNESS OF REACTOR PRESSURE VESSEL STEEL. MCAFEE, W.J.; BASS, B.R.; BRYSON, J.W.; et al. Oak Ridge National Laboratory. March 1995. 76pp. 9503270300. ORNL/TM-12866. 83268:191.

The preliminary phases of 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 completed by the Heavy-Section Steel Technology (HSST) Program. The primary objectives of this effort were to analytically and experimentally investigate the effect of biaxial loading on fracture toughness, to quantify this effect through use of existing stress-based, dual-parameter, fracture-toughness correlations, or to propose and verify alternate correlations. A cruciform beam specimen with a two-dimensional shallow, through-thickness flaw and a special loading fixture was designed and fabricated. Tests were performed using biaxial loading ratios of 0:1 (uniaxial), 0.6:1, and 1:1 (equibiaxial). Critical fracture toughness values were calculated for each test. Biaxial loading of 0.6:1 resulted in a reduction in the lower bound fracture toughness of $\approx 12\%$ as compared to the uniaxial tests. The biaxial loading of 1:1 yielded two subsets of toughness values; one agreed well with the uniaxial data while one was reduced by $\approx 43\%$ when compared to the uniaxial data. The results were evaluated using the J-Q theory and the Dodds-Anderson (D-A) micromechanical scaling model. The D-A model predicted no biaxial effect while the J-Q method gave inconclusive results. When applied to the 1:1 biaxial data, these constraint methodologies failed to predict the observed reduction in fracture-toughness obtained in one experiment. A strain-based constraint methodology that considers the relationship between applied biaxial load, the plastic zone size at the flaw tip, and fracture-toughness was formulated and applied successfully to the data. Evaluation of this dual-parameter strain-based model led to the conclusion that it has the capability of representing fracture behavior of RPV steels in the transition region, including the effects of out-of-plane loading on fracture-toughness. This report is designated as HSST Report No. 150.

NUREG/CR-6275: MECHANICAL PROPERTIES OF THERMALLY AGED CAST STAINLESS STEELS FROM SHIPPINGPORT REACTOR COMPONENTS. CHOPRA, O.K.; SHACK, W.J. Argonne National Laboratory. April 1995. 170pp. 9506010453. ANL-94/37. 84121:001.

Thermal embrittlement of static-cast CF-8 stainless steel components from the decommissioned Shippingport reactor has

been characterized. Cast stainless steel materials were obtained from four cold-leg check valves, three hot-leg main shut-off valves, and two pump volutes. The actual time-at-temperature for the materials was ≈ 13 y at ≈ 281 degrees C for the hot-leg components and ≈ 264 degrees C for the cold-leg components. Baseline mechanical properties for as-cast material were determined from tests on either recovery-annealed material or material from the cooler region of the component. The Shippingport materials show modest decreases in fracture toughness and Charpy-impact properties and a small increase in tensile strength because of relatively low service temperatures and ferrite content of the steel. The procedure and correlations developed at Argonne National Laboratory for estimating mechanical properties of cast stainless steels predict accurate or slightly lower values for Charpy-impact energy, tensile flow stress, fracture toughness J-R curve, and J(IC) of the materials. The kinetics of thermal embrittlement and degree of embrittlement at saturation were established from materials that were aged further in the laboratory. The results were consistent with the estimates. The correlations successfully predicted the mechanical properties of the Ringhals 2 reactor hot- and cross-over-leg elbows (CF-8M) after service of ≈ 15 y and the KRB reactor pump cover plate (CF-8) after ≈ 8 y of service.

NUREG/CR-6276: A COMPILATION OF CURRENT REGULATIONS, STANDARDS, AND GUIDELINES IN REMOTE AFTERLOADING BRACHYTHERAPY. TORTORELLI, J.P.; SIMION, G.P.; KOZLOWSKI, S.D. EG&G Idaho, Inc. February 1995. 94pp. 9503270298. EGG-2746. 83268:095.

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-6277 V01: HUMAN FACTORS EVALUATION OF TELETHERAPY. Identification Of Problems And Alternative Approaches. HENRIKSEN, K.; KAYE, R.D.; JONES, R.; et al. Hughes Training, Inc. July 1995. 92pp. 9508010236. 84908:099.

A series of human factors evaluations were undertaken to better understand the contributing factors to human error in the teletherapy environment. Teletherapy is a multidisciplinary methodology for treating cancerous tissue through selective exposure to an external beam of ionizing radiation. A team of human factors specialists, assisted by a panel of radiation oncologists, medical physicists, and radiation therapists, conducted site visits to radiation oncology departments at community hospitals, university centers, and free-standing clinics. A function and task analysis was initially performed to guide subsequent evaluations in the areas of system-user interfaces, procedures, training and qualifications, and organizational policies and practices. The present work focuses solely on training and qualifications of personnel (e.g., training received before and during employment), and the potential impact of organizational factors on the performance of teletherapy. Organizational factors include such topics as adequacy of staffing, performance evaluations, commonly occurring errors, implementation of quality assurance programs, and organizational climate.

NUREG/CR-6277 V02: HUMAN FACTORS EVALUATION OF TELETHERAPY. Function And Task Analysis. KAYE, R.D.; HENRIKSEN, K.; JONES, R.; et al. Hughes Training, Inc. July 1995. 250pp. 9508020293. 84910:067.
See NUREG/CR-6277,V01 abstract.

NUREG/CR-6277 V03: HUMAN FACTORS EVALUATION OF TELETHERAPY. Human-System Interfaces And Procedures. KAYE, R.D.; HENRIKSEN, K.; JONES, R.; et al. Hughes Training, Inc. July 1995. 70pp. 9508020295. 84915:001.
See NUREG/CR-6277,V01 abstract.

34 Main Citations and Abstracts

NUREG/CR-6277 V04: HUMAN FACTORS EVALUATION OF TELETHERAPY. Training And Organizational Analysis. HENRIKSEN, K.; KAYE, R.D.; JONES, R.; et al. Hughes Training, Inc. July 1995. 71pp. 9508020296. 84908:191.
See NUREG/CR-6277, V01 abstract.

NUREG/CR-6277 V05: HUMAN FACTORS EVALUATION OF TELETHERAPY. Literature Review. HENRIKSEN, K.; KAYE, R.D.; JONES, R.; et al. Hughes Training, Inc. July 1995. 113pp. 9508020306. 84915:071.
See NUREG/CR-6277, V01 abstract.

NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY. OFOEGBU, G.I.; HSIUNG, S.-M.; CHOWDHURY, A.H.; et al. Center for Nuclear Waste Regulatory Analyses. April 1995. 103pp. 9505040377. CNWRA 94-017. 83783:158.

The results of a field investigation on the groundwater-hydrologic effect of mining-induced earthquakes are presented. The investigation was conducted at the Lucky Friday Mine Idaho, the groundwater pressure in three fracture zones was monitored over a 24-mo period. The magnitude, source location, and associated ground motions of mining-induced seismic events were also monitored. Several seismic events of magnitude 1.0 or larger were recorded, many of which caused a change in the groundwater pressure. The magnitude of groundwater-pressure change varied with the seismic-event magnitude and source distance. The data was examined using regression analysis. The statistical models obtained predicted the effects of small-magnitude seismic events more satisfactorily than those of larger ones. The observed change in groundwater pressure due to seismic events of magnitude 3.0 or more were larger than those predicted using the statistical model. Based on these results, it is suggested that the effect of earthquakes on groundwater flow may be better understood through mechanistic modeling. The mechanical processes and material behavior that would need to be incorporated in such a model are examined. They include a description of the effect of stress change on the permeability and water storage capacity of a fractured rock mass; transient fluid flow; and the generation and transmission of seismic waves through the rock mass.

NUREG/CR-6284: CRITICALITY SAFETY CRITERIA FOR LICENSE REVIEW OF LOW-LEVEL WASTE FACILITIES. HOPPER, C.M.; ODEGAARDEN, R.H.; PARKS, C.V.; et al. Oak Ridge National Laboratory. March 1995. 45pp. 9504120093. ORNL/TM-12845. 83474:165.

This report provides recommended safety criteria for NRC licensed burial facilities. These criteria have been developed with accepted and consistent nuclear criticality safety evaluation techniques. Additionally, this report provides the bases for the recommended safety criteria by documenting the evaluation methods and assumptions, and by reporting the results of all single-package and array calculations. These criteria were developed with care to assure consistency with data and practices provided in current standards on nuclear criticality safety as well as conformity of the criteria to applicable NRC regulations. The recommended safety criteria are expressed in terms of surface-density spacing criteria, thereby greatly simplifying the application of license conditions for nuclear criticality safety control. This approach was used by an NRC licensee at the Barnwell waste burial facility by limiting the specific controls to the fewest number of parameters consistent with good nuclear safety practice. The use of a surface-density criteria can eliminate the need for numerous license amendments for variations in package contents and specifications.

NUREG/CR-6285: SEVERE ACCIDENT NATURAL CIRCULATION STUDIES AT THE INEL. BAYLESS, P.D.; BROWNSON, D.A.; DOBBE, C.A.; et al. Idaho National Engineering Laboratory. February 1995. 200pp. 9503150171. INEL-94/0016. 83108:014.

Severe accident natural circulation flows have been investigated at the Idaho National Engineering Laboratory to better understand these flows and their potential impacts on the progres-

sion of a pressurized water reactor severe accident. Parameters affecting natural circulation in the reactor vessel and hot legs were identified and ranked based on their perceived importance. Reviews of the scaling of the 1/7-scale experiments performed by Westinghouse were undertaken. RELAP5/MOD3 calculations of two of the experiments showed generally good agreement between the calculated and observed behavior. Analyses of hydrogen behavior in the reactor vessel showed that hydrogen stratification is not likely to occur, and that an initially stratified layer of hydrogen would quickly mix with a recirculating steam flow. An analysis of the upper plenum behavior in the Three Mile Island, Unit 2 reactor concluded that vapor temperatures could have been significantly higher than the temperatures seen by the control rod drive lead screws, supporting the premise that a strong natural circulation flow was likely present during the accident. SCDAP/RELAP5 calculations of a commercial pressurized water reactor severe accident without operator actions showed that the natural circulation flows enhance the likelihood of ex-vessel piping failures long before failure of the reactor vessel lower head.

NUREG/CR-6287: MANAGEMENT CONCEPTS AND SAFETY APPLICATIONS FOR NUCLEAR FUEL FACILITIES. EISNER, H.; SCOTT, R.S. George Washington Univ., Washington, DC. DELICATE, W.S. The KEVRIC Company, Inc. May 1995. 131pp. 9506020467. 84147:137.

This report presents an overview of effectiveness of management control of safety. It reviews several modern management control theories as well as the general functions of management and relates them to safety issues at the corporate and at the process safety management (PSM) program level. Following these discussions, a formal and structured technique for assessing management of the safety function is suggested. Seven modern management control theories are summarized, including business process reengineering, the learning organization, capability maturity, total quality management, quality assurance and control, reliability centered maintenance, and industrial process safety. Each of these theories is examined for its principal characteristics and implications for safety management. The five general management functions of planning, organizing, directing, monitoring, and integrating, which together provide control over all company operations, are discussed. Under the broad categories of Safety Culture, Leadership and Commitment, and Operating Excellence, key corporate safety elements and their subelements are examined. The three categories under which PSM program-level safety issues are described are Technology, Personnel, and Facilities.

NUREG/CR-6291 V01: NUCLEAR PLANT ANALYZER. Installation Manual. SNIDER, D.M.; WAGNER, K.L.; GRUSH, W.H.; et al. Idaho National Engineering Laboratory. January 1995. 28pp. 9502150283. INEL-94/0123. 82730:001.

This report contains the installation instructions for the Nuclear Plant Analyzer (NPA) System. The NPA System consists of the Computer Visual System (CVS) program, the NPA libraries, the associated utility programs. The NPA was developed at the Idaho National Engineering Laboratory under the sponsorship of the U.S. Nuclear Regulatory Commission to provide a highly flexible graphical user interface for displaying the results of these analysis codes. The NPA also provides the user with a convenient means of interactively controlling the host program through user-defined pop-up menus. The NPA was designed to serve primarily as an analysis tool. After a brief introduction to the Computer Visual System and the NPA, an analyst can quickly create a simple picture or set of pictures to aide in the study of a particular phenomenon. These pictures can range from simple collections of square boxes and straight lines to complex representations of emergency response information displays.

NUREG/CR-6291 V02: NUCLEAR PLANT ANALYZER. Analyzer Reference Manual. SNIDER, D.M.; WAGNER, K.L.; GRUSH, W.H.; et al. Idaho National Engineering Laboratory. January 1995. 71pp. 9502150287. INEL-94/0123. 82730:030.

The Nuclear Plant Analyzer (NPA) system provides both a highly flexible graphical user interface for displaying simulation data and, where applicable, a convenient means of interactively controlling the host program through user-defined pop-up menus. The NPA system was developed at the Idaho National Engineering Laboratory under the sponsorship of the U.S. Nuclear Regulatory Commission (NRC). The Computer Visual System and the Analyzer are the primary components of the NPA system. This report contains the reference manual for the Analyzer. It describes both the NPA libraries that constitute the Analyzer and a set of auxiliary programs used in conjunction with the Analyzer.

NUREG/CR-6291 V03: NUCLEAR PLANT ANALYZER. Computer Visual System Reference Manual. SNIDER, D.M.; WAGNER, K.L.; GRUSH, W.H.; et al. Idaho National Engineering Laboratory. January 1995. 128pp. 9502150297. INEL-94/0123. 82730:101.

The Computer Visual System (CVS) Reference Manual describes that part of the Nuclear Plant Analyzer (NPA) system used to create pictures (masks). This manual is intended to guide a user in creating, editing, and animating masks for use in the NPA. The NPA was developed at the Idaho National Engineering Laboratory under the sponsorship of the U.S. Nuclear Regulatory Commission to provide a highly flexible graphical user interface for displaying the results of these analysis codes. The NPA also provides the user with a convenient means of interactively controlling the host program through user-defined pop-up menus. The NPA was designed to serve primarily as an analysis tool. After a brief introduction to the Computer Visual System and the NPA, an analyst can quickly create a simple picture or set of pictures to aid in the study of a particular phenomenon. These pictures can range from simple collections of square boxes and straight lines to complex representations of emergency response information displays.

NUREG/CR-6291 V04: NUCLEAR PLANT ANALYZER. Programmer's Manual. SNIDER, D.M.; WAGNER, K.L.; GRUSH, W.H.; et al. Idaho National Engineering Laboratory. January 1995. 202pp. 9502150309. INEL-94/0123. 82730:229.

The Nuclear Plant Analyzer (NPA) system provides both a highly flexible graphical user interface for displaying simulation data and, where applicable, a convenient means of interactively controlling the host program through user-defined pop-up menus. The NPA system was developed at the Idaho National Engineering Laboratory under the sponsorship of the U.S. Nuclear Regulatory Commission (NRC). This manual is intended to serve as a programmers' guide for the NPA system. As such, it includes technical details regarding the design and implementation of the Computer Visual Systems (CVS) program, the Analyzer, data files used by CVS and the Analyzer, and a series of auxiliary programs that provide important services to NPA users.

NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Main Report. HECHT, H.; HECHT, M.; DINSMORE, G.; et al. SoHaR, Inc. March 1995. 206pp. 9504100125. 83419:107.

High integrity systems include all protective (safety and mitigation) systems for nuclear power plants, and also systems for which comparable reliability requirements exist in other fields, such as in the process industries, in air traffic control, and in patient monitoring and medical systems. Verification aims at determining that each stage in the software development completely and correctly implements requirements that were established in preceding phase, while validation determines that the overall performance of a computer system completely and correctly meets system requirements. Volume I of the report reviews existing classifications for high integrity systems and for the verification and validation procedures, based on assumptions

about the environment in which these systems will be utilized. The final chapter of Volume I deals with a framework for standards in this field. Volume II contains appendices dealing with specific methodologies for system classification, for dependability evaluation, and for two software tools that can automate otherwise very labor intensive verification and validation activities.

NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Appendices A-D. HECHT, H.; HECHT, M.; DINSMORE, G.; et al. SoHaR, Inc. March 1995. 54pp. 9504100137. 83419:304.

See NUREG/CR-6293, V01 abstract.

NUREG/CR-6297: FRACTURE EVALUATIONS OF FUSION LINE CRACKS IN NUCLEAR PIPE BIMETALLIC WELDS. SCOTT, P.; FRANCINI, R.; RAHMAN, S.; et al. Battelle Memorial Institute, Columbus Laboratories. April 1995. 111pp. 9505030455. BMI-2182. 83744:194.

In both BWRs and PWRs there are many locations where carbon steel pipe or components are joined to stainless steel pipe or components with a bimetallic weld. The objective of the research described in this report was to assess the accuracy of current fracture analyses for the case of a crack along a carbon steel to austenitic weld fusion line. To achieve the program objective, material property data and data from a large-diameter pipe fracture experiment were developed to assess current analytical methods. The bimetallic welds evaluated in this program were bimetallic welds obtained from a cancelled Combustion Engineering plant. The welds joined sections of the carbon steel cold-leg piping system to stainless steel safe ends that were to be welded to stainless steel pump housings. The major conclusion drawn as a result of these efforts was that the fracture behavior of the bimetallic weld evaluated in this program could be evaluated with reasonable accuracy using the strength and toughness properties of the carbon steel pipe material in conjunction with conventional elastic-plastic fracture mechanics or limit-load analyses. This may not be generally true for all bimetallic welds, as discussed in this report.

NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE. KRISHNASWAMY, P.; SCOTT, P.; MOHAN, R.; et al. Battelle Memorial Institute, Columbus Laboratories. November 1995. 328pp. 9512040318. BMI-2183. 86383:001.

This topical report summarizes the work performed for the Nuclear Regulatory Commission's (NRC) research program entitled "Short Cracks in Piping and Piping Welds" that specifically focuses on pipes with short, circumferential surface cracks. The following details are provided in this report: (i) material property determinations, (ii) pipe fracture experiments, (iii) development, modification and validation of fracture analysis methods, and (iv) impact of this work on the ASME Section XI Flaw Evaluation Procedures. The material properties developed and used in the analysis of the experiments are included in this report and have been implemented into the NRC's PIFRAC database. Six full-scale pipe experiments were conducted during this program. The analyses methods reported here fall into three categories (i) limit-load approaches, (ii) design criteria, and (iii) elastic-plastic fracture methods. These methods were evaluated by comparing the analytical predictions with experimental data. The results, using 44 pipe experiments from this and other programs, showed that the SC.TNP1 and DPZP analyses were the most accurate in predicting maximum load. New Z-factors were developed using these methods. These are being considered for updating the ASME Section XI criteria.

NUREG/CR-6299: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE. MOHAN, R.; MARSCHALL, C.W.; KRISHNASWAMY, P.; et al. Battelle Memorial Institute, Columbus Laboratories. April 1995. 122pp. 9506020484. 84147:268.

This topical report summarizes the work on angled crack growth and combined loading effects performed within the Nuclear Regulatory Commission's research program, entitled "Short Cracks in Piping and Piping Welds". The major impetus for this work stemmed from the observation that initial circumferential cracks in carbon steel pipes exhibited angular crack growth. This failure mode was little understood, and the effect of angled crack growth from an initially circumferential crack raised questions of how pipes under combined loading with torsional stresses would behave. There were three major conclusions from this work. The first was that virtually all ferritic nuclear pipes will have toughness anisotropy. The second was that the ratio of the normalized crack driving force (as a function of angle) to the normalized toughness (also as a function of the angle of crack growth) showed that there was an equal likelihood of cracks growing at any angle between 25 and 65 degrees. This agreed with the scatter of crack growth angles observed in pipe tests. Third, for combined loads with torsional stresses, an effective moment allows pure bending analyses to be used up to crack initiation. Crack opening area under combined loads could also be determined in this manner.

NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES. RAHMAN,S.; BRUST,F.W.; GHADIALI,N.; et al. Battelle Memorial Institute, Columbus Laboratories. April 1995. 227pp. 9505030464. BMI-2184. 83745:001.

Leak-before-break (LBB) analyses for circumferentially cracked pipes are currently being conducted in the nuclear industry to justify elimination of pipe whip restraints and jet impingement shields which are present because of the expected dynamic effects from pipe rupture. The application of the LBB methodology frequently requires calculation of leak rates. The leak rates depend on the crack-opening area of the through-wall crack in the pipe. In addition to LBB analyses which assume a hypothetical flaw size, there is also interest in the integrity of actual leaking cracks corresponding to current leakage detection requirements in NRC Regulatory Guide 1.45, or for assessing temporary repair of Class 2 and 3 pipes that have leaks as are being evaluated in ASME Section XI. This study was requested by the NRC to review, evaluate, and refine current analytical models for crack-opening-area analyses of pipes with circumferential through-wall cracks. Twenty-five pipe experiments were analyzed to determine the accuracy of the predictive models. Several practical aspects of crack-opening such as; crack-face pressure, off-center cracks, restraint of pressure-induced bending, cracks in thickness transition regions, weld residual stresses, crack-morphology models, and thermal-hydraulic analysis, were also investigated.

NUREG/CR-6305: BLT-EC (BREACH, LEACH, TRANSPORT, AND EQUILIBRIUM CHEMISTRY), A FINITE-ELEMENT MODEL FOR ASSESSING THE RELEASE OF RADIONUCLIDES FROM LOW-LEVEL WASTE DISPOSAL UNITS. Background, Theory, And Model Description. MACKINNON,R.J.; SULLIVAN,T.M. Brookhaven National Laboratory. SIMONSON,S.A.; et al. Massachusetts Institute of Technology, Cambridge, MA. August 1995. 155pp. 9509200030. BNL-NUREG-52446. 85540:001.

Performance assessment models typically account for sorption and dissolution-precipitation by using an empirical distribution coefficient $K(d)$ that combines the effects of all chemical reactions between solid and aqueous phases. There is an increasing awareness that performance assessments based solely on empirically based $K(d)$ models may be incomplete, particularly for applications involving radionuclides having sorption and solubility properties that are sensitive to variations in the in situ chemical environment. To accommodate such variations and to assess impact on radionuclide mobility, one must model radionuclide release, transport, and chemical processes in a coupled fashion. This modeling was incorporated into the two-dimensional, finite-element, computer code BLT-EC which can predict container degradation, waste-form leaching, and ad-

vective-dispersive, multispecies, solute transport. BLT-EC accounts for retardation directly by modeling the chemical processes of complexation, sorption, dissolution-precipitation, ion-exchange, and oxidation-reduction reactions. Herein we: (1) describe in detail physical and chemical processes that control the release and migration of radionuclides from shallow land LLW disposal facilities; (2) formulate the mathematical models that represent these processes; (3) outline how these models are incorporated and implemented in BLT-EC; and (4) demonstrate application of BLT-EC on sample problems.

NUREG/CR-6307: SUMMARY OF COMMENTS RECEIVED AT WORKSHOP ON USE OF A SITE SPECIFIC ADVISORY BOARD (SSAB) TO FACILITATE PUBLIC PARTICIPATION IN DECOMMISSIONING CASES. CAPLIN,J.; PAGE,G.; SMITH,D.; et al. Advanced Systems Technology, Inc. June 1995. 115pp. 9507120259. 84634:190.

The Nuclear Regulatory Commission (NRC) is conducting an enhanced participatory rulemaking to establish radiological criteria for the decommissioning of NRC-licensed facilities. As part of this rulemaking, on August 20, 1994 the NRC published a proposed rule for public comment. Paragraph 20.1406(b) of the proposed rule would require that the licensee convene a Site Specific Advisory Board (SSAB) if the licensee proposed release of the site for restricted use after decommissioning. To encourage comment the NRC held a workshop on the subject of SSABs on December 6, 7, and 8, 1994. This report summarizes the 567 comments categorized from the transcript of the workshop. The commenters at the workshop generally supported public participation in decommissioning cases. Many participants favored promulgating requirements in the NRC's rules. Some industry participants favored relying on voluntary exchanges between the public and the licensees. Many participants indicated that a SSAB or something functionally equivalent is needed in controversial decommissioning cases, but that some lesser undertaking can achieve meaningful public participation in other cases. No analysis or response to the comments is included in this report.

NUREG/CR-6308: AN OVERVIEW OF INSTABILITY AND FINGERING DURING IMMISCIBLE FLUID FLOW IN POROUS AND FRACTURED MEDIA. CHEN,G.; TANIGUCHI,M.; NEUMAN,S.P. Arizona, Univ. of, Tucson, AZ. April 1995. 137pp. 9507060331. 84532:043.

Wetting front instability is an important phenomenon affecting fluid flow and contaminant transport in unsaturated soils and rocks. It causes the development of fingers which travel faster than would a uniform front and thus bypass much of the medium. Water saturation and solute concentration in such fingers tend to be higher than in the surrounding medium. During infiltration, fingering may cause unexpectedly rapid arrival of water and solute at the water table. This notwithstanding, most models of subsurface flow and transport ignore instability and fingering. In this report, we survey the literature to assess the extent to which this may or may not be justified. Our overview covers experiments, theoretical studies, and computer simulations of instability and fingering during immiscible two-phase flow and transport, with emphasis on infiltration into soils and fractured rocks. Our description of instability in an ideal fracture (Hele-Shaw cell) includes an extension of existing theory to fractures and interfaces having arbitrary orientations in space. Our discussion of instability in porous media includes a slight but important correction of existing theory for the case of an inclined interface. We conclude by outlining some potential directions for future research. Among these, we single out the effect of soil and rock heterogeneities on instability and preferential flow as meriting special attention in the context of nuclear waste storage in unsaturated media.

NUREG/CR-6310: AN ANALYSIS OF POTASSIUM IODIDE (KI) PROPHYLAXIS FOR THE GENERAL PUBLIC IN THE EVENT OF A NUCLEAR ACCIDENT. BEHLING, H.; BEHLING, K. S. Cohen & Associates, Inc. AMARASOORIYA, H.; et al. SCIEN-TECH, Inc. February 1995. 200pp. 9503150166. 83107:157.

A generic difficulty countered in cost-benefit analyses is the quantification of major elements that define the costs and the benefits in commensurate units. In this study, the costs of making KI available for public use, and the avoidance of thyroidal health effects predicted to be realized from the availability of that KI (i.e., the benefits), are defined in the commensurate units of dollars.

NUREG/CR-6311: EVALUATING PREDICTION UNCERTAINTY. MCKAY, M.D. Los Alamos National Laboratory. March 1995. 69pp. 9503270307. LA-12915-MS. 83270:274.

The probability distribution of a model prediction is presented as a proper basis for evaluating the uncertainty in a model prediction that arises from uncertainty in input values. Determination of important model inputs and subsets of inputs is made through comparison of the prediction distribution with conditional prediction probability distributions. Replicated Latin hypercube sampling and variance ratios are used in estimation of the distributions and in construction of importance indicators. The assumption of a linear relation between model output and inputs is not necessary for the indicators to be effective. A sequential methodology which includes an independent validation step is applied in two analysis applications to select subsets of input variables which are the dominant causes of uncertainty in the model predictions. Comparison with results from methods which assume linearity shows how those methods may fail. Finally, suggestions for treating structural uncertainty for submodels are presented.

NUREG/CR-6312: ASSESSMENT OF FIBER OPTIC PRESSURE SENSORS. HASHEMIAN, H.M.; BLACK, C.L.; FARMER, J.P. Analysis & Measurement Services Corp. April 1995. 132pp. 9506010456. 84121:170.

The principle of operation of fiber optic pressure sensors and the potential of these sensors for use in nuclear power plants are described in this report. Also included is a review of current research on fiber optic sensing technologies, a comparison of fiber optic pressure sensors with conventional pressure sensors, a discussion on advantages and disadvantages of fiber optic pressure sensors, a review of failure modes of these sensors, and results of a survey of fiber optic sensor manufacturers.

NUREG/CR-6313 V01: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEM. Summary of Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems. GOODENOW, T.C.; SHIPMAN, R.L.; HOLLAND, H.M. Epoch Engineering, Inc. June 1995. 74pp. 9507200222. 84749:285.

Epoch Engineering, Incorporated (EEI) has completed a series of vibration measurements comparing their newly-developed Robust Laser Interferometer (RLI) with accelerometer instrumentation systems. EEI has successfully demonstrated, on several pieces of commonplace machinery, that non-contact, line-of-sight measurements are practical and yield results equal to or, in some cases, better than customary field implementations of accelerometers. The demonstration included analysis and comparison of such phenomena as nonlinearity, transverse sensitivity, harmonics, and signal-to-noise ratio. Fast Fourier Transformations were performed on the accelerometer and the laser system outputs to provide a comparison basis. The RLI was demonstrated, within the limits of the task, to be a viable, line-of-sight, non-contact alternative to accelerometer systems. Several different kinds of machinery were instrumented and compared, including a small pump, a gear-driven cement mixer, a rotor kit, and two small fans. Known machinery vibration sources were verified and RLI system output file formats were verified to be compatible with commercial computer programs used for vibration monitoring and trend analysis. The RLI was

also observed to be less subject to electromagnetic interference (EMI) and more capable at very low frequencies.

NUREG/CR-6313 V02: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEMS. Supplemental Appendices Presenting Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems. GOODENOW, T.C.; SHIPMAN, R.L.; HOLLAND, H.M. Epoch Engineering, Inc. June 1995. 200pp. 9507200236. 84750:092.

See NUREG/CR-6313, V01 abstract.

NUREG/CR-6315: CANDU REACTORS, THEIR REGULATION IN CANADA, AND THE IDENTIFICATION OF RELEVANT NRC SAFETY ISSUES. CHARAK, I.; KIER, P.H. Argonne National Laboratory. April 1995. 48pp. 9505180578. ANL-95/5. 83980:034.

Atomic Energy of Canada, Limited (AECL) and its subsidiary in the United States, are considering submitting the CANDU 3 design for standard design certification under 10 CFR Part 52. CANDU reactors are pressurized heavy water power reactors. They have some substantially different safety responses and safety systems than the LWRs that the commercial power reactor licensing regulations of the U.S. Nuclear Regulatory Commission (NRC) have been developed to deal with. In this report, the authors discuss the basic design characteristics of CANDU reactors, specifically of the CANDU 3 where possible, and some safety-related consequences of these characteristics. The authors also discuss the Canadian regulatory provisions, and the CANDU safety systems that have evolved to satisfy the Canadian regulatory requirements as of December 1992. Finally, the authors identify NRC regulations, mainly in 10 CFR Parts 50 and 100, with issues for CANDU 3 reactor designs. In all, eleven such regulatory issues are identified. They are: (1) the ATWS rule (Section 50.62); (2) station blackout (Section 50.63); (3) conformance with Standard Review Plan (SRP); (4) appropriateness of the source term (Section 50.34(f) & Section 100.11); (5) applicability of reactor coolant pressure boundary (RCPB) requirements (Section 50.55a, etc); (6) ECCS acceptance criteria (Section 50.46(b)); (7) combustible gas control (Section 50.44, etc); (8) power coefficient of reactivity (GDC 11); (9) seismic design (Part 100); (10) environmental impacts of the fuel cycle (Section 51.51); and (11) (standards Section 50.55a).

NUREG/CR-6316 V01: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. MILLER, L.A.; HAYES, J.E.; MIRSKY, S.M. Science Applications International Corp. (formerly Science Applications, Inc.). March 1995. 178pp. 9504180393. SAIC-95/1028. 83566:198.

This report provides the project summary of the results of the Expert System Verification and Validation (V&V) activity that was jointly funded by the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute to develop guidelines for the V&V of expert and other systems. This is the first volume of an eight-volume report. The project began with a survey of conventional V&V methods that covers 153 different techniques. Quantitative cost-benefit and an effectiveness measures were developed to permit comparisons among all the methods for three levels of stringency of V&V: low, medium, and high (Classes 3 to 1, respectively). A survey was conducted concerning V&V practices in use for expert systems, finding that they were not common, but that there was considerable activity in developing methods for knowledge bases. Selected V&V methods were applied to two existing expert systems used in nuclear power applications. Other V&V methods were investigated in an empirical experiment to assess their practical utility. A method for generating validation scenarios was developed. Finally, a set of guidelines recommending specific V&V methods for 16 different system-development situations was developed.

NUREG/CR-6316 V02: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Assessment Of Conventional Software Verification And Validation Methods. MILLER, L.A.; GROUNDWATER, E.H.; HAYES, J.E.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). March 1995. 190pp. 9504180403. SAIC-95/1028. 83567:017.

By means of literature survey, a comprehensive set of methods was identified for the verification and validation of conventional software. The 153 methods so identified were classified according to their appropriateness of various phases of a development lifecycle -- requirements, design, and implementation; the last category was subdivided into two, static testing and dynamic testing methods. The methods were then characterized in terms of eight rating factors, four concerning ease-of-use of the methods and four concerning the methods; power to detect defects. Based on these and an Effectiveness Metric. The Effectiveness Metric was further refined to provide three different estimates for each method, depending on three classes of needed stringency of V&V (determined by ratings of a system's complexity and required integrity). Methods were then rank-ordered for each of the three classes in terms of their overall cost-benefits and effectiveness. The applicability was then assessed of each method for the four identified components of knowledge-based and expert systems, as well as the system as a whole.

NUREG/CR-6316 V03: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Documentation Of Expert System Verification And Validation Methodologies. GROUNDWATER, E.H.; MILLER, L.A.; MIRSKY, S.M. Science Applications International Corp. (formerly Science Applications, Inc.). March 1995. 112pp. 9504180421. SAIC-95/1028. 83567:204.

This report is the third volume in a series of reports describing the results of the Expert System Verification and Validation (V&V) project that is jointly funded by the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute to develop guidelines for the V&V of expert and other systems. The purpose of this activity was to survey and document techniques presently in use for expert systems V&V. Via extensive telephone contacts, site visits, and through bibliographic searches a wide sampling of expert system V&V was accomplished. The major finding was that V&V of expert systems is not nearly as established or prevalent as V&V of conventional software systems. There were few examples of V&V in the early stage of development. However, there is a very active research area concerning the development of methods to assess the knowledge bases of expert and knowledge-based systems.

NUREG/CR-6316 V04: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Evaluation Of Knowledge Base Certification Methods. MILLER, L.A.; HAYES, J.E.; MIRSKY, S.M. Science Applications International Corp. (formerly Science Applications, Inc.). March 1995. 136pp. 9504180425. SAIC-95/1028. 83568:001.

This report is the fourth volume in a series of reports describing the results of the Expert System Verification and Validation (V&V) project that is jointly funded by the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute to develop guidelines for the V&V of expert and other systems. Here are presented the results of the Knowledge Base Certification activity that was concerned with developing and testing various static analysis methods for assuring the quality of knowledge bases. The testing procedure used was that of a behavioral experiment involving evaluation of four different V&V methods. The study used two real nuclear expert systems: a boiling water reactor emergency operating procedures tracking system, and a pressurized water reactor safety assessment system. The twenty participants were from three nuclear utilities, the USNRC Technical Training Center, the University of Maryland, EG&G

Idaho, and SAIC. The major conclusion was that the use of tools in static knowledge base certification results in significant improvement in detecting all types of defects, avoiding false alarms, and completing the effort in less time. The simulated knowledge-checking tool, based on supplemental information, was the most effective of the tools.

NUREG/CR-6316 V05: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Rationale And Description Of V&V Guideline Packages And Procedures. MILLER, L.A.; HAYES, J.E.; MIRSKY, S.M. Science Applications International Corp. (formerly Science Applications, Inc.). March 1995. 103pp. 9504180441. SAIC-95/1028. 83568:131.

This report is the fifth volume of a series describing the results of the Expert System Verification and Validation (V&V) project jointly funded by the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute, to formulate guidelines for the V&V of expert and other systems. This report provides the rationale for and description of those guidelines. The actual guidelines are presented in Volume 7, "User's Manual." Three factors determine what V&V is needed: (1) the stage of the development lifecycle; (2) whether the overall system or a specialized component needs to be tested; and (3) the stringency of V&V that is needed. A V&V guideline package is provided for each of the combinations of these three variables. The package specifies the V&V methods recommended and the order in which they should be administered, the assurances each method provides, the qualifications needed by the V&V team, the performance measures that should be taken, and the decision criteria. In addition to the guideline packages, highly detailed step-by-step procedures are provided for 11 of the most important methods, to ensure that they can be implemented correctly. The guidelines can apply to conventional as well as to AI systems.

NUREG/CR-6316 V06: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Validation Scenarios. MIRSKY, S.M.; HAYES, J.E.; MILLER, L.A. Science Applications International Corp. (formerly Science Applications, Inc.). March 1995. 57pp. 9504180445. SAIC-95/1028. 83567:312.

This report is the sixth volume in a series of reports describing the results of the Expert System Verification and Validation (V&V) project that is jointly funded by the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute to develop guidelines for the V&V of expert and other systems. This activity was concerned with the development of a methodology for selecting "validation scenarios." These are defined as "realistic dynamic tests of software which covers only the intended range of applications of the software and are designed to sample important subsets of functions, usually for selected situations known to be challenging or problematic, to provide assurance that the system achieves the tested functions with the required accuracy and performance." Such scenarios are used after all the V&V testing of the system is completed. Five categories of validation scenarios were defined: PLANT, TEST, BASICS, CODE, and LICENSING. A sixth type, REGRESSION, is a composite of the others and refers to the practice of using trusted scenarios to ensure that software modifications did not unintentionally change non-modified functions. A generalized procedure was developed for generating appropriate sets of validation scenarios from these basic categories.

NUREG/CR-6316 V07: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. User's Manual. MILLER, L.A.; HAYES, J.E.; MIRSKY, S.M. Science Applications International Corp. (formerly Science Applications, Inc.). March 1995. 250pp. 9504180447. SAIC-95/1028. 83569:001.

This report provides a step-by-step guide, or user manual, for personnel responsible for the planning and execution of the ver-

ification and validation (V&V), and also developmental testing, of expert systems, conventional software systems, and also various other types of artificial intelligence systems. While the guide was developed primarily for applications in the utility industry, it applies well to all industries. The user manual has three sections. In Section 1 the user assesses the stringency of V&V needed for the system under consideration, identifies the development stage the system is in, and identifies the component(s) of the system to be tested next. These three pieces of information determine which package of V&V methods, called a Guideline Package, is most appropriate for those conditions. The V&V Guidelines Packages are provided in Section 2. Each package consists of an ordered set of V&V techniques to be applied to the system, along with guides as to the review/evaluation team, and the measurement criteria. In Section 3, the details of 11 of the most important (or least-well explained in the literature) methods are presented to assist the user in the accurate application of these techniques.

NUREG/CR-6316 V08: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Bibliography. MILLER, L.A.; HAYES, J.E.; MIRSKY, S.M. Science Applications International Corp. (formerly Science Applications, Inc.). March 1995. 55pp. 9504180451. SAIC-95/1028. 83568:308.

This volume contains all of the technical references found in Volumes 1-7 concerning the development of guidelines for the verification and validation of expert systems, knowledge-based systems, other AI systems, object-oriented systems, and conventional systems.

NUREG/CR-6318: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-7. OSBORNE, M.F.; LORENZ, R.A.; TRAVIS, J.R.; et al. Oak Ridge National Laboratory. May 1995. 65pp. 9506220072. ORNL/TM-12937. 84394:001.

Test VI-7 was the final 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 Monticello boiling water reactor (BWR). The fuel had experienced a burnup of ~40 Mwd/kg U. It was heated in an induction furnace for successive 20-min periods at 2000 and 2300 K in a moist air-helium atmosphere. Integral releases were 69% for ⁸⁵Kr, 52% for (¹²⁵Sb), 71% for both (¹³⁴Cs) and (¹³⁷Cs), and 0.04% for (¹⁵⁴Eu). For the non-gamma-emitting species, release values for 42% for I, 4.1% for Ba, 5.3% for Mo, and 1.2% for Sr were determined. The total mass released from the furnace to the collection system, including fission products, fuel, and structural materials, was 0.89 g, with 37% being collected on the thermal gradient tubes and 63% downstream on filters. Posttest examination of the fuel specimen indicated that most of the cladding was completely oxidized to ZrO₂, but that oxidation was not quite complete at the upper end. The release behaviors for the most volatile elements, Kr and Cs, were in good agreement with the ORNL-Booth Model.

NUREG/CR-6322: BUCKLING ANALYSIS OF SPENT FUEL BASKET. LEE, A.S.; BUMPAS, S.E. Lawrence Livermore National Laboratory. May 1995. 60pp. 9505230342. UCRL-ID-119697. 84026:245.

The basket for a spent fuel shipping cask is subjected to compressive stresses that may cause global instability of the basket assemblies or local buckling of the individual members. Adopting the common buckling design practice in which the stability capacity of the entire structure is based on the performance of the individual members of the assemblies, the typical spent fuel basket, which is composed of plates and tubular structural members, can be idealized as an assemblage of columns, beam-columns and plates. This report presents the flexural buckling formulas for five load cases that are common in the basket buckling analysis: column under axial loads, column under axial and bending loads, plates under uniaxial loads, plates under biaxial loadings, and plate under biaxial loads and lateral pressure. The acceptance criteria from the ASME Boiler

and Pressure Vessel Code are used to determine the adequacy of the basket components. Special acceptance criteria are proposed to address the unique material characteristics of austenitic stainless steel, a material which is frequently used in the basket assemblies.

NUREG/CR-6323: RELATIVE RISK ANALYSIS IN REGULATING THE USE OF RADIATION-EMITTING MEDICAL DEVICES. A Preliminary Application. JONES, E.D.; BANKS, W.W.; ALTENBACH, T.J.; et al. Lawrence Livermore National Laboratory. September 1995. 181pp. 9510060245. UCRL-ID-120051. 85746:001.

This report describes a preliminary application of an analysis approach for assessing relative risks in the use of radiation-emitting medical devices. Results are presented on human-initiated actions and failure modes that are most likely to occur in the use of the Gamma Knife, a gamma irradiation therapy device. This effort represents an initial step in a U.S. Nuclear Regulatory Commission (NRC) plan to evaluate the potential role of risk analysis in regulating the use of nuclear medical devices. For this preliminary application of risk assessment, the focus was to develop a basic process using existing techniques for identifying the most likely risk contributors and their relative importance. The approach taken developed relative risk rankings and profiles that incorporated the type and quality of data available and could present results in an easily understood form. This work was performed by the Lawrence Livermore National Laboratory for the NRC.

NUREG/CR-6324: QUALITY ASSURANCE FOR GAMMA KNIVES. JONES, E.D.; BANKS, W.W.; FISCHER, L.E. Lawrence Livermore National Laboratory. September 1995. 146pp. 9510060248. UCRL-ID-120056. 85745:121.

This report describes and summarizes the results of a quality assurance (QA) study of the Gamma Knife, a nuclear medical device used for the gamma irradiation of intracranial lesions. The study's focus was on the physical aspects of QA and did not address issues that are essentially medical, such as patient selection or prescription of dose. A risk-based QA assessment approach was used. In this report, sample programs for quality control and assurance are included. The use of the Gamma Knife was found to conform to existing standards and guidelines concerning radiation safety and quality control of external beam therapies (shielding, safety reviews, radiation surveys, interlock systems, exposure monitoring, good medical physics practices, etc.) and to be compliant with NRC teletherapy regulations. There are, however, current practices for the Gamma Knife not covered by existing, formalized regulations, standards, or guidelines. These practices have been adopted by Gamma Knife users and continue to be developed with further experience. Some of these have appeared in publications or presentations and are slowly finding their way into recommendations of professional organizations.

NUREG/CR-6325: AN IMPLICIT STEADY-STATE INITIALIZATION PACKAGE FOR THE RELAP5 COMPUTER CODE. PAULSEN, M.P.; PETERSON, C.E. Computer Simulation & Analysis, Inc. August 1995. 142pp. 9509130168. 85433:001.

A direct steady-state initialization (DSSI) method has been developed and implemented in the RELAP5 hydrodynamic analysis program. It provides a means for users to specify a small set of initial conditions which are then propagated through the remainder of the system. The DSSI scheme utilizes the steady-state form of the RELAP5 balance equations for nonequilibrium two-phase flow. It also employs the RELAP5 component models and constitutive model packages for wall-to-phase and inter-phase momentum and heat exchange. A fully implicit solution of the linearized hydrodynamic equations is implemented. An implicit coupling scheme is used to augment the standard steady-state heat conduction solution for steam generator use. It solves the primary-side tube region energy equations, heat conduction equations, wall heat flux boundary conditions, and over-

all energy balance equations as a coupled system of equations and improves convergence. The DSSI methods for initializing RELAP5 problems to steady-state conditions has been compared with the transient solution scheme using a suite of test problems including; adiabatic single-phase liquid and vapor flow through channels with and without heating and area changes; a heated two-phase test bundle representative of BWR core conditions; and a single-loop PWR model.

NUREG/CR-6327: MODELS FOR EMBRITTLEMENT RECOVERY DUE TO ANNEALING OF REACTOR PRESSURE VESSEL STEELS. EASON, E.D.; WRIGHT, J.E.; NELSON, E.E.; et al. Modeling & Computer Services. May 1995. 76pp. 9505190018. MCS 950302. 83982:001.

The irradiation embrittlement of the reactor pressure vessel in nuclear power plants can be reduced by thermal annealing at temperatures higher than the normal operating conditions. The objective of this work was to analyze the pertinent data and develop quantitative models for estimating the recovery in 30 ft-lb (41 J) Charpy transition temperature and Charpy and Charpy upper shelf energy due to annealing. An analysis data base was developed, reviewed for completeness and accuracy, and documented as part of this work. Independent variables considered in the analysis included material chemistries, annealing time and temperature, irradiation time and temperature, fluence, and flux. To identify important variables and functional forms for predicting embrittlement recovery, pattern recognition, transformation analysis, and residual analysis were applied together with current understanding of the mechanisms governing embrittlement and recovery. Models were calibrated using multivariable surface-fitting techniques. The quality of fit was evaluated by considering both the Charpy annealing data used for fitting and a surrogate hardness data base. Several iterations of model calibration, evaluation with respect to mechanistic and statistical considerations, and comparison with the trends in hardness data produced improved correlation models for estimating Charpy upper shelf energy and transition temperature after irradiation and annealing.

NUREG/CR-6328: ADEQUACY OF THE 123-GROUP CROSS-SECTION LIBRARY FOR CRITICALITY ANALYSES OF WATER-MODERATED URANIUM SYSTEMS. PARKS, C.V.; WRIGHT, R.Q.; JORDAN, W.C. Oak Ridge National Laboratory. August 1995. 29pp. 9508300310. ORNL/TM-12970. 85287:291.

In a recent criticality analysis for an array of water-moderated packages containing high-enriched uranium, the 123-group cross-section library in the SCALE system was observed to have a nonconservative discrepancy of approximately 3 to 3.5% when compared to more recently developed libraries. A simple representative system of $\text{UO}_2/\text{F}_2\text{O} \cdot \text{H}_2\text{O}$ was used to identify that the problem results from a lack of resonance data for $(235)\text{U}$. Only a single set of self-shielded cross sections, most likely corresponding to a water-moderated infinite dilute system, was provided with the original data. The $\text{UO}_2/\text{F}_2\text{O} \cdot \text{H}_2\text{O}$ study indicates that this limitation may cause nonconservative discrepancies as high as 5.5% for some water-moderated, high-enriched uranium systems. Characteristics of the systems where the discrepancy is evident are identified and discussed.

NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS. LACH, D.; MELBER, B.; BRICHOUX, J.; et al. Battelle Human Affairs Research Centers. June 1995. 100pp. 9507060313. PNL-10548. 84533:007.

This report presents the findings of a regulatory impact survey of nuclear materials licensees of the United States Nuclear Regulatory Commission (NRC). Commissioners of the NRC directed staff to provide the Commission with first hand information from licensees that could be used to improve the overall regulatory program. A self-administered, mail-out survey questionnaire was used to collect data from a sample of licensees who had interaction with the NRC during the previous 12

months. A total of 371 respondents of the 589 who were sent questionnaires returned completed surveys, for a response rate of 63%. The body of the report presents the findings of the survey including a brief introduction to the approach used, followed by survey findings regarding regulations, policies and regulatory guidance; experience with licensing applications, renewals and amendments; inspections; reporting requirements; and enforcement actions. The appendices of the report include a copy of the survey as administered to licensees, a fuller description of the survey design and data collection methods, and detailed graphic material describing survey responses.

NUREG/CR-6331: ATMOSPHERIC RELATIVE CONCENTRATIONS IN BUILDING WAKES. RAMSDALL, J.V.; SIMONEN, C.A.; SMYTH, S.B. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1995. 129pp. 9506220081. PNL-10521. 84394:066.

This report documents the ARCON95 computer code developed for the U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research for use in control room habitability assessments. The document includes a user's guide to the code, a description of the technical basis for the code, and a programmer's guide to the code. The ARCON95 code uses hourly meteorological data and recently developed methods for estimating dispersion in the vicinity of buildings to calculate relative concentrations at control room air intakes that would be exceeded no more than five percent of the time. These concentrations are calculated for averaging periods ranging from one hour to 30 days in duration. Relative concentrations calculated by ARCON95 are significantly lower than concentrations calculated using the currently accepted procedure when winds are less than two meters per second. For higher wind speeds, ARCON95 calculates about the same concentrations as the current procedure.

NUREG/CR-6333: BREATH VERSION 1.1 - COUPLED FLOW AND ENERGY TRANSPORT IN POROUS MEDIA. Simulator Description And User Guide. STOTHOFF, S.A. Southwest Research Institute. July 1995. 93pp. 9509200049. CNWRA94-020. 85539:001.

This document describes the BREATH computer code, including the mathematical and numerical formulation for the simulator, usage description, and sample input files with corresponding output files. The BREATH computer code is designed to simulate one-dimensional flow of a liquid phase and dispersive transport of the corresponding vapor species, coupled with energy transfer, in a heterogeneous porous medium. The BREATH code is organized into a flow equation simulator and an energy equation simulator, which can be coupled or used independently. Both simulators use linear finite element basis functions. A modified Picard iteration scheme is used to solve the nonlinear sets of equations. Heuristic algorithms are available to control time stepping and the active solution domain. The BREATH simulator has been developed for use in auxiliary analyses which are a part of the Nuclear Regulatory Commission Iterative Performance Assessment program. The simulator was developed in response to the observation from Total System Performance Assessments by both the Nuclear Regulatory Commission and the U.S. Department of Energy that total-system performance at the Yucca Mountain site in Nevada is highly sensitive to the infiltration rate. Accordingly, this first version of the code is primarily intended to simulate processes important to infiltration and evaporation in climatic and hydrologic near-surface environments representative of the Yucca Mountain site.

NUREG/CR-6334: NEW SENSOR FOR MEASUREMENT OF LOW AIR FLOW VELOCITY. Phase I Final Report. HASHEMIAN, H.M.; HASHEMIAN, M.; RIGGSBEE, E.T. Analysis & Measurement Services Corp. August 1995. 145pp. 9508230253. 85120:001.

This is the report of a six-month feasibility study of a new sensor to measure ambient air flow velocity and direction for

health physics applications in nuclear facilities. The information from this sensor is to be used to determine where to place air samplers to sample airborne radioactive material that is representative of the air inhaled by radiation workers. A new sensor was developed in this project and successfully tested in the AMS laboratory for measurement of low flow rates of air. The sensor uses a conventional thermocouple as its sensing element and is therefore referred to as a "thermocouple flow sensor". The dynamic response of the thermocouple is measured using an in-situ response time testing method. The response time information is then converted to a flow signal using predetermined response time-versus-flow correlation for the thermocouple. The thermocouple flow sensor has the potential to aid in determining in-door air flow patterns. This may be accomplished by using multiple thermocouples to measure air flow velocities in several locations in the room and use the velocity information with computational fluid dynamics or neural network models to establish air flow patterns.

NUREG/CR-6335: FATIGUE STRAIN-LIFE BEHAVIOR OF CARBON AND LOW-ALLOY STEELS, AUSTENITIC STAINLESS STEELS, AND ALLOY 600 IN LWR ENVIRONMENTS. KEISLER, J.; CHOPRA, O.K.; SHACK, W.J. Argonne National Laboratory. August 1995. 82pp. 9508300329. ANL-95/15. 85288:221.

The existing fatigue strain vs. life (S-N) data, foreign and domestic, for carbon and low-alloy steels, austenitic stainless steels, and Alloy 600 used in the construction of nuclear power plant components have been compiled and categorized according to material, loading, and environmental conditions. Statistical models have been developed for estimating the effects of the various service conditions on the fatigue life of these materials. 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 adjusting the probability distribution curves for smooth test specimens for the effect of mean stress and applying design margins to account for the uncertainties due to component size/geometry and surface finish. The significance of the effect of environment on the current Code design curve and on the proposed interim design curves published in NUREG/CR-5999 is discussed. Estimations of the probability of fatigue cracking in sample components from BWRs and PWRs are presented.

NUREG/CR-6343: ON-LINE TESTING OF CALIBRATION OF PROCESS INSTRUMENTATION CHANNELS IN NUCLEAR POWER PLANTS. Phase II Final Report. HASHEMIAN, H.M. Analysis & Measurement Services Corp. November 1995. 342pp. 9512260334. 86625:001.

This report presents the results of a comprehensive research and development program to develop and validate a new technique for remote testing of calibration of process instrumentation channels in nuclear power plants. This technique can be used to verify the calibration of instrument channels (including the field sensors) while the plant is on-line. The method is simple and passive, uses the existing plant instrumentation, and includes the effects of process operating conditions on the calibration of the instruments. The new technique is based on monitoring the steady-state output of process instrumentation channels on a periodic or continuous basis during each fuel cycle to identify any significant drift. If the drift is more than an allowable value, then the channel is calibrated. Otherwise, the channel is not calibrated. To separate process drift from instrument drift, analytical models or a calibrated reference channel may be used to track the process independently and determine if it is drifting. On-line monitoring is performed not only during normal operation but also during plant startup and shutdown periods to obtain data to verify instrument calibrations throughout their operating range.

NUREG/CR-6347: MULTI-PHASE REACTIVE TRANSPORT THEORY. LICHTNER, P.C. Southwest Research Institute. LICHTNER, P.C. Center for Nuclear Waste Regulatory Analyses. July 1995. 102pp. 9508090044. CNWRA94-018. 84963:087.

Physicochemical processes in the near-field region of a high-level waste repository may involve a diverse set of phenomena including flow of liquid and gas, gaseous diffusion, and chemical reaction of the host rock with aqueous solutions at elevated temperatures. This report develops some of the formalism for describing simultaneous multicomponent solute and heat transport in a two-phase system for partially saturated porous media. Diffusion of gaseous species is described using the Dusty Gas Model which provides for simultaneous Knudsen and Fickian diffusion in addition to Darcy flow. A new form of the Dusty Gas Model equations is derived for binary diffusion which separates the total diffusive flux into segregative and nonsegregative components. Migration of a wetting front is analyzed using the quasi-stationary state approximation to the Richards' equation. Heat-pipe phenomena are investigated for both gravity- and capillary-driven reflux of liquid water. An expression for the burnout permeability is derived for a gravity-driven heat-pipe. Finally an estimate is given for the change in porosity and permeability due to mineral dissolution which could occur in the region of condensate formation in a heat-pipe.

NUREG/CR-6348: THERMALLY DRIVEN MOISTURE REDISTRIBUTION IN PARTIALLY SATURATED POROUS MEDIA. GREEN, R.T.; MANTEUFEL, R.D.; MEYER, K.A.; et al. Center for Nuclear Waste Regulatory Analyses. December 1995. 209pp. 9601290290. CNWRA 95-005. 86892:001.

Experimental and theoretical studies have been conducted to develop a quantitative understanding of the thermohydrologic phenomena induced by emplacement of high-level radioactive waste (HLW) in an unsaturated fractured porous media. A series of laboratory experiments was conducted in a variety of media to study the physics of thermally driven moisture redistribution through porous media. Principles of similarity theory were applied to develop dimensionless parameters to be used to predict thermohydrologic behavior at the field and repository scales from information gained from the laboratory and field scales. Results from the dimensional analyses indicated that dimensionless parameters are useful to design field-scale experiments so as to maximize the information gained from the experiment and to predict heat and mass transfer of larger scale porous media systems. Dimensional analysis did not prove to be useful in scaling systems containing discrete features such as faults or fractures. A major conclusion of this report is that conduct of a heater test at field scale is crucial to investigate and understand heat and mass transfer through partially saturated fractured, porous media at the scale of a geologic repository.

NUREG/CR-6349: COST-BENEFIT CONSIDERATIONS IN REGULATORY ANALYSIS. MUBAYI, V.; SAILOR, V. Brookhaven National Laboratory. ANANDALINGAM, G. Pennsylvania, Univ. of, Philadelphia, PA. October 1995. 148pp. 9511270458. BNL-NUREG-52466. 86317:081.

Justification for safety enhancements at nuclear facilities, e.g., a compulsory backfit to nuclear power plants, requires a value-impact analysis of the increase in overall public protection versus the cost of implementation. It has been customary to assess the benefits in terms of radiation dose to the public averted by the introduction of the safety enhancement. Comparison of such benefits with the costs of the enhancement then requires an estimate of the monetary value of averted dose (dollars/person rem). This report reviews available information on a variety of factors that affect this valuation and assesses the continuing validity of the figure of \$1000/person-rem averted, which has been widely used as a guideline in performing value-impact analyses. Factors that bear on this valuation include the health risks of radiation doses, especially the higher risk estimates of the BEIR V committee, recent calculations of doses and offsite costs by consequence codes for hypothesized

severe accidents at U.S. nuclear power plants under the NUREG-1150 program, and recent information on the economic consequences of the Chernobyl accident in the Soviet Union and estimates of risk avoidance based on the willingness-to-pay criterion. The report analyzes these factors and presents results on the dollars/person-rem ratio arising from different assumptions on the values of these factors.

NUREG/CR-6351: REVIEW OF SCENARIO SELECTION APPROACHES FOR PERFORMANCE ASSESSMENT OF HIGH-LEVEL WASTE REPOSITORIES AND RELATED ISSUES. BONANO, E.J. Southwest Research Institute. BACA, R.G. Beta Corp, International. August 1995. 78pp. 9509070092. CNWRA 94-002. 85401:230.

The selection of scenarios representing plausible realizations of the future conditions with associated probabilities of occurrence that can affect the long-term performance of a high-level radioactive waste (HLW) repository is the commonly used method for treating the uncertainty in the prediction of the future states of the system. This method, conventionally referred to as the "scenario approach," while common is not the only method to deal with this uncertainty; other methods, such as the environmental simulation approach (ESA), have also been proposed. Two of the difficulties with the scenario approach are the lack of uniqueness in the definition of the term "scenario" and the lack of uniqueness in the approach to formulate scenarios, which relies considerably on subjective judgments. Consequently, it is difficult to assure that a complete and unique set of scenarios can be defined for use in a performance assessment. Because scenarios are key to the determination of the long-term performance of the repository system, this lack of uniqueness can present a considerable challenge when attempting to reconcile the set of scenarios, and their level of detail, obtained using different approaches, particularly among proponents and regulators of a HLW repository.

NUREG/CR-6354 DRF FC: PERFORMANCE TESTING OF ELECTRONIC PERSONAL DOSIMETERS. Draft Report For Comment. SWINTH, K.L.; MCDONALD, J.C.; SISK, D.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1995. 172pp. 9508230227. PNL-10560. 85127:119.

In radiation protection, incremental control of worker radiation exposures is important to ensure that periodic dose limits are not exceeded. Electronic personal dosimeters, EPDs, are widely used for this application. As their reliability has improved users have shown an interest in their use for both incremental control and as the primary dosimeter to track the dose of record for the worker. In this application they would replace the traditional film or thermoluminescent dosimeter whose performance is thoroughly understood. The EPD brings with it some of the problems of instruments which are not seen with the traditional dosimeters. The report contains results of a survey of users and a survey of vendor literature that highlight some of the limitations and problems of EPDs. The radiation protection community is concerned that the reliability and accuracy of the data from the EPD be comparable to traditional methods if they assume this additional role. The report lists type tests, test methods and calibration methods intended to ensure the required reliability.

NUREG/CR-6355: A LIMITED ASSESSMENT OF THE ASEP HUMAN RELIABILITY ANALYSIS PROCEDURE USING SIMULATOR EXAMINATION RESULTS. GORE, B.F.; DUKELOW, J.S.; MITT, T.M.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1995. 150pp. 9511020380. PNL-10573. 86090:030.

The ASEP post-accident, post-diagnosis, nominal HRA procedure described in NUREG/CR-4772 is assessed within the context of an individual's performance of critical tasks on the simulator portion of requalification examinations administered to nuclear power plant operators. The data for this study are derived from simulator examination reports from the NRC requalification examination cycle. The ASEP procedure was used to estimate human error probability (HEP) values for the critical tasks, and

the HEP results were compared with the failure rates observed in the examinations. The ASEP procedure was applied by PNL operator license examiners who supplemented the limited information in the examination reports with expert judgment based upon their extensive simulator examination experience. Two tests were performed to assess the bias of the ASEP HEPs compared with the data from the requalification examinations. The first compared to the average of the ASEP HEP values with the fraction of the population actually failed. It found the existence of a statistically significant factor of two bias on the average. The second test partitioned the critical tasks into subgroups based on the ASEP HEP values and compared the subgroup average ASEP HEP values with the observed subgroup failure rates. It found little or no bias for small ASEP HEP values, but a larger bias for Larger ASEP HEP Values.

NUREG/CR-6356: HYDRAULIC CHARACTERIZATION OF HYDROTHERMALLY ALTERED NOPAL TUFF. GREEN, R.T.; MEYER-JAMES, K.A. Southwest Research Institute. RICE, G. George Rice & Associates. July 1995. 69pp. 9509150281. CNWRA 94-027. 85521:001.

Understanding the mechanics of variably saturated flow in fractured-porous media is of fundamental importance to evaluating the isolation performance of the proposed high-level radioactive waste repository for the Yucca Mountain site. Developing that understanding must be founded on the analysis and interpretation of laboratory and field data. This report presents an analysis of the unsaturated hydraulic properties of tuff cores from the Pena Blanca natural analog site in Mexico. The basic intent of the analysis was to examine possible trends and relationships between the hydraulic properties and the degree of hydrothermal alteration exhibited by the tuff samples. These data were used in flow simulations to evaluate the significance of a particular conceptual (composite) model and of distinct hydraulic properties on the rate and nature of water flow.

NUREG/CR-6358 V01: ASSESSMENT OF UNITED STATES INDUSTRY STRUCTURAL CODES AND STANDARDS FOR APPLICATION TO ADVANCED NUCLEAR POWER REACTORS. Final Report. ADAMS, T.M.; STEVENSON, J.D. Stevenson & Associates. October 1995. 188pp. 9510310370. 86032:154.

Through out its history, the USNRC has been committed to the use of US industry consensus standards for the design, construction, and licensing of commercial nuclear power facilities. The existing industry standards are based on the current class of light water reactors and as such may not adequately address design and construction features of the next generation of Advanced Light Water Reactors and other types of Advanced Reactors. As part of their on-going commitment to industry standards, the USNRC commissioned this study to evaluate U.S. industry structural standards for application to Advanced Reactors. The initial review effort included: (1) the review and study of the relevant reactor design basis documentation for eight Advanced Reactor Designs, (2) the review of the USNRCs design requirements for advanced reactors, (3) the review of the latest revisions of the relevant industry structural standards, and (4) the identification of the need for changes to these standards. The results of these studies were used to develop recommended changes to industry consensus structural standards which will be used in the construction of Advanced Reactors. Over seventy sets of proposed standard changes were recommended and the need for the development of four new standards was identified. In addition to the recommended standard changes, several other sets of information were extracted for use by USNRC in other programs. This information included: (1) detailed observations on the response of structures and distribution system supports to the recent Northridge, CA (1994) and Kobe, Japan (1995) earthquakes, (2) comparison of versions of certain standards cited in the standard review plan to the most current versions, and (3) comparison of the seismic and wind design basis for the subject reactor designs. Finally

provided is a suggested plan of action for implementation of the recommended industry standard changes.

NUREG/CR-6358 V02: ASSESSMENT OF UNITED STATES INDUSTRY STRUCTURAL CODES AND STANDARDS FOR APPLICATION TO ADVANCED NUCLEAR POWER REACTORS. Appendices. ADAMS, T.M.; STEVENSON, J.D. Stevenson & Associates. October 1995. 150pp. 9510310376. 86032:008.

See NUREG/CR-6358, V02 abstract.

NUREG/CR-6368: EXPERIMENTAL INVESTIGATION OF SEDIMENTATION OF LOCA-GENERATED FIBROUS DEBRIS AND SLUDGE IN BWR SUPPRESSION POOLS. SOUTO, F.J.; RAO, D.V. Science & Engineering Associates, Inc. December 1995. 82pp. 9601290176. SEA 95-554-06A9. 86881:271.

Several tests were conducted in a 1:2.4 scale model of a Mark I suppression pool to investigate the behavior of fibrous insulation and sludge debris under LOCA conditions. NUKON(TM) shreds, manually cut and torn up in a leaf shredder, and iron oxide particles were used to simulate fibrous and sludge debris, respectively. The suppression pool model included four downcomers fitted with pistons to simulate the steam-water oscillations during chugging expected during a LOCA. The study was conducted to provide debris settling velocity data for the models used in the BLOCKAGE computer code, developed to estimate the ECCS pump head loss due to clogging of the strainers with LOCA generated debris. The tests showed that the debris, both fibrous and particulate, remains fully mixed during chugging; they also showed that, during chugging, the fibrous debris underwent fragmentation into smaller sizes, including individual fibers. Measured concentrations showed that fibrous debris settled slower than the sludge, and that the settling behavior of each material is independent of the presence of the other material. Finally, these tests showed that the assumption of considering uniform debris concentration during strainer calculations is reasonable. The tests did not consider the effects of the operation of the ECCS on the transport of debris in the suppression pool.

NUREG/CR-6382: COMPARISONS OF ASTM STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN, NUREG-0800, AND RELATED DOCUMENTS. ANKRUM, A.R.; BOHLANDER, K.L.; GILBERT, E.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1995. 124pp. 9511020389. PNL-10747. 86090:163.

This report provides the results of comparisons of the cited and latest versions of ASTM standards cited in the NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (NUREG 0800) and related documents. The comparisons were performed by Battelle Pacific Northwest Laboratories in support of the NRC's Standard Review Plan Update and Development Program. Significant changes to the standards, from the cited version to the latest version, are described and discussed in a tabular format for each standard. Recommendations for updating each citation in the Standard Review Plan are presented. Technical considerations and suggested changes are included for related regulatory documents (i.e., Regulatory Guides and the Code of Federal Regulations) citing the standard. The results and recommendations presented in this document have not been subjected to NRC staff review.

NUREG/CR-6385: COMPARISONS OF ANS, ASME, AWS, AND NFPA STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN, NUREG-0800, AND RELATED DOCUMENTS. ANKRUM, A.R.; BOHLANDER, K.L.; GILBERT, E.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1995. 130pp. 9512150115. PNL-10768. 86534:126.

This report provides the results of comparisons of the cited and latest versions of ANS, ASME, AWS and NFPA standards cited in the NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (NUREG 0800) and related documents. The comparisons were per-

formed by Battelle Pacific Northwest Laboratories in support of the NRC's Standard Review Plan Update and Development Program. Significant changes to the standards, from the cited version to the latest version, are described and discussed in a tabular format for each standard. Recommendations for updating each citation in the Standard Review Plan are presented. Technical considerations and suggested changes are included for related regulatory documents (i.e., Regulatory Guides and the Code of Federal Regulations) citing the standard. The results and recommendations presented in this document have not been subjected to NRC staff review.

NUREG/CR-6386: COMPARISONS OF ANSI STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN, NUREG-0800, AND RELATED DOCUMENTS. ANKRUM, A.R.; BOHLANDER, K.L.; GILBERT, E.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1995. 126pp. 9512260328. PNL-10769. 86626:130.

This report provides the results of comparisons of the cited and latest versions of ANSI standards cited in the NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (NUREG 0800) and related documents. The comparisons were performed by Battelle Pacific Northwest Laboratories in support of the NRC's Standard Review Plan Update and Development Program. Significant changes to the standards, from the cited version to the latest version, are described and discussed in a tabular format for each standard. Recommendations for updating each citation in the Standard Review Plan are presented. Technical considerations and suggested changes are included for related regulatory documents (i.e., Regulatory Guides and the Code of Federal Regulations) citing the standard. The results and recommendations presented in this document have not been subjected to NRC staff review.

NUREG/CR-6390: RADIOLOGICAL CHARACTERIZATION OF SPENT CONTROL ROD ASSEMBLIES. LEPEL, E.A.; ROBERTSON, D.E.; THOMAS, C.W.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. October 1995. 59pp. 9511060201. PNL-10806. 86094:200.

This report describes the results of non-destructive and laboratory radionuclide measurements, as well as waste classification assessments, of BWR and PWR spent control rod assemblies. The radionuclide inventories of these spent control rods were determined by three separate methodologies, including 1) direct assay techniques, 2) calculational techniques, and 3) by sampling and laboratory radiochemical analyses. For the BWR control rod blade (CRB) and PWR burnable poison rod assembly (BPRA), (60)Co and (63)Ni, present in the stainless steel cladding, were the most abundant neutron activation products. The most abundant radionuclide in the PWR rod cluster control assembly (RCCA) was (108m)Ag (130 yr half-life) produced in the Ag-In-Cd alloy used as the neutron poison. This radionuclide will be the dominant contributor to the gamma dose rate for many hundreds of years. The results of the direct assay methods agree very well ($\pm 10\%$) with the sampling/radiochemical measurements. The results of the calculational methods agreed fairly well with the empirical measurements for the BPRA, but often varied by a factor of 5 to 10 for the CRB and the RCCA assemblies. If concentration averaging and encapsulation, as allowed by 10CFR61.55, is performed, then each of the entire control assemblies would be classified as Class C low-level radioactive waste.

NUREG/CR-6398: EVALUATION OF THE COMPUTERIZED PROCEDURES MANUAL II (COPMA-II). CONVERSE, S.A. North Carolina State Univ., Raleigh, NC. November 1995. 57pp. 9512260314. 86627:098.

The purpose of this study was to evaluate the effects of a computerized procedure system, the Computerized Procedure Manual II (COPMA-II), on the performance and mental workload of licensed reactor operators. To evaluate COPMA-II, eight

teams of two operators were trained to operate a scaled pressurized water reactor facility (SPWRF) with traditional paper procedures and with COPMA-II. Following training, each team operated the SPWRF under normal operating conditions with both paper procedures and COPMA-II. The teams then performed one of two accident scenarios with paper procedures, but performed the remaining accident scenario with COPMA-II. Performance measures and subjective estimates of mental workload were recorded for each performance trial. The most important finding of the study was that the operators committed only half as many errors during the accident scenarios with COPMA-II as they committed with paper procedures. However, time to initiate a procedure was fastest for paper procedures for accident scenario trials. For performance under normal operating conditions, there was no difference in time to initiate or to complete a procedure, or in the number of errors committed with paper procedures and with COPMA-II. There were no consistent differences in the mental workload ratings operators recorded for trials with paper procedures and COPMA-II.

NUREG/CR-6420: SELF-MONITORING SURVEILLANCE SYSTEM FOR PRESTRESSING TENDONS. TABATABAI, H. Construction Engineering Laboratory, Inc. December 1995. 73pp. 9601290169. 86879:282.

Assured safety and operational reliability of post-tensioned concrete components of nuclear power plants are of great importance to the public, electric utilities, and regulatory agencies. Prestressing tendons provide principal reinforcement for containment structures. In this phase of the research effort, the feasibility of developing a passive surveillance system for identification of ruptures in tendon wires was evaluated and verified. A one-tenth scale ring model of the Palo Verde nuclear containment structure was built inside the Structural Laboratory. Dynamic scaling (similitude) relationships were used to relate measured sensor responses (to intentional wire breaks) to the expected prototype response. Strong and recognizable signatures were detected by the accelerometers used. It is concluded that the unbonded prestressing tendons provide an excellent

path for transmission of stress waves resulting from wire breaks. Accelerometers placed on the bearing plates at the ends of tendons recorded high-intensity waveforms. However, accelerometers placed on concrete surfaces revealed substantial attenuation and reduced intensities. Locations of wire breaks were determined accurately through measurement of differences in arrival times of the signal at the two sensors. Pattern recognition systems utilized in conjunction with the proposed concept will provide a basis for an integrated and automated tool for identification of wire breaks.

NUREG/GR-0014: BALDCYPRESS TREE RING ELEMENTAL CONCENTRATIONS AT REELFOOT LAKE, TENNESSEE, FROM AD 1795 TO AD 1820. VAN ARSDALE, R. University of Memphis, Memphis, TN. HALL, G. Rutgers, New Jersey, State Univ. of, Piscataway, NJ. November 1995. 27pp. 9511290103. 86353:271.

Many 200 year old baldcypress trees in Reelfoot Lake, Tenn., lived through the great New Madrid earthquakes of 1811-1812. This study was undertaken to determine if the elemental composition of baldcypress tree rings showed any systematic variation through the earthquake period of AD 1795-1820. Cores were collected from two Reelfoot Lake baldcypress trees and analyzed using Inductively Coupled Plasma Mass Spectrometry. Individual yearly rings and five-year ring segments were analyzed to determine their elemental compositions. The cores were analyzed for Li through U but only Ba, Ce, Cs, Cu, I, La, Mg, Mn, Nd, Rb, Sm, Sr, and ZN were found to be above the detection limit. Of these only Ce, I, La, Nd, Rb, and Sm showed any systematic changes within individual cores. Comparison of three cores taken from one tree reveal that tree-ring elemental concentrations and any changes through time are very different among the cores. When comparing the elemental concentrations of tree rings for the same years in the two trees neither elemental concentrations nor changes through time were similar. The elemental concentrations show no systemic changes through the earthquake period of AD 1795-1820.

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NUREG/CR-5884 V02: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION.Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....
NUREG/CR-6054: ESTIMATING PRESSURIZED WATER REACTOR DECOMMISSIONING COSTS. A User's Manual For The PWR Cost Estimating Computer Program (CECP) Software.

BINNING,P.

NUREG/CR-6114 V02: AUXILIARY ANALYSES IN SUPPORT OF PERFORMANCE ASSESSMENT OF A HYPOTHETICAL LOW-LEVEL WASTE FACILITY.Two-Phase Flow And Contaminant Transport In Unsaturated Soils With Application To Low-Level Radioactive Waste Disposal.

BLACK,C.L.

NUREG/CR-6312: ASSESSMENT OF FIBER OPTIC PRESSURE SENSORS.

BLACK,S.C.

NUREG-1526: LESSONS LEARNED FROM EARLY IMPLEMENTATION OF THE MAINTENANCE RULE AT NINE NUCLEAR POWER PLANTS.

BLEY,D.C.

NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.

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NUREG/CP-0142 V01: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 1-5.
NUREG/CP-0142 V02: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 6-11.
NUREG/CP-0142 V03: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 12-16.
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NUREG/CR-5973 R02: CODES AND STANDARDS AND OTHER GUIDANCE CITED IN REGULATORY DOCUMENTS.
NUREG/CR-6382: COMPARISONS OF ASTM STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.
NUREG/CR-6385: COMPARISONS OF ANS,ASME,AWS, AND NFPA STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.
NUREG/CR-6386: COMPARISONS OF ANSI STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.

BONANO,E.J.

NUREG/CR-6351: REVIEW OF SCENARIO SELECTION APPROACHES FOR PERFORMANCE ASSESSMENT OF HIGH-LEVEL WASTE REPOSITORIES AND RELATED ISSUES.

BOURDON,S.M.

NUREG/CR-6150 V05: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Developmental Assessment.

BRAMWELL,D.

NUREG/CR-6100: GATE VALVE AND MOTOR-OPERATOR RESEARCH FINDINGS.

BREY,R.R.

NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990,1991,1992, And 1993.

BRICHOUX,J.

NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.

BRICKSTAD,B.

NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

BRIDEAU,J.

NUREG/CR-6224: PARAMETRIC STUDY OF THE POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS.

BROWN,E.J.

NUREG/CP-0146: PROCEEDINGS OF THE WORKSHOP ON GATE VALVE PRESSURE LOCKING AND THERMAL BINDING.

BROWN,T.D.

NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF,UNIT 1.Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage.Main Report And Appendices.
NUREG/CR-6143 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF,UNIT 1.Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage.Supporting MELCOR Calculations.

BROWN,W.S.

NUREG-0700 R01 DFC: HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE.Draft Report For Comment.

- NUREG-1507 DRFT FC: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS. Draft Report For Comment.
- BROWNSON, D.A.**
NUREG/CR-6285: SEVERE ACCIDENT NATURAL CIRCULATION STUDIES AT THE INEL.
- BRUST, F.W.**
NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES. Experiments And Analysis, March 1990 - December 1994.
NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.
NUREG/CR-6299: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE.
NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.
- BRYSON, J.W.**
NUREG/CR-6273: BIAxIAL LOADING EFFECTS ON FRACTURE TOUGHNESS OF REACTOR PRESSURE VESSEL STEEL.
- BUMPAS, S.E.**
NUREG/CR-6322: BUCKLING ANALYSIS OF SPENT FUEL BASKET.
- BURSON, S.B.**
NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS.
- CALLAN, J.R.**
NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.
NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function And Task Analysis.
NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Supporting Analyses Of Human-System Interfaces, Procedures And Practices, Training And Organizational Practices And Procedures.
- CALLEY, M.B.**
NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Verification And Validation (V&V) Manual.
- CAMPBELL, P.L.**
NUREG-1482: GUIDELINES FOR INSERVICE TESTING AT NUCLEAR POWER PLANTS.
- CAMPBELL, V.**
NUREG-1516 DRFT FC: MANAGEMENT OF RADIOACTIVE MATERIAL SAFETY PROGRAMS AT MEDICAL FACILITIES. Draft Report For Comment.
- CAMPER, L.W.**
NUREG-1516 DRFT FC: MANAGEMENT OF RADIOACTIVE MATERIAL SAFETY PROGRAMS AT MEDICAL FACILITIES. Draft Report For Comment.
- CAPLIN, J.**
NUREG/CR-6307: SUMMARY OF COMMENTS RECEIVED AT WORKSHOP ON USE OF A SITE SPECIFIC ADVISORY BOARD (SSAB) TO FACILITATE PUBLIC PARTICIPATION IN DECOMMISSIONING CASES.
- CAPPS, E.L.**
NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Verification And Validation (V&V) Manual.
- CASADA, D.A.**
NUREG/CR-6089: DETECTION OF PUMP DEGRADATION.
- CELIA, M.A.**
NUREG/CR-6114 V02: AUXILIARY ANALYSES IN SUPPORT OF PERFORMANCE ASSESSMENT OF A HYPOTHETICAL LOW-LEVEL WASTE FACILITY. Two-Phase Flow And Contaminant Transport In Unsaturated Soils With Application To Low-Level Radioactive Waste Disposal.
- CHAMBERS, R.**
NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.
- CHARAK, I.**
NUREG/CR-6315: CANDU REACTORS, THEIR REGULATION IN CANADA, AND THE IDENTIFICATION OF RELEVANT NRC SAFETY ISSUES.
- CHEN, G.**
NUREG/CR-6308: AN OVERVIEW OF INSTABILITY AND FINGERING DURING IMMISCIBLE FLUID FLOW IN POROUS AND FRACTURED MEDIA.
- CHEN, P.**
NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Final Report.
NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Draft Report For Comment.
- CHEN, T.F.**
NUREG/CR-6242: CASKS (COMPUTER ANALYSIS OF STORAGE CASKS): A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR STORAGE CASK DESIGN REVIEW. User's Manual To Version 1b (Including Program Reference).
- CHENG, T.C.**
NUREG/CR-6150 V02: SCDAP/RELAP/MOD 3.1 CODE MANUAL. Damage Progression Model Theory.
- CHINE, E.**
NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.
NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function And Task Analysis.
NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Supporting Analyses Of Human-System Interfaces, Procedures And Practices, Training And Organizational Practices And Procedures.
- CHOI, Y.H.**
NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.
NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.
- CHOPRA, O.K.**
NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, October 1993 - March 1994.
NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, April-September 1994.
NUREG/CR-6275: MECHANICAL PROPERTIES OF THERMALLY AGED CAST STAINLESS STEELS FROM SHIPPINGPORT REACTOR COMPONENTS.
NUREG/CR-6335: FATIGUE STRAIN-LIFE BEHAVIOR OF CARBON AND LOW-ALLOY STEELS, AUSTENITIC STAINLESS STEELS, AND ALLOY 600 IN LWR ENVIRONMENTS.
- CHOWDHURY, A.H.**
NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.
- CHU, T.L.**
NUREG/CR-6144 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Summary Of Results.
- CHUNG, H.M.**
NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, October 1993 - March 1994.
NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, April-September 1994.
- CLARK, R.E.**
NUREG/CR-6116 V06: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Graphical Evaluation Module (GEM) Reference Manual.

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CLEAVES,D.

NUREG/CR-6263 V01: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Executive Summary.
NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Main Report.

CLETCHER,J.W.

NUREG/CR-4674 V21: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Main Report And Appendices A-H.
NUREG/CR-4674 V22: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Appendix I.

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NUREG-1525: ASSESSMENT OF THE NRC ENFORCEMENT PROGRAM.

COHEN,H.

NUREG/CR-6263 V01: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Executive Summary.
NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Main Report.

COLE,R.K.

NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS.Primer And User's Guides. Version 1.8.3 September 1994.
NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS.Reference Manuals.Version 1.8.3 September 1994.

COLLINS,J.L.

NUREG/CR-6318: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-7.

CONGER,R.

NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.

CONLEY,D.A.

NUREG/CR-5462: AGING STUDY OF BOILING WATER REACTOR HIGH PRESSURE INJECTION SYSTEMS.

CONVERSE,S.A.

NUREG/CR-6398: EVALUATION OF THE COMPUTERIZED PROCEDURES MANUAL II (COPMA-II).

COOKE,R.M.

NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Main Report.
NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Appendices A And B.
NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment.Appendices C,D,E,F,G,H.

COOPER,S.E.

NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.

COPINGER,D.A.

NUREG/CR-4674 V21: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Main Report And Appendices A-H.
NUREG/CR-4674 V22: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Appendix I.

CORREIA,R.P.

NUREG-1526: LESSONS LEARNED FROM EARLY IMPLEMENTATION OF THE MAINTENANCE RULE AT NINE NUCLEAR POWER PLANTS.

CORWIN,W.R.

NUREG/CR-5591 V03: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Progress Report For October 1991 - September 1992.
NUREG/CR-5591 V04 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For April-September 1993.
NUREG/CR-5591 V05 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For September 1993 Through March 1994.

NUREG/CR-5591 V05 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Progress Report For April 1994 Through September 1994.
NUREG/CR-5591 V06 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For October 1994 Through March 1995.

CORYELL,E.W.

NUREG/CR-6150 V01: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Interface Theory.
NUREG/CR-6150 V02: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Damage Progression Model Theory.
NUREG/CR-6150 V03: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.User's Guide And Input Manual.
NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.
NUREG/CR-6150 V05: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Developmental Assessment.

COX,D.F.

NUREG/CR-5857: AGING OF TURBINE DRIVES FOR SAFETY-RELATED PUMPS IN NUCLEAR POWER PLANTS.
NUREG/CR-6016: AGING AND SERVICE WEAR OF AIR-OPERATED VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.
NUREG/CR-6192: AGING AND SERVICE WEAR OF SPRING-LOADED PRESSURE RELIEF VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.

CUNNINGHAM,M.

NUREG/CR-5758 V05: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports CY 1994.

CYBULSKIS,P.

NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM.Final Report.
NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM.Draft Report For Comment.

DAVIS,E.C.

NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Program.Test Results For Fiscal Years 1986, 1987,1988, And 1989.

DAVIS,K.L.

NUREG/CR-6150 V01: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Interface Theory.
NUREG/CR-6150 V02: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Damage Progression Model Theory.
NUREG/CR-6150 V03: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.User's Guide And Input Manual.
NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.
NUREG/CR-6150 V05: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Developmental Assessment.

DELICATE,W.S.

NUREG/CR-6287: MANAGEMENT CONCEPTS AND SAFETY APPLICATIONS FOR NUCLEAR FUEL FACILITIES.

DEWALL,K.G.

NUREG/CR-6100: GATE VALVE AND MOTOR-OPERATOR RESEARCH FINDINGS.

DEY,M.

NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM.Final Report.
NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM.Draft Report For Comment.

DINSMORE,G.

NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS.Main Report.
NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS.Appendices A-D.

DOBBE,C.A.

NUREG/CR-6285: SEVERE ACCIDENT NATURAL CIRCULATION STUDIES AT THE INEL.

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NUREG/CR-6191: SIZE AND DEFORMATION LIMITS TO MAINTAIN CONSTRAINT IN K(IC) AND J(C) TESTING OF BEND SPECIMENS.

- NUREG/CR-6259: CONSTRAINT EFFECTS ON FRACTURE INITIATION LOADS IN HSST WIDE-PLATE TESTS.
- DODGE,F.T.**
NUREG/CR-6348: THERMALLY DRIVEN MOISTURE REDISTRIBUTION IN PARTIALLY SATURATED POROUS MEDIA.
- DOLAN,B.W.**
NUREG/CR-4674 V21: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Main Report And Appendices A-H.
NUREG/CR-4674 V22: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Appendix I.
- DOTY,K.**
NUREG/CR-2907 V13: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1992.
NUREG/CR-2907 V14: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS.
- DUKELOW,J.S.**
NUREG/CR-6355: A LIMITED ASSESSMENT OF THE ASEP HUMAN RELIABILITY ANALYSIS PROCEDURE USING SIMULATOR EXAMINATION RESULTS.
- DURBIN,N.**
NUREG/CR-5758 V05: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports CY 1994.
- EASON,E.D.**
NUREG/CR-6327: MODELS FOR EMBRITTLEMENT RECOVERY DUE TO ANNEALING OF REACTOR PRESSURE VESSEL STEELS.
- EASTERLY,J.C.**
NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Supporting Analyses Of Human-System Interfaces,Procedures And Practices,Training And Organizational Practices And Procedures.
- EBERT,D.D.**
NUREG/CR-6257: CANDU 3 TRANSIENT ANALYSIS USING ATOMIC ENERGY OF CANADA LTD CODES.
- EDSON,J.L.**
NUREG/CR-5462: AGING STUDY OF BOILING WATER REACTOR HIGH PRESSURE INJECTION SYSTEMS.
- EISENBERG,N.A.**
NUREG-1464: NRC ITERATIVE PERFORMANCE ASSESSMENT PHASE 2.Development Of Capabilities For Review Of A Performance Assessment For A High-Level Waste Repository.
- EISNER,H.**
NUREG/CR-6287: MANAGEMENT CONCEPTS AND SAFETY APPLICATIONS FOR NUCLEAR FUEL FACILITIES.
- EMRIT,R.**
NUREG-0933 S18: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-0933 S19: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
- ERCK,R.A.**
NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1993 - March 1994.
- FADDEN,M.**
NUREG-0525 V02 R03: SAFEGUARDS SUMMARY EVENT LIST (SSEL).January 1, 1990 Through December 31, 1994.
- FARMER,J.P.**
NUREG/CR-6312: ASSESSMENT OF FIBER OPTIC PRESSURE SENSORS.
- FAUVER,D.N.**
NUREG-1444 S01: SITE DECOMMISSIONING MANAGEMENT PLAN.
- FEINER,F.**
NUREG/CP-0142 V01: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 1-5.
NUREG/CP-0142 V02: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 6-11.
NUREG/CP-0142 V03: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 12-16.
- NUREG/CP-0142 V04: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 17-24.
- FERRELL,C.M.**
NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS.
- FIELD,I.**
NUREG/CR-5758 V05: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports CY 1994.
- FINDLAY,M.W.**
NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Program.Test Results For Fiscal Years 1986, 1987,1988, And 1989.
- FINEMAN,C.F.**
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- FIRST,M.W.**
NUREG/CP-0141: PROCEEDINGS OF THE 23RD DOE/NRC NUCLEAR AIR CLEANING CONFERENCE.Held In Buffalo,New York,July 25-28,1994.
- FISCHER,L.E.**
NUREG/CR-6323: RELATIVE RISK ANALYSIS IN REGULATING THE USE OF RADIATION-EMITTING MEDICAL DEVICES.A Preliminary Application.
NUREG/CR-6324: QUALITY ASSURANCE FOR GAMMA KNIVES.
- FLETCHER,C.D.**
NUREG/CR-5535 V05 R1: RELAP5/MOD3 CODE MANUAL.User's Guideline.
- FORESTER,J.**
NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF,UNIT 1.Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage.Main Report And Appendices.
- FOWLER,R.D.**
NUREG/CR-6116 V10: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Data Loading Manual.
- FOX,P.B.**
NUREG/CR-6284: CRITICALITY SAFETY CRITERIA FOR LICENSE REVIEW OF LOW-LEVEL WASTE FACILITIES.
- FRANCINI,B.**
NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES.Experiments And Analysis,March 1990 - December 1994.
- FRANCINI,R.**
NUREG/CR-6297: FRACTURE EVALUATIONS OF FUSION LINE CRACKS IN NUCLEAR PIPE BIMETALLIC WELDS.
NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.
- FREDRICKSON,P.**
NUREG-1528: RECONSTITUTION OF THE MANUAL CHAPTER 2512 CONSTRUCTION INSPECTION PROGRAM FOR WATTS BAR UNIT 1.
- FULLER,M.**
NUREG-1516 DRFT FC: MANAGEMENT OF RADIOACTIVE MATERIAL SAFETY PROGRAMS AT MEDICAL FACILITIES.Draft Report For Comment.
- GALYEAN,W.J.**
NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Verification And Validation (V&V) Manual.
- GANT,K.S.**
NUREG-1514: GUIDANCE FOR A LARGE TABLETOP EXERCISE FOR A NUCLEAR POWER PLANT.

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GAVENDA,D.J.

NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, April-September 1994.

GERHARD,M.A.

NUREG/CR-5657: AUTOCASK (AUTOMATIC GENERATION OF 3-D CASK MODELS). A Microcomputer Based System For Shipping Cask Design Review Analysis.

NUREG/CR-6242: CASKS (COMPUTER ANALYSIS OF STORAGE CASKS): A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR STORAGE CASK DESIGN REVIEW. User's Manual To Version 1b (Including Program Reference).

GHADIALI,N.

NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING WELDS. Seventh Program Report March 1993 - December 1994.

NUREG/CR-6004: PROBABILISTIC PIPE FRACTURE EVALUATIONS FOR LEAK-RATE-DETECTION APPLICATIONS.

NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES. Experiments And Analysis, March 1990 - December 1994.

NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.

NUREG/CR-6299: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE.

NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

GILBERT,E.R.

NUREG/CR-6382: COMPARISONS OF ASTM STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN, NUREG-0800, AND RELATED DOCUMENTS.

NUREG/CR-6385: COMPARISONS OF ANS, ASME, AWS, AND NFPA STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN, NUREG-0800, AND RELATED DOCUMENTS.

NUREG/CR-6386: COMPARISONS OF ANSI STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN, NUREG-0800, AND RELATED DOCUMENTS.

GOGOLAK,C.V.

NUREG-1505 DRFT FC: A NONPARAMETRIC STATISTICAL METHODOLOGY FOR THE DESIGN AND ANALYSIS OF FINAL STATUS DECOMMISSIONING SURVEYS. Draft Report For Comment.

GOLDIN,D.

NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Final Report.

NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Draft Report For Comment.

GOODENOW,T.C.

NUREG/CR-6313 V01: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEM. Summary of Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.

NUREG/CR-6313 V02: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEMS. Supplemental Appendices Presenting Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.

GOOSSENS,L.H.J.

NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Main Report.

NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Appendices A And B.

NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment. Appendices C,D,E,F,G,H.

GORE,B.F.

NUREG/CR-6355: A LIMITED ASSESSMENT OF THE ASEP HUMAN RELIABILITY ANALYSIS PROCEDURE USING SIMULATOR EXAMINATION RESULTS.

GREEN,R.T.

NUREG/CR-6348: THERMALLY DRIVEN MOISTURE REDISTRIBUTION IN PARTIALLY SATURATED POROUS MEDIA.

NUREG/CR-6356: HYDRAULIC CHARACTERIZATION OF HYDROTHERMALLY ALTERED NOPAL TUFF.

GREENE,N.M.

NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/

PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.

GREENE,R.H.

NUREG/CR-6089: DETECTION OF PUMP DEGRADATION.

GRIFFIN,J.I.

NUREG-1527: NRC'S OBJECT-ORIENTED SIMULATOR INSTRUCTOR STATION.

GRIFFIN,J.P.

NUREG-1527: NRC'S OBJECT-ORIENTED SIMULATOR INSTRUCTOR STATION.

GRODIN,M.

NUREG-1415 V07 N02: OFFICE OF THE INSPECTOR GENERAL. Semiannual Report To Congress, October 1, 1994 - March 31, 1995.

NUREG-1415 V08 N01: OFFICE OF THE INSPECTOR GENERAL. Semiannual Report To Congress, April 1, 1995 - September 30, 1995.

GROSSMAN,N.

NUREG-1519: SURFACE INTERACTIONS OF CESIUM AND BORIC ACID WITH STAINLESS STEEL.

GROUNDWATER,E.H.

NUREG/CR-6316 V02: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Assessment Of Conventional Software Verification And Validation Methods.

NUREG/CR-6316 V03: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Documentation Of Expert System Verification And Validation Methodologies.

GROVE,E.J.

NUREG/CR-5954: EFFECT ON AGING ON PWR CHEMICAL AND VOLUME CONTROL SYSTEM.

GRUBER,E.E.

NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, April-September 1994.

GRUSH,W.H.

NUREG/CR-6291 V01: NUCLEAR PLANT ANALYZER. Installation Manual.

NUREG/CR-6291 V02: NUCLEAR PLANT ANALYZER. Analyzer Reference Manual.

NUREG/CR-6291 V03: NUCLEAR PLANT ANALYZER. Computer Visual System Reference Manual.

NUREG/CR-6291 V04: NUCLEAR PLANT ANALYZER. Programmer's Manual.

GWYNNE,J.W.

NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.

NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function And Task Analysis.

HAGEMEYER,D.

NUREG-0713 V15: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES, 1993. Twenty-Sixth Annual Report.

HAGGARD,D.I.

NUREG/CR-6390: RADIOLOGICAL CHARACTERIZATION OF SPENT CONTROL ROD ASSEMBLIES.

HAGRMAN,D.L.

NUREG/CR-6150 V02: SCDAP/RELAP/MOD 3.1 CODE MANUAL. Damage Progression Model Theory.

NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.

HAGRMAN,D.T.

NUREG/CR-6150 V01: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. Interface Theory.

NUREG/CR-6150 V02: SCDAP/RELAP/MOD 3.1 CODE MANUAL. Damage Progression Model Theory.

NUREG/CR-6150 V03: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. User's Guide And Input Manual.

- NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.
 NUREG/CR-6150 V05: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Developmental Assessment.
- HALL,G.**
 NUREG/GR-0014: BALDCYPRESS TREE RING ELEMENTAL CONCENTRATIONS AT REELFOOT LAKE,TENNESSEE,FROM AD 1795 TO AD 1820.
- HAMILTON,M.A.**
 NUREG/CR-6188 V02: MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE.Annual Report For FY 1994.
- HAMPTON,N.L.**
 NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.
- HARPER,F.T.**
 NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Main Report.
 NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Appendices A And B.
 NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment.Appendices C,D,E,F,G,H.
- HARTFIELD,R.A.**
 NUREG-0020 V19: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of December 31, 1994.(Gray Book I)
- HASHEMIAN,H.M.**
 NUREG/CR-6312: ASSESSMENT OF FIBER OPTIC PRESSURE SENSORS.
 NUREG/CR-6334: NEW SENSOR FOR MEASUREMENT OF LOW AIR FLOW VELOCITY.Phase I Final Report.
 NUREG/CR-6343: ON-LINE TESTING OF CALIBRATION OF PROCESS INSTRUMENTATION CHANNELS IN NUCLEAR POWER PLANTS.Phase II Final Report.
- HASHEMIAN,M.**
 NUREG/CR-6334: NEW SENSOR FOR MEASUREMENT OF LOW AIR FLOW VELOCITY.Phase I Final Report.
- HATTRUP,M.**
 NUREG/CR-5758 V05: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports CY 1994.
 NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.
- HAYES,J.E.**
 NUREG/CR-6316 V01: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.
 NUREG/CR-6316 V02: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Survey And Assessment Of Conventional Software Verification And Validation Methods.
 NUREG/CR-6316 V04: GUIDELINES FOR THE VERIFICATION AND VALIDATION EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Evaluation Of Knowledge Base Certification Methods.
 NUREG/CR-6316 V05: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Rationale And Description Of V&V Guideline Packages And Procedures.
 NUREG/CR-6316 V06: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Validation Scenarios.
 NUREG/CR-6316 V07: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.User's Manual.
 NUREG/CR-6316 V08: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Bibliography.
- HECHT,H.**
 NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS.Main Report.
 NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS.Appendices A-D.
- HECHT,M.**
 NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS.Main Report.
 NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS.Appendices A-D.
- HECHT,S.**
 NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS.Main Report.
 NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS.Appendices A-D.
- HELD,P.R.**
 NUREG/CR-6251: STAINLESS STEEL SUBMERGED ARC WELD FUSION LINE TOUGHNESS.
- HELTON,J.C.**
 NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Main Report.
 NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Appendices A And B.
 NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment.Appendices C,D,E,F,G,H.
- HENDERSON,P.**
 NUREG-1516 DRFT FC: MANAGEMENT OF RADIOACTIVE MATERIAL SAFETY PROGRAMS AT MEDICAL FACILITIES.Draft Report For Comment.
- HENRIKSEN,K.**
 NUREG/CR-6277 V01: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Identification Of Problems And Alternative Approaches.
 NUREG/CR-6277 V02: HUMAN FACTORS EVALUATION OF TELE-THERAPY.Function And Task Analysis.
 NUREG/CR-6277 V03: HUMAN FACTORS EVALUATION OF TELE-THERAPY.Human-System Interfaces And Procedures.
 NUREG/CR-6277 V04: HUMAN FACTORS EVALUATION OF TELE-THERAPY.Training And Organizational Analysis.
 NUREG/CR-6277 V05: HUMAN FACTORS EVALUATION OF TELE-THERAPY.Literature Review.
- HILTON,L.D.**
 NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Program.Test Results For Fiscal Years 1986, 1987,1988, And 1989.
 NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990,1991,1992, And 1993.
- HINS,A.G.**
 NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April-September 1994.
- HODGE,S.A.**
 NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS.Primer And User's Guides. Version 1.8.3 September 1994.
 NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS.Reference Manuals.Version 1.8.3 September 1994.
- HOFFMAN,C.L.**
 NUREG/CR-6116 V06: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Graphical Evaluation Module (GEM) Reference Manual.
- HOHORST,J.K.**
 NUREG/CR-6150 V01: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Interface Theory.
 NUREG/CR-6150 V02: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Damage Progression Model Theory.
 NUREG/CR-6150 V03: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.User's Guide And Input Manual.

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- NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE
MANUAL.MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.
- NUREG/CR-6150 V05: SCDAP/RELAP5/MOD 3.1 CODE
MANUAL.Developmental Assessment.
- HOLLAND,H.M.**
NUREG/CR-6313 V01: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEM.Summary of Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.
- NUREG/CR-6313 V02: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEMS.Supplemental Appendices Presenting Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.
- HOPPER,C.M.**
NUREG/CR-6284: CRITICALITY SAFETY CRITERIA FOR LICENSE REVIEW OF LOW-LEVEL WASTE FACILITIES.
- HORA,S.C.**
NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Main Report.
- NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Appendices A And B.
- NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment.Appendices C,D,E,F,G,H.
- HSIUNG,S-M.**
NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.
- HUBER,D.**
NUREG-1415 V07 N02: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report To Congress, October 1, 1994 - March 31, 1995.
- NUREG-1415 V08 N01: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report To Congress, April 1, 1995 - September 30, 1995.
- HUFFERT,A.M.**
NUREG-1505 DRFT FC: A NONPARAMETRIC STATISTICAL METHODOLOGY FOR THE DESIGN AND ANALYSIS OF FINAL STATUS DECOMMISSIONING SURVEYS.Draft Report For Comment.
- NUREG-1506 DRFT FC: MEASUREMENT METHODS FOR RADIOLOGICAL SURVEYS IN SUPPORT OF NEW DECOMMISSIONING CRITERIA.Draft Report For Comment.
- NUREG-1507 DRFT FC: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS.Draft Report For Comment.
- HUGHES,K.**
NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.
- HUGHES,T.H.**
NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, April-September 1994.
- HUNTER,T.H.**
NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.
- HUTCHINSON,J.W.**
NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION.A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions.
- HYBERTSON,D.**
NUREG/CR-6263 V01: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Executive Summary.
- NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Main Report.
- HYMAN,C.R.**
NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS.Primer And User's Guides. Version 1.8.3 September 1994.
- NUREG/CR-6119 V02: MELCOR COMPUTER CODE
MANUALS.Reference Manuals.Version 1.8.3 September 1994.
- IDDINGS,F.A.**
NUREG/CR-6074 V04: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING.Technical Report On The Findings Of Task 4.Investigation Of Sealed Source for Paper Mill Digester.
- INGERSOLL,D.T.**
NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.
- JASTROW,J.D.**
NUREG/CR-5229 V07: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1994.Annual Report.
- NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Program.Test Results For Fiscal Years 1986, 1987, 1988, And 1989.
- NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990, 1991, 1992, And 1993.
- JAUDON,J.**
NUREG-1528: RECONSTITUTION OF THE MANUAL CHAPTER 2512 CONSTRUCTION INSPECTION PROGRAM FOR WATTS BAR UNIT 1.
- JEDD,J.L.**
NUREG/CR-6257: CANDU 3 TRANSIENT ANALYSIS USING ATOMIC ENERGY OF CANADA LTD CODES.
- JO,J.**
NUREG/CR-6144 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1.Evaluation Of Severe Accident Risk During Mid-Loop Operations.Main Report.
- NUREG/CR-6144 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1.Evaluation Of Severe Accident Risk During Mid-Loop Operations.Appendices.
- JOHNSEN,E.C.**
NUREG/CR-6150 V03: SCDAP/RELAP5/MOD 3.1 CODE
MANUAL.User's Guide And Input Manual.
- NUREG/CR-6150 V05: SCDAP/RELAP5/MOD 3.1 CODE
MANUAL.Developmental Assessment.
- JOHNSON,G.L.**
NUREG/CR-6242: CASKS (COMPUTER ANALYSIS OF STORAGE CASKS): A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR STORAGE CASK DESIGN REVIEW.User's Manual To Version 1b (Including Program Reference).
- JOHNSON,J.**
NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1.Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage.Main Report And Appendices.
- JOHNSON,J.C.**
NUREG/CR-6114 V02: AUXILIARY ANALYSES IN SUPPORT OF PERFORMANCE ASSESSMENT OF A HYPOTHETICAL LOW-LEVEL WASTE FACILITY.Two-Phase Flow And Contaminant Transport In Unsaturated Soils With Application To Low-Level Radioactive Waste Disposal.
- JOHNSON,J.D.**
NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
- NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
- NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
- JOHNSON,T.C.**
NUREG-1444 S01: SITE DECOMMISSIONING MANAGEMENT PLAN.

JONES,E.D.

NUREG/CR-6323: RELATIVE RISK ANALYSIS IN REGULATING THE USE OF RADIATION-EMITTING MEDICAL DEVICES.A Preliminary Application.
NUREG/CR-6324: QUALITY ASSURANCE FOR GAMMA KNIVES.

JONES,J.

NUREG-1516 DRFT FC: MANAGEMENT OF RADIOACTIVE MATERIAL SAFETY PROGRAMS AT MEDICAL FACILITIES.Draft Report For Comment.

JONES,J.A.

NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Main Report.
NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Appendices A And B.
NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment.Appendices C,D,E,F,G,H.

JONES,J.L.

NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Verification And Validation (V&V) Manual.

JONES,K.R.

NUREG/CR-6285: SEVERE ACCIDENT NATURAL CIRCULATION STUDIES AT THE INEL.
NUREG/CR-6291 V01: NUCLEAR PLANT ANALYZER.Installation Manual.
NUREG/CR-6291 V02: NUCLEAR PLANT ANALYZER.Analyzer Reference Manual.
NUREG/CR-6291 V03: NUCLEAR PLANT ANALYZER.Computer Visual System Reference Manual.
NUREG/CR-6291 V04: NUCLEAR PLANT ANALYZER.Programmer's Manual.

JONES,R.

NUREG/CR-6277 V01: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Identification Of Problems And Alternative Approaches.
NUREG/CR-6277 V02: HUMAN FACTORS EVALUATION OF TELE-THERAPY.Function And Task Analysis.
NUREG/CR-6277 V03: HUMAN FACTORS EVALUATION OF TELE-THERAPY.Human-System Interfaces And Procedures.
NUREG/CR-6277 V04: HUMAN FACTORS EVALUATION OF TELE-THERAPY.Training And Organizational Analysis.
NUREG/CR-6277 V05: HUMAN FACTORS EVALUATION OF TELE-THERAPY.Literature Review.

JORDAN,W.C.

NUREG/CR-6328: ADEQUACY OF THE 123-GROUP CROSS-SECTION LIBRARY FOR CRITICALITY ANALYSES OF WATER-MODERATED URANIUM SYSTEMS.

JOY,D.R.

NUREG-0430 V14: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.July 1, 1993 - June 30, 1994.(Gray Book II)
NUREG-1065 R02: ACCEPTABLE STANDARD FORMAT AND CONTENT FOR THE FUNDAMENTAL NUCLEAR MATERIAL CONTROL (FNMC) PLAN REQUIRED FOR LOW-ENRICHED URANIUM FACILITIES.

KASSNER,T.F.

NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1993 - March 1994.
NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April-September 1994.

KASUMOVIC,J.

NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.
NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Supporting Analyses Of Human-System Interfaces,Procedures And Practices,Training And Organizational Practices And Procedures.

KAYE,R.D.

NUREG/CR-6277 V01: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Identification Of Problems And Alternative Approaches.

NUREG/CR-6277 V02: HUMAN FACTORS EVALUATION OF TELE-THERAPY.Function And Task Analysis.
NUREG/CR-6277 V03: HUMAN FACTORS EVALUATION OF TELE-THERAPY.Human-System Interfaces And Procedures.
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KEISLER,J.

NUREG/CR-6335: FATIGUE STRAIN-LIFE BEHAVIOR OF CARBON AND LOW-ALLOY STEELS, AUSTENITIC STAINLESS STEELS, AND ALLOY 600 IN LWR ENVIRONMENTS.

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KHAN,T.A.

NUREG/CP-0143: PROCEEDINGS OF THE THIRD INTERNATIONAL WORKSHOP ON THE IMPLEMENTATION OF ALARA AT NUCLEAR POWER PLANTS.Held At Hauppauge, Long Island, New York.
NUREG/CR-3469 V08: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.

KIER,P.H.

NUREG/CR-6315: CANDU REACTORS, THEIR REGULATION IN CANADA, AND THE IDENTIFICATION OF RELEVANT NRC SAFETY ISSUES.

KILINSKI,T.

NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES.Experiments And Analysis,March 1990 - December 1994.
NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.

KIM,I.S.

NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.

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NUREG-1444 S01: SITE DECOMMISSIONING MANAGEMENT PLAN.

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NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.

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NUREG-1517: REPORT OF THE SOUTH TEXAS PROJECT ALLEGATIONS REVIEW TEAM.

KONZEK,G.J.

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NUREG/CR-6310: AN ANALYSIS OF POTASSIUM IODIDE (KI) PROPHYLAXIS FOR THE GENERAL PUBLIC IN THE EVENT OF A NUCLEAR ACCIDENT.

KOVACH, L.A.

NUREG/CP-0147: PROCEEDINGS OF THE WORKSHOP ON THE ROLE OF NATURAL ANALOGS IN GEOLOGIC DISPOSAL OF HIGH-LEVEL NUCLEAR WASTE. Held In San Antonio, Texas, July 22-25, 1991.

KOZAK, M.W.

NUREG/CR-5927 V02: EVALUATION OF A PERFORMANCE ASSESSMENT METHODOLOGY FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES. Validation Needs.

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NUREG/CR-6276: A COMPILATION OF CURRENT REGULATIONS, STANDARDS, AND GUIDELINES IN REMOTE AFTERLOADING BRACHYTHERAPY.

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NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Main Report.

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KRISHNASWAMY, P.

NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING WELDS. Seventh Program Report March 1993 - December 1994.

NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES. Experiments And Analysis, March 1990 - December 1994.

NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.

NUREG/CR-6299: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE.

NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

KVARFORDT, K.J.

NUREG/CR-6116 V06: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Graphical Evaluation Module (GEM) Reference Manual.

LACH, D.

NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.

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NUREG/CR-6154 V02: EXPERIMENTAL RESULTS FROM CONTAINMENT PIPING BELLWS SUBJECTED TO SEVERE ACCIDENT CONDITIONS. Results From Bellows Tested In Corroded Conditions.

NUREG/CR-6184: SEPARATE EFFECTS TESTING AND ANALYSES TO INVESTIGATE LINER TEARING OF THE 1:16-SCALE REINFORCED CONCRETE CONTAINMENT BUILDING.

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LAWLESS, M.T.

NUREG/CR-6159: USING MICRO SAINT TO PREDICT PERFORMANCE IN A NUCLEAR POWER PLANT CONTROL ROOM. A Test Of Validity And Feasibility.

LAWRENCE, J.D.

NUREG/CP-0145: WORKSHOP ON DEVELOPING SAFE SOFTWARE. Held At Hotel Del Coronado, San Diego, CA, July 22-23, 1992.

LEE, A.S.

NUREG/CR-6322: BUCKLING ANALYSIS OF SPENT FUEL BASKET.

LEE, M.P.

NUREG-1464: NRC ITERATIVE PERFORMANCE ASSESSMENT PHASE 2. Development Of Capabilities For Review Of A Performance Assessment For A High-Level Waste Repository.

LEE, R.Y.

NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS.

LEPAGE, R.P.

NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.

NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function And Task Analysis.

NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Supporting Analyses Of Human-System Interfaces, Procedures And Practices, Training And Organizational Practices And Procedures.

LEPEL, E.A.

NUREG/CR-6390: RADIOLOGICAL CHARACTERIZATION OF SPENT CONTROL ROD ASSEMBLIES.

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NUREG/CR-6347: MULTI-PHASE REACTIVE TRANSPORT THEORY.

LIEBERMAN, J.

NUREG-1525: ASSESSMENT OF THE NRC ENFORCEMENT PROGRAM.

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NUREG/CR-6144 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Evaluation Of Severe Accident Risk During Mid-Loop Operations. Main Report.

NUREG/CR-6144 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Evaluation Of Severe Accident Risk During Mid-Loop Operations. Appendices.

LIU, X.H.

NUREG/CR-6264 V01: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. An Assessment Of The J(M) Parameter.

LORENZ, R.A.

NUREG/CR-6261: A SUMMARY OF ORNL FISSION PRODUCT RELEASE TESTS WITH RECOMMENDED RELEASE RATES AND DIFFUSION COEFFICIENTS.

NUREG/CR-6318: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-7.

LUCADAMO, K.

NUREG/CR-2907 V13: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1992.

NUREG/CR-2907 V14: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS.

LUCKAS, W.J.

NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.

LUI, C.H.

NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Main Report.

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MACFARLANE, R.E.

NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.

MACKINNON, R.J.

NUREG/CR-6305: BLT-EC (BREACH, LEACH, TRANSPORT, AND EQUILIBRIUM CHEMISTRY), A FINITE-ELEMENT MODEL FOR ASSESS-

ING THE RELEASE OF RADIONUCLIDES FROM LOW-LEVEL WASTE DISPOSAL UNITS. Background, Theory, And Model Description.

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NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.

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NUREG/CR-6348: THERMALLY DRIVEN MOISTURE REDISTRIBUTION IN PARTIALLY SATURATED POROUS MEDIA.

MARSHALL, C.

NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.

MARSHALL, C.W.

NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES. Experiments And Analysis, March 1990 - December 1994.

NUREG/CR-6299: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE.

MARTINEZ-GURIDI

NUREG/CR-6172: REVIEWING PSA-BASED ANALYSES TO MODIFY TECHNICAL SPECIFICATIONS AT NUCLEAR POWER PLANTS.

MASON, R.E.

NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.

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NUREG/CR-6273: BIAxIAL LOADING EFFECTS ON FRACTURE TOUGHNESS OF REACTOR PRESSURE VESSEL STEEL.

MCCARTIN, T.J.

NUREG-1464: NRC ITERATIVE PERFORMANCE ASSESSMENT PHASE 2. Development Of Capabilities For Review Of A Performance Assessment For A High-Level Waste Repository.

MCCOMAS, M.L.

NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.

MCCONNELL, J.W.

NUREG/CR-5229 V07: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1994. Annual Report.

NUREG/CR-6188 V02: MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE. Annual Report For FY 1994.

NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Development Program. Test Results For Fiscal Years 1986, 1987, 1988, And 1989.

NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990, 1991, 1992, And 1993.

MCDONALD, J.C.

NUREG/CR-6354 DRF FC: PERFORMANCE TESTING OF ELECTRONIC PERSONAL DOSIMETERS. Draft Report For Comment.

MCDUFFIE, P.N.

NUREG/CR-5884 V01: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....

NUREG/CR-5884 V02: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....

MCELHANEY, K.L.

NUREG/CR-5944 V02: A CHARACTERIZATION OF CHECK VALVE DEGRADATION AND FAILURE EXPERIENCE IN THE NUCLEAR POWER INDUSTRY. 1991 Failures.

NUREG/CR-6016: AGING AND SERVICE WEAR OF AIR-OPERATED VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.

MCKAY, M.D.

NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Main Report.

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NUREG/CR-6311: EVALUATING PREDICTION UNCERTAINTY.

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NUREG/CR-5975 R01: INCENTIVE REGULATION OF INVESTOR-OWNED NUCLEAR POWER PLANTS BY PUBLIC UTILITY REGULATORS.

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NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.

MEINHOLD, C.B.

NUREG/CR-6112: IMPACT OF REDUCED DOSE LIMITS ON NRC LICENSED ACTIVITIES. Major Issues In The Implementation Of ICRP/ NCRP Dose Limit Recommendations. Final Report.

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NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.

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NUREG/CR-6356: HYDRAULIC CHARACTERIZATION OF HYDROTHERMALLY ALTERED NOPAL TUFF.

MICHAUD, W.F.

NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, October 1993 - March 1994.

MILLER, K.M.

NUREG-1506 DRFT FC: MEASUREMENT METHODS FOR RADIOLOGICAL SURVEYS IN SUPPORT OF NEW DECOMMISSIONING CRITERIA. Draft Report For Comment.

MILLER, L.A.

NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Main Report And Appendices.

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NUREG/CR-6316 V04: GUIDELINES FOR THE VERIFICATION AND VALIDATION EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Evaluation Of Knowledge Base Certification Methods.
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 NUREG/CR-6316 V08: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Bibliography.

MILLER,R.L.

NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.MATPRO-A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.

MINARICK,J.W.

NUREG/CR-4674 V21: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Main Report And Appendices A-H.
 NUREG/CR-4674 V22: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Appendix I.

MINTON,L.

NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM.Final Report.
 NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM.Draft Report For Comment.

MIRSKY,S.M.

NUREG/CR-6316 V01: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.
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 NUREG/CR-6316 V08: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Bibliography.

MITTS,T.M.

NUREG/CR-6355: A LIMITED ASSESSMENT OF THE ASEP HUMAN RELIABILITY ANALYSIS PROCEDURE USING SIMULATOR EXAMINATION RESULTS.

MIURA,N.

NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES.Experiments And Analysis,March 1990 - December 1994.

MOHAN,R.

NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.
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MOK,G.C.

NUREG/CR-6242: CASKS (COMPUTER ANALYSIS OF STORAGE CASKS): A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR STORAGE CASK DESIGN REVIEW.User's Manual To Version 1b (Including Program Reference).

MONTELEONE,S.

NUREG/CP-0140 V01: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING.Plenary Session, Advanced Instrumentation & Control Hardware & Software, Human Factors Research, IPE & PRA.
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 NUREG/CP-0140 V03: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING.Primary Systems Integrity, Structural And Seismic Engineering, Aging Research, Products And Applications.
 NUREG/CP-0148: TRANSACTIONS OF THE TWENTY-THIRD WATER REACTOR SAFETY INFORMATION MEETING.

MONTGOMERY,J.

NUREG-1516 DRFT FC: MANAGEMENT OF RADIOACTIVE MATERIAL SAFETY PROGRAMS AT MEDICAL FACILITIES.Draft Report For Comment.

MOORE,R.A.

NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.
 NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Supporting Analyses Of Human-System Interfaces,Procedures And Practices,Training And Organizational Practices And Procedures.

MORISSEAU,D.S.

NUREG/CR-6277 V01: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Identification Of Problems And Alternative Approaches.
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MORTON,D.K.

NUREG/CR-6260: APPLICATION OF NUREG/CR-5999 INTERIM FATIGUE CURVES TO SELECTED NUCLEAR POWER PLANT COMPONENTS.

MUBAYI,V.

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MURPHY,D.D.

NUREG-1517: REPORT OF THE SOUTH TEXAS PROJECT ALLEGATIONS REVIEW TEAM.

MURPHY,W.M.

NUREG/CP-0147: PROCEEDINGS OF THE WORKSHOP ON THE ROLE OF NATURAL ANALOGS IN GEOLOGIC DISPOSAL OF HIGH-LEVEL NUCLEAR WASTE.Held In San Antonio,Texas,July 22-25, 1991.

NEILSON,R.M.

NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Program.Test Results For Fiscal Years 1986, 1987,1988, And 1989.
NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990,1991,1992, And 1993.

NELSON,E.E.

NUREG/CR-6327: MODELS FOR EMBRITTLEMENT RECOVERY DUE TO ANNEALING OF REACTOR PRESSURE VESSEL STEELS.

NEUMAN,S.P.

NUREG/CR-6308: AN OVERVIEW OF INSTABILITY AND FINGERING DURING IMMISCIBLE FLUID FLOW IN POROUS AND FRACTURED MEDIA.

NEYMOTIN,L.

NUREG/CR-6144 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1.Evaluation Of Severe Accident Risk During Mid-Loop Operations.Main Report.
NUREG/CR-6144 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1.Evaluation Of Severe Accident Risk During Mid-Loop Operations.Appendices.

NICHOLSON,W.L.

NUREG/CR-6355: A LIMITED ASSESSMENT OF THE ASEP HUMAN RELIABILITY ANALYSIS PROCEDURE USING SIMULATOR EXAMINATION RESULTS.

NICKOLAUS,J.R.

NUREG/CR-5973 R02: CODES AND STANDARDS AND OTHER GUIDANCE CITED IN REGULATORY DOCUMENTS.

NICOLETTE,V.F.

NUREG/CR-6017: FIRE MODELING OF THE HEISS DAMPF REAKTOR CONTAINMENT.

NITZEL,M.E.

NUREG/CR-6260: APPLICATION OF NUREG/CR-5999 INTERIM FATIGUE CURVES TO SELECTED NUCLEAR POWER PLANT COMPONENTS.

NORTON,L.

NUREG-1415 V07 N02: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report To Congress, October 1, 1994 - March 31, 1995.

NOVACK,S.D.

NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Verification And Validation (V&V) Manual.

NOWLEN,S.P.

NUREG/CR-6173: A SUMMARY OF THE FIRE TESTING PROGRAM AT THE GERMAN HDR TEST FACILITY.
NUREG/CR-6220: AN ASSESSMENT OF FIRE VULNERABILITY FOR AGED ELECTRICAL RELAYS.

O'BRIEN,J.E.

NUREG/CR-6285: SEVERE ACCIDENT NATURAL CIRCULATION STUDIES AT THE INEL.

O'DONNELL,E.

NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Progress Report Of Field Experiments At A Humid Region Site,Beltsville,Maryland.

O'HARA,J.M.

NUREG-0700 R01 DFC: HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE.Draft Report For Comment.

ODEGAARDEN,R.H.

NUREG/CR-6284: CRITICALITY SAFETY CRITERIA FOR LICENSE REVIEW OF LOW-LEVEL WASTE FACILITIES.

ODETTE,G.R.

NUREG/CR-6327: MODELS FOR EMBRITTLEMENT RECOVERY DUE TO ANNEALING OF REACTOR PRESSURE VESSEL STEELS.

OFEGBU,G.I.

NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.

OLAGUE,N.E.

NUREG/CR-5927 V02: EVALUATION OF A PERFORMANCE ASSESSMENT METHODOLOGY FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES.Validation Needs.

OLSEN,C.S.

NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.

ORNSTEIN,H.L.

NUREG-1275 V11: OPERATING EXPERIENCE FEEDBACK REPORT - TURBINE-GENERATOR OVERSPEED PROTECTION SYSTEM.Commercial Power Reactors.

OSBORNE,M.F.

NUREG/CR-6261: A SUMMARY OF ORNL FISSION PRODUCT RELEASE TESTS WITH RECOMMENDED RELEASE RATES AND DIFFUSION COEFFICIENTS.
NUREG/CR-6318: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-7.

PAFFORD,D.J.

NUREG/CR-6285: SEVERE ACCIDENT NATURAL CIRCULATION STUDIES AT THE INEL.

PAGE,G.

NUREG/CR-6307: SUMMARY OF COMMENTS RECEIVED AT WORKSHOP ON USE OF A SITE SPECIFIC ADVISORY BOARD (SSAB) TO FACILITATE PUBLIC PARTICIPATION IN DECOMMISSIONING CASES.

PAIK,S.

NUREG/CR-6150 V02: SCDAP/RELAP/MOD 3.1 CODE MANUAL.Damage Progression Model Theory.

PARKS,C.V.

NUREG/CR-0200 V1 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION.Control Modules.
NUREG/CR-0200 V2P1R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION.Functional Modules F1-F8.
NUREG/CR-0200 V2P2R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION.Functional Modules F9-F16.
NUREG/CR-0200 V3 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Miscellaneous.
NUREG/CR-6284: CRITICALITY SAFETY CRITERIA FOR LICENSE REVIEW OF LOW-LEVEL WASTE FACILITIES.
NUREG/CR-6328: ADEQUACY OF THE 123-GROUP CROSS-SECTION LIBRARY FOR CRITICALITY ANALYSES OF WATER-MODERATED URANIUM SYSTEMS.

PARKS,M.B.

NUREG/CR-6154 V02: EXPERIMENTAL RESULTS FROM CONTAINMENT PIPING BELLOWS SUBJECTED TO SEVERE ACCIDENT CONDITIONS.Results From Bellows Tested In Corroded Conditions.

PASLER-SAUER,J.

NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Main Report.
NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Appendices A And B.
NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment.Appendices C,D,E,F,G,H.

PAUL,D.

NUREG/CR-6004: PROBABILISTIC PIPE FRACTURE EVALUATIONS FOR LEAK-RATE-DETECTION APPLICATIONS.

PAULSEN,M.P.

NUREG/CR-6325: AN IMPLICIT STEADY-STATE INITIALIZATION PACKAGE FOR THE RELAP5 COMPUTER CODE.

PAWLOWSKI,R.A.

NUREG/CR-6382: COMPARISONS OF ASTM STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.

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NUREG/CR-6386: COMPARISONS OF ANSI STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.

PENNELL,W.E.

NUREG/CR-4219 V10 N2: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For April-September 1993.
NUREG/CR-4219 V11 N1: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For October 1993 - March 1994.

NUREG/CR-6273: BIAxIAL LOADING EFFECTS ON FRACTURE TOUGHNESS OF REACTOR PRESSURE VESSEL STEEL.

PERANICH,M.

NUREG-1528: RECONSTITUTION OF THE MANUAL CHAPTER 2512 CONSTRUCTION INSPECTION PROGRAM FOR WATTS BAR UNIT 1.

PERSENSKY,J.J.

NUREG-0700 R01 DFC: HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE.Draft Report For Comment.
NUREG/CR-6159: USING MICRO SAINT TO PREDICT PERFORMANCE IN A NUCLEAR POWER PLANT CONTROL ROOM.A Test Of Validity And Feasibility.

PETERSON,C.E.

NUREG/CR-6325: AN IMPLICIT STEADY-STATE INITIALIZATION PACKAGE FOR THE RELAP5 COMPUTER CODE.

PETRONE,C.D.

NUREG-1526: LESSONS LEARNED FROM EARLY IMPLEMENTATION OF THE MAINTENANCE RULE AT NINE NUCLEAR POWER PLANTS.

PHILIP,J.

NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.

PILCH,M.M.

NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.

PIPER,R.K.

NUREG/CR-6354 DRF FC: PERFORMANCE TESTING OF ELECTRONIC PERSONAL DOSIMETERS.Draft Report For Comment.

POWERS,G.E.

NUREG-1505 DRFT FC: A NONPARAMETRIC STATISTICAL METHODOLOGY FOR THE DESIGN AND ANALYSIS OF FINAL STATUS DE-COMMISSIONING SURVEYS.Draft Report For Comment.

PRATT,S.L.

NUREG/CR-6390: RADIOLOGICAL CHARACTERIZATION OF SPENT CONTROL ROD ASSEMBLIES.

PRATT,W.T.

NUREG/CR-6144 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY,UNIT 1.Summary Of Results.

QUICK,K.S.

NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.

QUINN,M.L.

NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.
NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Supporting Analyses Of Human-System Interfaces,Procedures And Practices,Training And Organizational Practices And Procedures.

RADDATZ,C.T.

NUREG-0713 V15: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES,1993.Twenty-Sixth Annual Report.

RAHMAN,S.

NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING WELDS.Seventh Program Report March 1993 - December 1994.
NUREG/CR-6004: PROBABILISTIC PIPE FRACTURE EVALUATIONS FOR LEAK-RATE-DETECTION APPLICATIONS.
NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES.Experiments And Analysis,March 1990 - December 1994.

NUREG/CR-6297: FRACTURE EVALUATIONS OF FUSION LINE CRACKS IN NUCLEAR PIPE BIMETALLIC WELDS.

NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.

NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

RAMEY-SMITH,A.

NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.

RAMSDELL,J.V.

NUREG/CR-6331: ATMOSPHERIC RELATIVE CONCENTRATIONS IN BUILDING WAKES.

RAO,D.V.

NUREG/CR-6224: PARAMETRIC STUDY OF THE POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS.

NUREG/CR-6368: EXPERIMENTAL INVESTIGATION OF SEDIMENTATION OF LOCA-GENERATED FIBROUS DEBRIS AND SLUDGE IN BWR SUPPRESSION POOLS.

REYMANN,G.A.

NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.

REZOS,J.T.

NUREG/CR-6002: RISK-BASED MAINTENANCE MODELING.Prioritization Of Maintenance Importances And Quantification Of Maintenance Effectiveness.

RICE,G.

NUREG/CR-6348: THERMALLY DRIVEN MOISTURE REDISTRIBUTION IN PARTIALLY SATURATED POROUS MEDIA.

NUREG/CR-6356: HYDRAULIC CHARACTERIZATION OF HYDROTHERMALLY ALTERED NOPAL TUFF.

RIDGELY,J.N.

NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS.

RIDKY,R.W.

NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Progress Report Of Field Experiments At A Humid Region Site,Beltsville,Maryland.

RIGGSBEE,E.T.

NUREG/CR-6334: NEW SENSOR FOR MEASUREMENT OF LOW AIR FLOW VELOCITY.Phase I Final Report.

ROBERTSON,D.E.

NUREG/CR-6390: RADIOLOGICAL CHARACTERIZATION OF SPENT CONTROL ROD ASSEMBLIES.

RODABAUGH,E.C.

NUREG/CR-3243: COMPARISON OF ASME CODE FATIGUE EVALUATION METHODS FOR NUCLEAR CLASS 1 PIPING WITH CLASS 2 OR 3 PIPING.

ROGERS,R.D.

NUREG/CR-5229 V07: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1994.Annual Report.

NUREG/CR-6188 V02: MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE.Annual Report For FY 1994.

NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Program.Test Results For Fiscal Years 1986, 1987,1988, And 1989.

NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990,1991,1992, And 1993.

ROLLSTIN,J.A.

NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

ROMAN,W.

NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM.Final Report.
NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM.Draft Report For Comment.

ROSE,S.

NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM.Final Report.
NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM.Draft Report For Comment.

ROSENFELD,A.R.

NUREG/CR-6251: STAINLESS STEEL SUBMERGED ARC WELD FUSION LINE TOUGHNESS.
NUREG/CR-6297: FRACTURE EVALUATIONS OF FUSION LINE CRACKS IN NUCLEAR PIPE BIMETALLIC WELDS.

ROUSSIN,R.W.

NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.

RUDLAND,D.

NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING WELDS.Seventh Program Report March 1993 - December 1994.

RUSSELL,K.D.

NUREG/CR-6116 V06: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Graphical Evaluation Module (GEM) Reference Manual.

RUSSELL,M.J.

NUREG/CR-6100: GATE VALVE AND MOTOR-OPERATOR RESEARCH FINDINGS.

RUTHER,W.E.

NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1993 - March 1994.
NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April-September 1994.

SAILOR,V.

NUREG/CR-6349: COST-BENEFIT CONSIDERATIONS IN REGULATORY ANALYSIS.

SAMANTA,P.K.

NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.
NUREG/CR-6172: REVIEWING PSA-BASED ANALYSES TO MODIFY TECHNICAL SPECIFICATIONS AT NUCLEAR POWER PLANTS.

SANDERS,R.L.

NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS.Primer And User's Guides. Version 1.8.3 September 1994.
NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS.Reference Manuals.Version 1.8.3 September 1994.

SANECKI,J.E.

NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1993 - March 1994.

SANFORD,W.E.

NUREG/CR-5229 V07: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1994.Annual Report.
NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990,1991,1992, And 1993.

SATTISON,M.B.

NUREG/CR-6116 V06: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Graphical Evaluation Module (GEM) Reference Manual.

SAUNDERS,W.M.

NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.
NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Function And Task Analysis.

NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Supporting Analyses Of Human-System Interfaces,Procedures And Practices,Training And Organizational Practices And Procedures.

SCHAEFER,C.

NUREG/CR-6263 V01: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Executive Summary.
NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Main Report.

SCHICK,W.R.

NUREG/CR-6074 V05: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING.Technical Report On The Findings Of Task 4.Investigation Of Failed Radioactive Stainless Steel Troxler Gauges.

SCHLENKER,L.D.

NUREG/CR-6285: SEVERE ACCIDENT NATURAL CIRCULATION STUDIES AT THE INEL.

SCHLUETER,J.

NUREG-1516 DRFT FC: MANAGEMENT OF RADIOACTIVE MATERIAL SAFETY PROGRAMS AT MEDICAL FACILITIES.Draft Report For Comment.

SCHOENFELD,I.

NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.
NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Function And Task Analysis.
NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Supporting Analyses Of Human-System Interfaces,Procedures And Practices,Training And Organizational Practices And Procedures.

SCHULTZ,R.R.

NUREG/CR-5535 V05 R1: RELAP5/MOD3 CODE MANUAL.User's Guideline.

SCHULZ,R.K.

NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Progress Report Of Field Experiments At A Humid Region Site,Beltsville,Maryland.

SCOTT,P.

NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING WELDS.Seventh Program Report March 1993 - December 1994.
NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES.Experiments And Analysis,March 1990 - December 1994.
NUREG/CR-6297: FRACTURE EVALUATIONS OF FUSION LINE CRACKS IN NUCLEAR PIPE BIMETALLIC WELDS.
NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.

SCOTTI,R.S.

NUREG/CR-6287: MANAGEMENT CONCEPTS AND SAFETY APPLICATIONS FOR NUCLEAR FUEL FACILITIES.

SEELY,H.E.

NUREG/CR-5975 R01: INCENTIVE REGULATION OF INVESTOR-OWNED NUCLEAR POWER PLANTS BY PUBLIC UTILITY REGULATORS.

SERIG,D.I.

NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.
NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Function And Task Analysis.
NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Supporting Analyses Of Human-System Interfaces,Procedures And Practices,Training And Organizational Practices And Procedures.
NUREG/CR-6277 V01: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Identification Of Problems And Alternative Approaches.
NUREG/CR-6277 V02: HUMAN FACTORS EVALUATION OF TELE-THERAPY.Function And Task Analysis.

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 NUREG/CR-6277 V04: HUMAN FACTORS EVALUATION OF TELETHERAPY.Training And Organizational Analysis.
 NUREG/CR-6277 V05: HUMAN FACTORS EVALUATION OF TELETHERAPY.Literature Review.
- SETH,S.**
 NUREG/CR-6263 V01: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Executive Summary.
 NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Main Report.
- SHA,W.T.**
 NUREG/CR-6266: ANALYSIS OF BORON DILUTION IN A FOUR-LOOP PWR.
- SHACK,W.J.**
 NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1993 - March 1994.
 NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April-September 1994.
 NUREG/CR-6275: MECHANICAL PROPERTIES OF THERMALLY AGED CAST STAINLESS STEELS FROM SHIPPINGPORT REACTOR COMPONENTS.
 NUREG/CR-6335: FATIGUE STRAIN-LIFE BEHAVIOR OF CARBON AND LOW-ALLOY STEELS, AUSTENITIC STAINLESS STEELS, AND ALLOY 600 IN LWR ENVIRONMENTS.
- SHAFFER,C.**
 NUREG/CR-6224: PARAMETRIC STUDY OF THE POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS.
- SHIEH,A.S.**
 NUREG/CR-6150 V02: SCDAP/RELAP/MOD 3.1 CODE MANUAL.Damage Progression Model Theory.
- SHIH,C.F.**
 NUREG/CR-6264 V01: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION.An Assessment Of The J(M) Parameter.
 NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION.A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions.
- SHIPMAN,R.L.**
 NUREG/CR-6313 V01: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEM.Summary of Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.
 NUREG/CR-6313 V02: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEMS.Supplemental Appendices Presenting Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.
- SHIVER,A.W.**
 NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
- SHUMWAY,R.W.**
 NUREG/CR-6257: CANDU 3 TRANSIENT ANALYSIS USING ATOMIC ENERGY OF CANADA LTD CODES.
- SIEFKEN,L.J.**
 NUREG/CR-6150 V01: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Interface Theory.
 NUREG/CR-6150 V02: SCDAP/RELAP/MOD 3.1 CODE MANUAL.Damage Progression Model Theory.
 NUREG/CR-6150 V03: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.User's Guide And Input Manual.
 NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.
 NUREG/CR-6150 V05: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Developmental Assessment.
- SIMION,G.P.**
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- SIMONEN,C.A.**
 NUREG/CR-6331: ATMOSPHERIC RELATIVE CONCENTRATIONS IN BUILDING WAKES.
- SIMONSON,S.A.**
 NUREG/CR-6305: BLT-EC (BREACH, LEACH, TRANSPORT, AND EQUILIBRIUM CHEMISTRY), A FINITE-ELEMENT MODEL FOR ASSESSING THE RELEASE OF RADIONUCLIDES FROM LOW-LEVEL WASTE DISPOSAL UNITS.Background, Theory, And Model Description.
- SISK,D.R.**
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- SKAY,D.M.**
 NUREG-1517: REPORT OF THE SOUTH TEXAS PROJECT ALLEGATIONS REVIEW TEAM.
- SKOBLAR,L.**
 NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM.Final Report.
 NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM.Draft Report For Comment.
- SLATER,C.O.**
 NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.
- SMITH,C.L.**
 NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Verification And Validation (V&V) Manual.
 NUREG/CR-6116 V10: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Data Loading Manual.
- SMITH,D.**
 NUREG/CR-6307: SUMMARY OF COMMENTS RECEIVED AT WORKSHOP ON USE OF A SITE SPECIFIC ADVISORY BOARD (SSAB) TO FACILITATE PUBLIC PARTICIPATION IN DECOMMISSIONING CASES.
- SMITH,R.C.**
 NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS.Primer And User's Guides. Version 1.8.3 September 1994.
 NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS.Reference Manuals.Version 1.8.3 September 1994.
- SMITH,R.I.**
 NUREG/CR-5884 V01: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION.Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....
 NUREG/CR-5884 V02: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION.Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....
- SMYTH,S.B.**
 NUREG/CR-6331: ATMOSPHERIC RELATIVE CONCENTRATIONS IN BUILDING WAKES.
- SNIDER,D.M.**
 NUREG/CR-6291 V01: NUCLEAR PLANT ANALYZER.Installation Manual.
 NUREG/CR-6291 V02: NUCLEAR PLANT ANALYZER.Analyzer Reference Manual.
 NUREG/CR-6291 V03: NUCLEAR PLANT ANALYZER.Computer Visual System Reference Manual.
 NUREG/CR-6291 V04: NUCLEAR PLANT ANALYZER.Programmer's Manual.
- SOFFER,L.**
 NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS.

SOMMER,S.C.

NUREG/CR-5657: AUTOCASK (AUTOMATIC GENERATION OF 3-D CASK MODELS).A Microcomputer Based System For Shipping Cask Design Review Analysis.

SOPPET,W.K.

NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1993 - March 1994.

NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April-September 1994.

SOUTO,F.J.

NUREG/CR-6224: PARAMETRIC STUDY OF THE POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS.

NUREG/CR-6368: EXPERIMENTAL INVESTIGATION OF SEDIMENTATION OF LOCA-GENERATED FIBROUS DEBRIS AND SLUDGE IN BWR SUPPRESSION POOLS.

SPENCER,B.W.

NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.

SPIESMAN,J.B.

NUREG/CR-6382: COMPARISONS OF ASTM STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.

NUREG/CR-6385: COMPARISONS OF ANS,ASME,AWS, AND NFPA STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.

NUREG/CR-6386: COMPARISONS OF ANSI STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.

SPLETZER,B.L.

NUREG/CR-6184: SEPARATE EFFECTS TESTING AND ANALYSES TO INVESTIGATE LINER TEARING OF THE 1:16-SCALE REINFORCED CONCRETE CONTAINMENT BUILDING.

SPRUNG,J.L.

NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

STADLER,L.

NUREG-1350 V07: NUCLEAR REGULATORY COMMISSION INFORMATION DIGEST.1995 Edition.

STAMPS,D.W.

NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.

STAUNTON,R.H.

NUREG/CR-6016: AGING AND SERVICE WEAR OF AIR-OPERATED VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.

NUREG/CR-6192: AGING AND SERVICE WEAR OF SPRING-LOADED PRESSURE RELIEF VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.

STEELE,R.

NUREG/CR-6100: GATE VALVE AND MOTOR-OPERATOR RESEARCH FINDINGS.

STEVENSON,J.D.

NUREG/CR-6239 V01: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS.Main Report.

NUREG/CR-6239 V02: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Appendices.

NUREG/CR-6240: APPLICATION OF BOUNDING SPECTRA TO SEISMIC DESIGN OF PIPING BASED ON THE PERFORMANCE OF ABOVE GROUND PIPING IN POWER PLANTS SUBJECTED TO STRONG MOTION EARTHQUAKES.

NUREG/CR-6358 V01: ASSESSMENT OF UNITED STATES INDUSTRY STRUCTURAL CODES AND STANDARDS FOR APPLICATION TO ADVANCED NUCLEAR POWER REACTORS.Final Report.

NUREG/CR-6358 V02: ASSESSMENT OF UNITED STATES INDUSTRY STRUCTURAL CODES AND STANDARDS FOR APPLICATION TO ADVANCED NUCLEAR POWER REACTORS.Appendices.

STOTHOFF,S.A.

NUREG/CR-6333: BREATH VERSION 1.1 - COUPLED FLOW AND ENERGY TRANSPORT IN POROUS MEDIA.Simulator Description And User Guide.

STRUCKMEYER,R.

NUREG-0837 V14 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. October-December 1994.

NUREG-0837 V15 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. January-March 1995.

NUREG-0837 V15 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report.April-June 1995.

NUREG-0837 V15 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. July-September 1995.

STUART,D.S.

NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS.Primer And User's Guides. Version 1.8.3 September 1994.

NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS.Reference Manuals.Version 1.8.3 September 1994.

STUART,G.

NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Main Report.

STUBLER,W.F.

NUREG-0700 R01 DFC: HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE.Draft Report For Comment.

SUEN,C.J.

NUREG/CR-6305: BLT-EC (BREACH, LEACH, TRANSPORT, AND EQUILIBRIUM CHEMISTRY), A FINITE-ELEMENT MODEL FOR ASSESSING THE RELEASE OF RADIONUCLIDES FROM LOW-LEVEL WASTE DISPOSAL UNITS.Background, Theory, And Model Description.

SULLIVAN,S.G.

NUREG/CR-3469 V08: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.

SULLIVAN,T.M.

NUREG/CR-5229 V07: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1994.Annual Report.

NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990,1991,1992, And 1993.

NUREG/CR-6305: BLT-EC (BREACH, LEACH, TRANSPORT, AND EQUILIBRIUM CHEMISTRY), A FINITE-ELEMENT MODEL FOR ASSESSING THE RELEASE OF RADIONUCLIDES FROM LOW-LEVEL WASTE DISPOSAL UNITS.Background, Theory, And Model Description.

SUMMERS,R.M.

NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS.Primer And User's Guides. Version 1.8.3 September 1994.

NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS.Reference Manuals.Version 1.8.3 September 1994.

SUN,J.G.

NUREG/CR-6266: ANALYSIS OF BORON DILUTION IN A FOUR-LOOP PWR.

SVEDEMAN,S.J.

NUREG/CR-6348: THERMALLY DRIVEN MOISTURE REDISTRIBUTION IN PARTIALLY SATURATED POROUS MEDIA.

SWINTH,K.L.

NUREG/CR-6354 DRF FC: PERFORMANCE TESTING OF ELECTRONIC PERSONAL DOSIMETERS.Draft Report For Comment.

TA,A.

NUREG/CR-6263 V01: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Executive Summary.

NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Main Report.

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TABATABAI,H.

NUREG/CR-6420: SELF-MONITORING SURVEILLANCE SYSTEM FOR PRESTRESSING TENDONS.

TADIOS,E.L.

NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.

TAM,P.S.

NUREG-0847 S15: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)
NUREG-0847 S16: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)
NUREG-0847 S17: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)
NUREG-0847 S18: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)
NUREG-0847 S19: 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)

TANG,D.

NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Main Report.
NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Appendices A-D.

TANIGUCHI,M.

NUREG/CR-6308: AN OVERVIEW OF INSTABILITY AND FINGERING DURING IMMISCIBLE FLUID FLOW IN POROUS AND FRACTURED MEDIA.

THOMAS,C.W.

NUREG/CR-6390: RADIOLOGICAL CHARACTERIZATION OF SPENT CONTROL ROD ASSEMBLIES.

THOMAS,W.

NUREG/CR-6224: PARAMETRIC STUDY OF THE POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS.

THOMPSON,M.G.

NUREG/CR-6354 DRF FC: PERFORMANCE TESTING OF ELECTRONIC PERSONAL DOSIMETERS. Draft Report For Comment.

THOMPSON,S.L.

NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS. Primer And User's Guides. Version 1.8.3 September 1994.
NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS. Reference Manuals. Version 1.8.3 September 1994.

TICHLER,J.

NUREG/CR-2907 V13: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1992.
NUREG/CR-2907 V14: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS.

TORTORELLI,J.P.

NUREG/CP-0144 V01: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Summary.
NUREG/CP-0144 V02: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Supporting Documents.
NUREG/CR-6276: A COMPILATION OF CURRENT REGULATIONS, STANDARDS, AND GUIDELINES IN REMOTE AFTERLOADING BRACHYTHERAPY.

TRAVIS,J.R.

NUREG/CR-6318: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-7.

TRAVIS,R.J.

NUREG/CR-5954: EFFECT ON AGING ON PWR CHEMICAL AND VOLUME CONTROL SYSTEM.

TRUMMER,D.J.

NUREG/CR-6242: CASKS (COMPUTER ANALYSIS OF STORAGE CASKS): A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR STORAGE CASK DESIGN REVIEW. User's Manual To Version 1b (Including Program Reference).

TUNG,V.X.

NUREG/CR-6285: SEVERE ACCIDENT NATURAL CIRCULATION STUDIES AT THE INEL.

ULERY,B.

NUREG/CR-6263 V01: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS. Candidate Guidelines, Technical Basis And Research Needs. Executive Summary.
NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS. Candidate Guidelines, Technical Basis And Research Needs. Main Report.

VAN ARSDALE,R.

NUREG/GR-0014: BALDCYPRESS TREE RING ELEMENTAL CONCENTRATIONS AT REELFOOT LAKE, TENNESSEE, FROM AD 1795 TO AD 1820.

VANDEN HEUVEL

NUREG/CR-4674 V21: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1994 A STATUS REPORT. Main Report And Appendices A-H.
NUREG/CR-4674 V22: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1994 A STATUS REPORT. Appendix I.

VANHORN,R.L.

NUREG/CR-6116 V10: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Data Loading Manual.

VEEH,R.H.

NUREG/CR-6188 V02: MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE. Annual Report For FY 1994.

VESELY,W.E.

NUREG/CR-6002: RISK-BASED MAINTENANCE MODELING. Prioritization Of Maintenance Importances And Quantification Of Maintenance Effectiveness.
NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.
NUREG/CR-6172: REVIEWING PSA-BASED ANALYSES TO MODIFY TECHNICAL SPECIFICATIONS AT NUCLEAR POWER PLANTS.

VIGIL,R.A.

NUREG/CR-6220: AN ASSESSMENT OF FIRE VULNERABILITY FOR AGED ELECTRICAL RELAYS.

WACHTEL,J.A.

NUREG-0700 R01 DFC: HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE. Draft Report For Comment.

WAGNER,K.L.

NUREG/CR-6291 V01: NUCLEAR PLANT ANALYZER. Installation Manual.
NUREG/CR-6291 V02: NUCLEAR PLANT ANALYZER. Analyzer Reference Manual.
NUREG/CR-6291 V03: NUCLEAR PLANT ANALYZER. Computer Visual System Reference Manual.
NUREG/CR-6291 V04: NUCLEAR PLANT ANALYZER. Programmer's Manual.

WANG,H.B.

NUREG-1517: REPORT OF THE SOUTH TEXAS PROJECT ALLEGATIONS REVIEW TEAM.

WARE,A.G.

NUREG/CR-6260: APPLICATION OF NUREG/CR-5999 INTERIM FATIGUE CURVES TO SELECTED NUCLEAR POWER PLANT COMPONENTS.

WATKINS,B.

NUREG-1415 V08 N01: OFFICE OF THE INSPECTOR GENERAL. Semiannual Report To Congress, April 1, 1995 - September 30, 1995.

WATKINS,J.C.

NUREG/CR-6100: GATE VALVE AND MOTOR-OPERATOR RESEARCH FINDINGS.

WEATHERBY,J.R.

NUREG/CR-6184: SEPARATE EFFECTS TESTING AND ANALYSES TO INVESTIGATE LINER TEARING OF THE 1:16-SCALE REINFORCED CONCRETE CONTAINMENT BUILDING.

WEBER,M.F.

NUREG-1444 S01: SITE DECOMMISSIONING MANAGEMENT PLAN.

WEBSTER,C.S.

NUREG/CR-6318: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-7.

WEINSTEIN,E.D.

NUREG-1514: GUIDANCE FOR A LARGE TABLETOP EXERCISE FOR A NUCLEAR POWER PLANT.

WESCOTT,R.G.

NUREG-1464: NRC ITERATIVE PERFORMANCE ASSESSMENT PHASE 2.Development Of Capabilities For Review Of A Performance Assessment For A High-Level Waste Repository.

WESTRA,C.

NUREG/CR-5758 V05: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports CY 1994.

WHITE,J.E.

NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.

WHITEHEAD,D.W.

NUREG/CR-6143 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF,UNIT 1.Summary Of Results.

NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF,UNIT 1.Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage.Main Report And Appendices.

WIBLIN,C.

NUREG/CR-6307: SUMMARY OF COMMENTS RECEIVED AT WORKSHOP ON USE OF A SITE SPECIFIC ADVISORY BOARD (SSAB) TO FACILITATE PUBLIC PARTICIPATION IN DECOMMISSIONING CASES.

WICKLIFF HICKS

NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990,1991,1992, And 1993.

WILKOWSKI,G.M.

NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING WELDS.Seventh Program Report March 1993 - December 1994.

NUREG/CR-6004: PROBABILISTIC PIPE FRACTURE EVALUATIONS FOR LEAK-RATE-DETECTION APPLICATIONS.

NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES.Experiments And Analysis,March 1990 - December 1994.

NUREG/CR-6251: STAINLESS STEEL SUBMERGED ARC WELD FUSION LINE TOUGHNESS.

NUREG/CR-6297: FRACTURE EVALUATIONS OF FUSION LINE CRACKS IN NUCLEAR PIPE BIMETALLIC WELDS.

NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.

NUREG/CR-6299: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE.

NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

WILSON,R.

NUREG/CR-5758 V05: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports CY 1994.

WOLFRAM,L.M.

NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Verification And Validation (V&V) Manual.

NUREG/CR-6116 V10: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Data Loading Manual.

WREATHALL,J.

NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.

WRIGHT,J.E.

NUREG/CR-6327: MODELS FOR EMBRITTLEMENT RECOVERY DUE TO ANNEALING OF REACTOR PRESSURE VESSEL STEELS.

WRIGHT,R.Q.

NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.

NUREG/CR-6328: ADEQUACY OF THE 123-GROUP CROSS-SECTION LIBRARY FOR CRITICALITY ANALYSES OF WATER-MODERATED URANIUM SYSTEMS.

XIA,L.

NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION.A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions.

XIE,J.W.

NUREG/CR-3469 V08: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.

YANG,K.T.

NUREG/CR-6017: FIRE MODELING OF THE HEISS DAMPF REAKTOR CONTAINMENT.

YARDUMIAN,J.

NUREG-0525 V02 R03: SAFEGUARDS SUMMARY EVENT LIST (SSEL).January 1, 1990 Through December 31, 1994.

YOUNG,M.L.

NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Main Report.

NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Appendices A And B.

NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment.Appendices C,D,E,F,G,H.

ZEIGLER,S.L.

NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Verification And Validation (V&V) Manual.

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NUREG/CR-6224: PARAMETRIC STUDY OF THE POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS.

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ACRS Report

NUREG-1125 V16: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS.1994 Annual.

AEOD

NUREG-1272 V08 N02: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA.1993 Annual Report - Nuclear Materials.

ALARA

NUREG/CP-0143: PROCEEDINGS OF THE THIRD INTERNATIONAL WORKSHOP ON THE IMPLEMENTATION OF ALARA AT NUCLEAR POWER PLANTS.Held At Hauppauge, Long Island, New York.
NUREG/CR-3469 V08: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.

ANS

NUREG/CR-6385: COMPARISONS OF ANS,ASME,AWS, AND NFPA STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.

ANSI

NUREG/CR-6386: COMPARISONS OF ANSI STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.

ASME

NUREG/CR-6385: COMPARISONS OF ANS,ASME,AWS, AND NFPA STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.

ASME Code

NUREG-1482: GUIDELINES FOR INSERVICE TESTING AT NUCLEAR POWER PLANTS.
NUREG/CR-3243: COMPARISON OF ASME CODE FATIGUE EVALUATION METHODS FOR NUCLEAR CLASS 1 PIPING WITH CLASS 2 OR 3 PIPING.

ASTM

NUREG/CR-6382: COMPARISONS OF ASTM STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.

AUTOCASK

NUREG/CR-5657: AUTOCASK (AUTOMATIC GENERATION OF 3-D CASK MODELS).A Microcomputer Based System For Shipping Cask Design Review Analysis.

Abnormal Occurrence

NUREG-0090 V17 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1994.
NUREG-0090 V18 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.January-March 1995.

Accelerometer System

NUREG/CR-6313 V01: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEM.Summary of Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.
NUREG/CR-6313 V02: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEMS.Supplemental Appendices Presenting Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.

Acceptance Criteria

NUREG-1280 R01: STANDARD FORMAT AND CONTENT ACCEPTANCE CRITERIA FOR THE MATERIAL CONTROL AND ACCOUNTING (MC&A) REFORM AMENDMENT.

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NUREG/CR-4674 V21: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Main Report And Appendices A-H.
NUREG/CR-4674 V22: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Appendix I.

Accounting

NUREG-1280 R01: STANDARD FORMAT AND CONTENT ACCEPTANCE CRITERIA FOR THE MATERIAL CONTROL AND ACCOUNTING (MC&A) REFORM AMENDMENT.

Advanced Light Water Reactor

NUREG/CR-6358 V01: ASSESSMENT OF UNITED STATES INDUSTRY STRUCTURAL CODES AND STANDARDS FOR APPLICATION TO ADVANCED NUCLEAR POWER REACTORS.Final Report.
NUREG/CR-6358 V02: ASSESSMENT OF UNITED STATES INDUSTRY STRUCTURAL CODES AND STANDARDS FOR APPLICATION TO ADVANCED NUCLEAR POWER REACTORS.Appendices.

Advanced Reactor

NUREG/CR-6312: ASSESSMENT OF FIBER OPTIC PRESSURE SENSORS.

Advisory Committee On Nuclear Waste

NUREG-1423 V05: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON NUCLEAR WASTE.July 1993 - June 1995.

Aging

NUREG/CR-5462: AGING STUDY OF BOILING WATER REACTOR HIGH PRESSURE INJECTION SYSTEMS.
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Aging Research

NUREG/CP-0140 V03: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING.Primary Systems Integrity, Structural And Seismic Engineering, Aging Research, Products And Applications.

Air Flow Velocity

NUREG/CR-6334: NEW SENSOR FOR MEASUREMENT OF LOW AIR FLOW VELOCITY.Phase I Final Report.

Air-Operated Valve

NUREG/CR-6016: AGING AND SERVICE WEAR OF AIR-OPERATED VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.

Airborne Effluent

NUREG/CR-2907 V13: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1992.
NUREG/CR-2907 V14: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS.

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Allegations Review Team

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Annual Report

NUREG-1145 V11: U.S. NUCLEAR REGULATORY COMMISSION 1994 ANNUAL REPORT.

Atmospheric Deposition

NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment. Appendices C, D, E, F, G, H.

Atmospheric Dispersion

NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Main Report.
NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Appendices A And B.
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Atomic Safety And Licensing Board Panel

NUREG-1363 V06: ATOMIC SAFETY AND LICENSING BOARD PANEL BIENNIAL REPORT. Fiscal Years 1993 - 1994.

Austenitic Stainless Steel

NUREG/CR-6335: FATIGUE STRAIN-LIFE BEHAVIOR OF CARBON AND LOW-ALLOY STEELS, AUSTENITIC STAINLESS STEELS, AND ALLOY 600 IN LWR ENVIRONMENTS.

Austenitic Steel

NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES. Experiments And Analysis, March 1990 - December 1994.

Average Dose

NUREG-0713 V15: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES, 1993. Twenty-Sixth Annual Report.

BREATH Version 1

NUREG/CR-6333: BREATH VERSION 1.1 - COUPLED FLOW AND ENERGY TRANSPORT IN POROUS MEDIA. Simulator Description And User Guide.

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NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.

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NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function And Task Analysis.
NUREG/CR-6276: A COMPILATION OF CURRENT REGULATIONS, STANDARDS, AND GUIDELINES IN REMOTE AFTERLOADING BRACHYTHERAPY.

Buckling Analysis

NUREG/CR-6322: BUCKLING ANALYSIS OF SPENT FUEL BASKET.

Budget

NUREG-1100 V11: BUDGET ESTIMATES. Fiscal Years 1996-1997.

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NUREG/CR-6331: ATMOSPHERIC RELATIVE CONCENTRATIONS IN BUILDING WAKES.

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NUREG/CR-6257: CANDU 3 TRANSIENT ANALYSIS USING ATOMIC ENERGY OF CANADA LTD CODES.

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NUREG/CR-6315: CANDU REACTORS, THEIR REGULATION IN CANADA, AND THE IDENTIFICATION OF RELEVANT NRC SAFETY ISSUES.

CASKS

NUREG/CR-6242: CASKS (COMPUTER ANALYSIS OF STORAGE CASKS): A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR STORAGE CASK DESIGN REVIEW. User's Manual To Version 1b (Including Program Reference).

COPMA-II

NUREG/CR-6398: EVALUATION OF THE COMPUTERIZED PROCEDURES MANUAL II (COPMA-II).

Calibration

NUREG/CR-6343: ON-LINE TESTING OF CALIBRATION OF PROCESS INSTRUMENTATION CHANNELS IN NUCLEAR POWER PLANTS. Phase II Final Report.

Candidate Guideline

NUREG/CR-6263 V01: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS. Candidate Guidelines, Technical Basis And Research Needs. Executive Summary.
NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS. Candidate Guidelines, Technical Basis And Research Needs. Main Report.

Capillary Barrier

NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS. Progress Report Of Field Experiments At A Humid Region Site, Beltsville, Maryland.

Certificates Of Compliance

NUREG-0383 V01 R18: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Report Of NRC Approved Packages.
 NUREG-0383 V02 R18: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Certificates Of Compliance.
 NUREG-0383 V03 R15: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Report Of NRC Approved Quality Assurance Programs for Radioactive Materials Packages.

Cesium Hydroxide

NUREG-1519: SURFACE INTERACTIONS OF CESIUM AND BORIC ACID WITH STAINLESS STEEL.

Check Valve

NUREG/CR-5944 V02: A CHARACTERIZATION OF CHECK VALVE DEGRADATION AND FAILURE EXPERIENCE IN THE NUCLEAR POWER INDUSTRY. 1991 Failures.

Circumferential Crack

NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES. Experiments And Analysis, March 1990 - December 1994.
 NUREG/CR-6299: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE.

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 NUREG/CR-6386: COMPARISONS OF ANSI STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN, NUREG-0800, AND RELATED DOCUMENTS.

Code Assessment

NUREG/CR-6285: SEVERE ACCIDENT NATURAL CIRCULATION STUDIES AT THE INEL.

Code System

NUREG/CR-0200 V1 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Control Modules.
 NUREG/CR-0200 V2P1R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Functional Modules F1-F8.
 NUREG/CR-0200 V2P2R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Functional Modules F9-F16.
 NUREG/CR-0200 V3 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Miscellaneous.

Codes And Standards

NUREG/CR-5973 R02: CODES AND STANDARDS AND OTHER GUIDANCE CITED IN REGULATORY DOCUMENTS.

Computer Analyses

NUREG/CR-0200 V1 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Control Modules.
 NUREG/CR-0200 V2P1R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Functional Modules F1-F8.
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NUREG/CR-0200 V3 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Miscellaneous.

Computerized Procedure

NUREG/CR-6398: EVALUATION OF THE COMPUTERIZED PROCEDURES MANUAL II (COPMA-II).

Concrete

NUREG/CR-6188 V02: MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE. Annual Report For FY 1994.

Constraint Effect

NUREG/CR-6259: CONSTRAINT EFFECTS ON FRACTURE INITIATION LOADS IN HSST WIDE-PLATE TESTS.

Construction Inspection Program

NUREG-1528: RECONSTITUTION OF THE MANUAL CHAPTER 2512 CONSTRUCTION INSPECTION PROGRAM FOR WATTS BAR UNIT 1.

Containment

NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Final Report.
 NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Draft Report For Comment.
 NUREG-1522: ASSESSMENT OF INSERVICE CONDITIONS OF SAFETY-RELATED NUCLEAR PLANT STRUCTURES.
 NUREG/CR-6017: FIRE MODELING OF THE HEISS DAMPF REAKTOR CONTAINMENT.
 NUREG/CR-6154 V02: EXPERIMENTAL RESULTS FROM CONTAINMENT PIPING BELLOWS SUBJECTED TO SEVERE ACCIDENT CONDITIONS. Results From Bellows Tested In Corroded Conditions.

Containment Building

NUREG/CR-6184: SEPARATE EFFECTS TESTING AND ANALYSES TO INVESTIGATE LINER TEARING OF THE 1:16-SCALE REINFORCED CONCRETE CONTAINMENT BUILDING.

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NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.

Contaminant Transport

NUREG/CR-6114 V02: AUXILIARY ANALYSES IN SUPPORT OF PERFORMANCE ASSESSMENT OF A HYPOTHETICAL LOW-LEVEL WASTE FACILITY. Two-Phase Flow And Contaminant Transport In Unsaturated Soils With Application To Low-Level Radioactive Waste Disposal.

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NUREG/CR-6390: RADIOLOGICAL CHARACTERIZATION OF SPENT CONTROL ROD ASSEMBLIES.

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NUREG/CR-6046: ALERTNESS, PERFORMANCE, AND OFF-DUTY SLEEP ON 8-HOUR AND 12-HOUR NIGHT SHIFTS IN A SIMULATED CONTINUOUS OPERATIONS CONTROL ROOM SETTING.
 NUREG/CR-6159: USING MICRO SAINT TO PREDICT PERFORMANCE IN A NUCLEAR POWER PLANT CONTROL ROOM. A Test Of Validity And Feasibility.
 NUREG/CR-6331: ATMOSPHERIC RELATIVE CONCENTRATIONS IN BUILDING WAKES.

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Core Damage

NUREG/CR-4674 V21: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1994 A STATUS REPORT. Main Report And Appendices A-H.
 NUREG/CR-4674 V22: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1994 A STATUS REPORT. Appendix I.

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NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS.

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- NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, October 1993 - March 1994.
NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, April-September 1994.

Cost Benefit

- NUREG/CR-6349: COST-BENEFIT CONSIDERATIONS IN REGULATORY ANALYSIS.

Cost Estimate

- NUREG/CR-6054: ESTIMATING PRESSURIZED WATER REACTOR DECOMMISSIONING COSTS. A User's Manual For The PWR Cost Estimating Computer Program (CECP) Software.

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- NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

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- NUREG/CR-6264 V01: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. An Assessment Of The J(M) Parameter.
NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions.

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- NUREG/CR-6335: FATIGUE STRAIN-LIFE BEHAVIOR OF CARBON AND LOW-ALLOY STEELS, AUSTENITIC STAINLESS STEELS, AND ALLOY 600 IN LWR ENVIRONMENTS.

Criticality Safety

- NUREG/CR-6328: ADEQUACY OF THE 123-GROUP CROSS-SECTION LIBRARY FOR CRITICALITY ANALYSES OF WATER-MODERATED URANIUM SYSTEMS.

Cross-Section Library

- NUREG/CR-6328: ADEQUACY OF THE 123-GROUP CROSS-SECTION LIBRARY FOR CRITICALITY ANALYSES OF WATER-MODERATED URANIUM SYSTEMS.

Damage Progression

- NUREG/CR-6150 V02: SCDAP/RELAP/MOD 3.1 CODE MANUAL. Damage Progression Model Theory.

Data Summary Report

- NUREG/CR-6318: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-7.

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- NUREG/CR-6224: PARAMETRIC STUDY OF THE POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS.
NUREG/CR-6368: EXPERIMENTAL INVESTIGATION OF SEDIMENTATION OF LOCA-GENERATED FIBROUS DEBRIS AND SLUDGE IN BWR SUPPRESSION POOLS.

Decommissioning

- NUREG-1307 R05: REPORT ON WASTE BURIAL CHARGES. Escalation Of Decommissioning Waste Disposal Costs At Low-Level Waste Burial Facilities.
NUREG-1444 S01: SITE DECOMMISSIONING MANAGEMENT PLAN.
NUREG-1505 DRFT FC: A NONPARAMETRIC STATISTICAL METHODOLOGY FOR THE DESIGN AND ANALYSIS OF FINAL STATUS DECOMMISSIONING SURVEYS. Draft Report For Comment.
NUREG-1506 DRFT FC: MEASUREMENT METHODS FOR RADIOLOGICAL SURVEYS IN SUPPORT OF NEW DECOMMISSIONING CRITERIA. Draft Report For Comment.
NUREG-1507 DRFT FC: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS. Draft Report For Comment.
NUREG/CR-5884 V01: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....
NUREG/CR-5884 V02: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....
NUREG/CR-6054: ESTIMATING PRESSURIZED WATER REACTOR DECOMMISSIONING COSTS. A User's Manual For The PWR Cost Estimating Computer Program (CECP) Software.

- NUREG/CR-6307: SUMMARY OF COMMENTS RECEIVED AT WORKSHOP ON USE OF A SITE SPECIFIC ADVISORY BOARD (SSAB) TO FACILITATE PUBLIC PARTICIPATION IN DECOMMISSIONING CASES.

- NUREG/CR-6390: RADIOLOGICAL CHARACTERIZATION OF SPENT CONTROL ROD ASSEMBLIES.

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- NUREG-1522: ASSESSMENT OF INSERVICE CONDITIONS OF SAFETY-RELATED NUCLEAR PLANT STRUCTURES.
NUREG/CR-5944 V02: A CHARACTERIZATION OF CHECK VALVE DEGRADATION AND FAILURE EXPERIENCE IN THE NUCLEAR POWER INDUSTRY. 1991 Failures.
NUREG/CR-6089: DETECTION OF PUMP DEGRADATION.

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- NUREG/CR-6261: A SUMMARY OF ORNL FISSION PRODUCT RELEASE TESTS WITH RECOMMENDED RELEASE RATES AND DIFFUSION COEFFICIENTS.

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- NUREG/CR-6266: ANALYSIS OF BORON DILUTION IN A FOUR-LOOP PWR.

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- NUREG/CR-2850 V13: DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1991.

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- NUREG/CR-6354 DRF FC: PERFORMANCE TESTING OF ELECTRONIC PERSONAL DOSIMETERS. Draft Report For Comment.

Dosimetry

- NUREG/CR-6112: IMPACT OF REDUCED DOSE LIMITS ON NRC LICENSED ACTIVITIES. Major Issues In The Implementation Of ICRP/NCRP Dose Limit Recommendations. Final Report.

Dusty Gas Model

- NUREG/CR-6347: MULTI-PHASE REACTIVE TRANSPORT THEORY.

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- NUREG/CR-6224: PARAMETRIC STUDY OF THE POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS.

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- NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.

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- NUREG/CR-5229 V07: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1994. Annual Report.
NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS TEST RESULTS. Low-Level Waste Data Base Program. Test Results For Fiscal Years 1986, 1987, 1988, And 1989.

Earthquake

- NUREG/CR-6239 V01: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Main Report.
NUREG/CR-6239 V02: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Appendices.
NUREG/CR-6240: APPLICATION OF BOUNDING SPECTRA TO SEISMIC DESIGN OF PIPING BASED ON THE PERFORMANCE OF ABOVE GROUND PIPING IN POWER PLANTS SUBJECTED TO STRONG MOTION EARTHQUAKES.
NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.

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- NUREG/CR-5591 V05 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Progress Report For April 1994 Through September 1994.

NUREG/CR-6275: MECHANICAL PROPERTIES OF THERMALLY AGED CAST STAINLESS STEELS FROM SHIPPINGPORT REACTOR COMPONENTS.
 NUREG/CR-6327: MODELS FOR EMBRITTLEMENT RECOVERY DUE TO ANNEALING OF REACTOR PRESSURE VESSEL STEELS.

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NUREG/CR-6310: AN ANALYSIS OF POTASSIUM IODIDE (KI) PROPHYLAXIS FOR THE GENERAL PUBLIC IN THE EVENT OF A NUCLEAR ACCIDENT.

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NUREG/CR-6333: BREATH VERSION 1.1 - COUPLED FLOW AND ENERGY TRANSPORT IN POROUS MEDIA.Simulator Description And User Guide.

Enforcement Action

NUREG-0940 V13 N4 P1: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED REACTOR LICENSEES.Quarterly Progress Report,October-December 1994.
 NUREG-0940 V13 N4 P2: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED MEDICAL LICENSEES.Quarterly Progress Report,October-December 1994.
 NUREG-0940 V13 N4 P3: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED MATERIAL LICENSEES (NON-MEDICAL).Quarterly Progress Report,October-December 1994.
 NUREG-0940 V14 N1 P1: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, REACTOR LICENSEES.Quarterly Progress Report,January-March 1995.
 NUREG-0940 V14 N1 P2: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, MEDICAL LICENSEES.Quarterly Progress Report,January-March 1995.
 NUREG-0940 V14 N1 P3: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, MATERIAL LICENSEES (NON-MEDICAL).Quarterly Progress Report,January-March 1995.
 NUREG-0940 V14 N2 P1: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED,INDIVIDUAL ACTIONS.Quarterly Progress Report,April-June 1995.
 NUREG-0940 V14 N2 P2: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED,REACTOR LICENSEES.Quarterly Progress Report,April-June 1995.
 NUREG-0940 V14 N2 P3: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MATERIAL LICENSEES.Quarterly Progress Report,April-June 1995.
 NUREG-1600: GENERAL STATEMENT OF POLICY AND PROCEDURE FOR NRC ENFORCEMENT ACTIONS.Enforcement Policy.

Enforcement Policy

NUREG-1600: GENERAL STATEMENT OF POLICY AND PROCEDURE FOR NRC ENFORCEMENT ACTIONS.Enforcement Policy.

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NUREG-1525: ASSESSMENT OF THE NRC ENFORCEMENT PROGRAM.

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NUREG/CR-6260: APPLICATION OF NUREG/CR-5999 INTERIM FATIGUE CURVES TO SELECTED NUCLEAR POWER PLANT COMPONENTS.

Environmental Simulation Approach

NUREG/CR-6351: REVIEW OF SCENARIO SELECTION APPROACHES FOR PERFORMANCE ASSESSMENT OF HIGH-LEVEL WASTE REPOSITORIES AND RELATED ISSUES.

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NUREG/CR-6305: BLT-EC (BREACH, LEACH, TRANSPORT, AND EQUILIBRIUM CHEMISTRY), A FINITE-ELEMENT MODEL FOR ASSESSING THE RELEASE OF RADIONUCLIDES FROM LOW-LEVEL WASTE DISPOSAL UNITS.Background, Theory, And Model Description.

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NUREG/CR-6305: BLT-EC (BREACH, LEACH, TRANSPORT, AND EQUILIBRIUM CHEMISTRY), A FINITE-ELEMENT MODEL FOR ASSESSING THE RELEASE OF RADIONUCLIDES FROM LOW-LEVEL WASTE DISPOSAL UNITS.Background, Theory, And Model Description.

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NUREG-1122 R01: KNOWLEDGE AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS:PRESSURIZED WATER REACTORS.

NUREG-1123 R01: KNOWLEDGE AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS: BOILING WATER REACTORS.

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NUREG-1478 R01: NON-POWER REACTOR OPERATOR LICENSING EXAMINER STANDARDS.

Exercise

NUREG-1514: GUIDANCE FOR A LARGE TABLETOP EXERCISE FOR A NUCLEAR POWER PLANT.

Failure Mode

NUREG/CP-0145: WORKSHOP ON DEVELOPING SAFE SOFTWARE.Held At Hotel Del Coronado,San Diego,CA,July 22-23,1992.

Failure Mode Analysis

NUREG/CR-6002: RISK-BASED MAINTENANCE MODELING.Prioritization Of Maintenance Importances And Quantification Of Maintenance Effectiveness.

Fatigue

NUREG/CR-6046: ALERTNESS, PERFORMANCE, AND OFF-DUTY SLEEP ON 8-HOUR AND 12-HOUR NIGHT SHIFTS IN A SIMULATED CONTINUOUS OPERATIONS CONTROL ROOM SETTING.

Fatigue Design Curve

NUREG/CR-6335: FATIGUE STRAIN-LIFE BEHAVIOR OF CARBON AND LOW-ALLOY STEELS, AUSTENITIC STAINLESS STEELS, AND ALLOY 600 IN LWR ENVIRONMENTS.

Fatigue Evaluation

NUREG/CR-3243: COMPARISON OF ASME CODE FATIGUE EVALUATION METHODS FOR NUCLEAR CLASS 1 PIPING WITH CLASS 2 OR 3 PIPING.

Fiber Optic

NUREG/CR-6312: ASSESSMENT OF FIBER OPTIC PRESSURE SENSORS.

Field Lysimeter

NUREG/CR-5229 V07: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1994.Annual Report.
 NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Program.Test Results For Fiscal Years 1986, 1987,1988, And 1989.
 NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990,1991,1992, And 1993.

Final Environmental Statement

NUREG-0498 S01: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2.Docket Nos. 50-390 And 50-391.(Tennessee Valley Authority)

Financial Statement

NUREG-1470 V04: FINANCIAL STATEMENT FOR FISCAL YEAR 1994.

Fire Modeling

NUREG/CR-6017: FIRE MODELING OF THE HEISS DAMPF REAKTOR CONTAINMENT.

Fire Testing Program

NUREG/CR-6173: A SUMMARY OF THE FIRE TESTING PROGRAM AT THE GERMAN HDR TEST FACILITY.

Fire Vulnerability

NUREG/CR-6220: AN ASSESSMENT OF FIRE VULNERABILITY FOR AGED ELECTRICAL RELAYS.

Fiscal Year

NUREG-1100 V11: BUDGET ESTIMATES.Fiscal Years 1996-1997.

Fission Product

NUREG/CR-6261: A SUMMARY OF ORNL FISSION PRODUCT RELEASE TESTS WITH RECOMMENDED RELEASE RATES AND DIFFUSION COEFFICIENTS.

Fission Product Release

NUREG/CR-6318: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-7.

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Fitness For Duty

NUREG/CR-5758 V05: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY. Annual Summary Of Program Performance Reports CY 1994.

Fracture

NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING WELDS. Seventh Program Report March 1993 - December 1994.
NUREG/CR-6299: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE.
NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

Fracture Behavior

NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.

Fracture Flow

NUREG/CR-6308: AN OVERVIEW OF INSTABILITY AND FINGERING DURING IMMISCIBLE FLUID FLOW IN POROUS AND FRACTURED MEDIA.

Fracture Mechanics

NUREG/CR-5591 V05 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Progress Report For April 1994 Through September 1994.
NUREG/CR-6004: PROBABILISTIC PIPE FRACTURE EVALUATIONS FOR LEAK-RATE-DETECTION APPLICATIONS.
NUREG/CR-6259: CONSTRAINT EFFECTS ON FRACTURE INITIATION LOADS IN HSST WIDE-PLATE TESTS.
NUREG/CR-6273: BIAXIAL LOADING EFFECTS ON FRACTURE TOUGHNESS OF REACTOR PRESSURE VESSEL STEEL.
NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.

Fracture Toughness

NUREG/CR-6191: SIZE AND DEFORMATION LIMITS TO MAINTAIN CONSTRAINT IN K(IC) AND J(C) TESTING OF BEND SPECIMENS.
NUREG/CR-6273: BIAXIAL LOADING EFFECTS ON FRACTURE TOUGHNESS OF REACTOR PRESSURE VESSEL STEEL.
NUREG/CR-6275: MECHANICAL PROPERTIES OF THERMALLY AGED CAST STAINLESS STEELS FROM SHIPPINGPORT REACTOR COMPONENTS.

Fractured Media

NUREG/CR-6308: AN OVERVIEW OF INSTABILITY AND FINGERING DURING IMMISCIBLE FLUID FLOW IN POROUS AND FRACTURED MEDIA.

Fractured Zone

NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.

Fuel Damage

NUREG/CR-6318: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-7.

Fundamental Nuclear Material Control

NUREG-1280 R01: STANDARD FORMAT AND CONTENT ACCEPTANCE CRITERIA FOR THE MATERIAL CONTROL AND ACCOUNTING (MC&A) REFORM AMENDMENT.

Fusion Line Crack

NUREG/CR-6297: FRACTURE EVALUATIONS OF FUSION LINE CRACKS IN NUCLEAR PIPE BIMETALLIC WELDS.

Gamma Knife

NUREG/CR-6324: QUALITY ASSURANCE FOR GAMMA KNIVES.

Gate Valve

NUREG/CP-0146: PROCEEDINGS OF THE WORKSHOP ON GATE VALVE PRESSURE LOCKING AND THERMAL BINDING.

Generic Safety Issues

NUREG-0933 S18: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-0933 S19: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-1435 S04: STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS. TMI Action Plan Requirements, Unresolved Safety Issues, Generic Safety Issues, Other Multiplant Action Issues.

Geologic Disposal

NUREG/CP-0147: PROCEEDINGS OF THE WORKSHOP ON THE ROLE OF NATURAL ANALOGS IN GEOLOGIC DISPOSAL OF HIGH-LEVEL NUCLEAR WASTE. Held In San Antonio, Texas, July 22-25, 1991.

Geologic Repository

NUREG-1323 R01: LICENSE APPLICATION REVIEW PLAN FOR A GEOLOGIC REPOSITORY FOR SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE. Draft Review Plan.

Graphical Evaluation Module

NUREG/CR-6116 V06: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Graphical Evaluation Module (GEM) Reference Manual.

Groundwater

NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.

Health Physic

NUREG/CR-6334: NEW SENSOR FOR MEASUREMENT OF LOW AIR FLOW VELOCITY. Phase I Final Report.

Heat Pipe Phenomena

NUREG/CR-6347: MULTI-PHASE REACTIVE TRANSPORT THEORY.

Heavy-Section Steel Irradiation Program

NUREG/CR-5591 V03: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Progress Report For October 1991 - September 1992.
NUREG/CR-5591 V04 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Semiannual Progress Report For April-September 1993.
NUREG/CR-5591 V05 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Semiannual Progress Report For September 1993 Through March 1994.
NUREG/CR-5591 V05 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Progress Report For April 1994 Through September 1994.
NUREG/CR-5591 V06 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Semiannual Progress Report For October 1994 Through March 1995.

Heavy-Section Steel Technology Program

NUREG/CR-4219 V10 N2: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM. Semiannual Progress Report For April-September 1993.
NUREG/CR-4219 V11 N1: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM. Semiannual Progress Report For October 1993 - March 1994.

Heiss Dampf Reaktor

NUREG/CR-6017: FIRE MODELING OF THE HEISS DAMPF REAKTOR CONTAINMENT.
NUREG/CR-6173: A SUMMARY OF THE FIRE TESTING PROGRAM AT THE GERMAN HDR TEST FACILITY.

High Integrity Software

NUREG/CR-6263 V01: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS. Candidate Guidelines, Technical Basis And Research Needs. Executive Summary.
NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS. Candidate Guidelines, Technical Basis And Research Needs. Main Report.

High Integrity System

NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Main Report.
NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Appendices A-D.

High Pressure Injection System

NUREG/CR-5462: AGING STUDY OF BOILING WATER REACTOR HIGH PRESSURE INJECTION SYSTEMS.

High-Level Nuclear Waste

NUREG/CP-0147: PROCEEDINGS OF THE WORKSHOP ON THE ROLE OF NATURAL ANALOGS IN GEOLOGIC DISPOSAL OF HIGH-LEVEL NUCLEAR WASTE. Held In San Antonio, Texas, July 22-25, 1991.

High-Level Radioactive Waste

NUREG-1323 R01: LICENSE APPLICATION REVIEW PLAN FOR A GEOLOGIC REPOSITORY FOR SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE. Draft Review Plan.

High-Level Waste Geologic Repository

NUREG/CR-6356: HYDRAULIC CHARACTERIZATION OF HYDROTHERMALLY ALTERED NOPAL TUFF.

High-Level Waste Repository

NUREG-1464: NRC ITERATIVE PERFORMANCE ASSESSMENT PHASE 2. Development Of Capabilities For Review Of A Performance Assessment For A High-Level Waste Repository.

NUREG/CR-6351: REVIEW OF SCENARIO SELECTION APPROACHES FOR PERFORMANCE ASSESSMENT OF HIGH-LEVEL WASTE REPOSITORIES AND RELATED ISSUES.

Human Factor

NUREG/CP-0140 V01: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING. Plenary Session, Advanced Instrumentation & Control Hardware & Software, Human Factors Research, IPE & PRA.
 NUREG/CP-0144 V01: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Summary.
 NUREG/CP-0144 V02: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Supporting Documents.
 NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.
 NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function And Task Analysis.
 NUREG/CR-6159: USING MICRO SAINT TO PREDICT PERFORMANCE IN A NUCLEAR POWER PLANT CONTROL ROOM. A Test Of Validity And Feasibility.
 NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.
 NUREG/CR-6398: EVALUATION OF THE COMPUTERIZED PROCEDURES MANUAL II (COPMA-II).

Human Factors Evaluation

NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Supporting Analyses Of Human-System Interfaces, Procedures And Practices, Training And Organizational Practices And Procedures.
 NUREG/CR-6277 V01: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Identification Of Problems And Alternative Approaches.
 NUREG/CR-6277 V02: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Function And Task Analysis.
 NUREG/CR-6277 V03: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Human-System Interfaces And Procedures.
 NUREG/CR-6277 V04: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Training And Organizational Analysis.
 NUREG/CR-6277 V05: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Literature Review.

Human Reliability

NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.
 NUREG/CR-6355: A LIMITED ASSESSMENT OF THE ASEP HUMAN RELIABILITY ANALYSIS PROCEDURE USING SIMULATOR EXAMINATION RESULTS.

Human factor

NUREG-0700 R01 DFC: HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE. Draft Report For Comment.

Human-System Interface

NUREG-0700 R01 DFC: HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE. Draft Report For Comment.

Hydraulic Characterization

NUREG/CR-6356: HYDRAULIC CHARACTERIZATION OF HYDROTHERMALLY ALTERED NOPAL TUFF.

Hydrology

NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS. Progress Report Of Field Experiments At A Humid Region Site, Beltsville, Maryland.

IRRAS

NUREG/CR-6116 V06: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Graphical Evaluation Module (GEM) Reference Manual.
 NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Verification And Validation (V&V) Manual.
 NUREG/CR-6116 V10: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Data Loading Manual.

Immiscible Displacement

NUREG/CR-6308: AN OVERVIEW OF INSTABILITY AND FINGERING DURING IMMISCIBLE FLUID FLOW IN POROUS AND FRACTURED MEDIA.

Industrial Radiography

NUREG-0713 V15: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES, 1993. Twenty-Sixth Annual Report.

Infiltration Analysis

NUREG/CR-6333: BREATH VERSION 1.1 - COUPLED FLOW AND ENERGY TRANSPORT IN POROUS MEDIA. Simulator Description And User Guide.

Information Digest

NUREG-1350 V07: NUCLEAR REGULATORY COMMISSION INFORMATION DIGEST. 1995 Edition.

Inservice Inspection

NUREG/CR-6313 V01: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEM. Summary of Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.
 NUREG/CR-6313 V02: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEMS. Supplemental Appendices Presenting Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.

Inservice Testing

NUREG-1482: GUIDELINES FOR INSERVICE TESTING AT NUCLEAR POWER PLANTS.

Instructor Station

NUREG-1527: NRC'S OBJECT-ORIENTED SIMULATOR INSTRUCTOR STATION.

Instrumentation

NUREG-1505 DRFT FC: A NONPARAMETRIC STATISTICAL METHODOLOGY FOR THE DESIGN AND ANALYSIS OF FINAL STATUS DECOMMISSIONING SURVEYS. Draft Report For Comment.
 NUREG-1506 DRFT FC: MEASUREMENT METHODS FOR RADIOLOGICAL SURVEYS IN SUPPORT OF NEW DECOMMISSIONING CRITERIA. Draft Report For Comment.
 NUREG-1507 DRFT FC: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS. Draft Report For Comment.

Interface Theory

NUREG/CR-6150 V01: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. Interface Theory.

Internal Event

NUREG/CR-6143 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Summary Of Results.
 NUREG/CR-6144 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Summary Of Results.

Irradiated Reactor Fuel

NUREG-0725 R10: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL.

J-Integral

NUREG/CR-6191: SIZE AND DEFORMATION LIMITS TO MAINTAIN CONSTRAINT IN K(IC) AND J(C) TESTING OF BEND SPECIMENS.

J-R Curve

NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING WELDS. Seventh Program Report March 1993 - December 1994.
 NUREG/CR-6264 V01: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. An Assessment Of The J(M) Parameter.
 NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions.
 NUREG/CR-6297: FRACTURE EVALUATIONS OF FUSION LINE CRACKS IN NUCLEAR PIPE BIMETALLIC WELDS.
 NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

J-Resistance

NUREG/CR-6264 V01: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. An Assessment Of The J(M) Parameter.
 NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions.

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LWR

- NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS.
NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, October 1993 - March 1994.
NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, April-September 1994.

Leak-Before-Break

- NUREG/CR-6004: PROBABILISTIC PIPE FRACTURE EVALUATIONS FOR LEAK-RATE-DETECTION APPLICATIONS.
NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES. Experiments And Analysis, March 1990 - December 1994.
NUREG/CR-6251: STAINLESS STEEL SUBMERGED ARC WELD FUSION LINE TOUGHNESS.
NUREG/CR-6299: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE.
NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

Leak-Test Program

- NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Final Report.
NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Draft Report For Comment.

Legal Issuances

- NUREG-0750 V40: NUCLEAR REGULATORY COMMISSION ISSUANCES. Opinions And Decisions Of The Nuclear Regulatory Commission With Selected Orders. July-December 1994.
NUREG-0750 V40 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-December 1994.
NUREG-0750 V40 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1994. Pages 169-318.
NUREG-0750 V40 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1994. Pages 319-387.
NUREG-0750 V41 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-March 1995.
NUREG-0750 V41 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-June 1995.
NUREG-0750 V41 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1995. Pages 1-69.
NUREG-0750 V41 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1995. Pages 71-178.
NUREG-0750 V41 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1995. Pages 179-243.
NUREG-0750 V41 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1995. Pages 245-319.
NUREG-0750 V41 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1995. Pages 321-380.
NUREG-0750 V41 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1995. Pages 381-496.
NUREG-0750 V42 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-September 1995.
NUREG-0750 V42 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1995. Pages 1-45.
NUREG-0750 V42 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1995. Pages 47-97.
NUREG-0750 V42 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1995. Pages 99-110.
NUREG-0750 V42 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1995. Pages 111-180.

Lessons Learned

- NUREG-1526: LESSONS LEARNED FROM EARLY IMPLEMENTATION OF THE MAINTENANCE RULE AT NINE NUCLEAR POWER PLANTS.

License Application

- NUREG-1323 R01: LICENSE APPLICATION REVIEW PLAN FOR A GEOLOGIC REPOSITORY FOR SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE. Draft Review Plan.

Licensed Fuel Facility Status Report

- NUREG-0430 V14: LICENSED FUEL FACILITY STATUS REPORT. Inventory Difference Data. July 1, 1993 - June 30, 1994. (Gray Book II)

Licensed Operating Reactors

- NUREG-0020 V19: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of December 31, 1994. (Gray Book I)

Licensee Event Report

- NUREG/CR-4674 V21: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1994 A STATUS REPORT. Main Report And Appendices A-H.
NUREG/CR-4674 V22: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1994 A STATUS REPORT. Appendix I.

Licensing Evaluation

- NUREG/CR-0200 V1 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Control Modules.
NUREG/CR-0200 V2P1R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Functional Modules F1-F8.
NUREG/CR-0200 V2P2R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Functional Modules F9-F16.
NUREG/CR-0200 V3 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Miscellaneous.

Light Water Reactor

- NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS.
NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, October 1993 - March 1994.
NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, April-September 1994.

Liner

- NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990, 1991, 1992, And 1993.

Liner Tearing

- NUREG/CR-6184: SEPARATE EFFECTS TESTING AND ANALYSES TO INVESTIGATE LINER TEARING OF THE 1:16-SCALE REINFORCED CONCRETE CONTAINMENT BUILDING.

Liquid Effluent

- NUREG/CR-2907 V13: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1992.
NUREG/CR-2907 V14: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS.

Low Enriched Uranium

- NUREG-1065 R02: ACCEPTABLE STANDARD FORMAT AND CONTENT FOR THE FUNDAMENTAL NUCLEAR MATERIAL CONTROL (FNMC) PLAN REQUIRED FOR LOW-ENRICHED URANIUM FACILITIES.

Low Power

- NUREG/CR-6143 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Summary Of Results.
NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Main Report And Appendices.
NUREG/CR-6143 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Supporting MELCOR Calculations.
NUREG/CR-6144 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Summary Of Results.
NUREG/CR-6144 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Evaluation Of Severe Accident Risk During Mid-Loop Operations. Main Report.
NUREG/CR-6144 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Evaluation Of Severe Accident Risk During Mid-Loop Operations. Appendices.

Low-Alloy Steel

- NUREG/CR-6335: FATIGUE STRAIN-LIFE BEHAVIOR OF CARBON AND LOW-ALLOY STEELS, AUSTENITIC STAINLESS STEELS, AND ALLOY 600 IN LWR ENVIRONMENTS.

Low-Level Nuclear Waste

NUREG/CR-6390: RADIOLOGICAL CHARACTERIZATION OF SPENT CONTROL ROD ASSEMBLIES.

Low-Level Radioactive Waste

NUREG/CR-6188 V02: MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE. Annual Report For FY 1994.

Low-Level Radioactive Waste Disposal

NUREG/CR-6114 V02: AUXILIARY ANALYSES IN SUPPORT OF PERFORMANCE ASSESSMENT OF A HYPOTHETICAL LOW-LEVEL WASTE FACILITY. Two-Phase Flow And Contaminant Transport In Unsaturated Soils With Application To Low-Level Radioactive Waste Disposal.

Low-Level Waste Data Base

NUREG/CR-5229 V07: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1994. Annual Report.

NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Program. Test Results For Fiscal Years 1986, 1987, 1988, And 1989.

NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990, 1991, 1992, And 1993.

Low-Level Waste Disposal

NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS. Progress Report Of Field Experiments At A Humid Region Site, Beltsville, Maryland.

NUREG/CR-5927 V02: EVALUATION OF A PERFORMANCE ASSESSMENT METHODOLOGY FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES. Validation Needs.

NUREG/CR-6305: BLT-EC (BREACH, LEACH, TRANSPORT, AND EQUILIBRIUM CHEMISTRY), A FINITE-ELEMENT MODEL FOR ASSESSING THE RELEASE OF RADIONUCLIDES FROM LOW-LEVEL WASTE DISPOSAL UNITS. Background, Theory, And Model Description.

Low-Level Waste Facility

NUREG/CR-6284: CRITICALITY SAFETY CRITERIA FOR LICENSE REVIEW OF LOW-LEVEL WASTE FACILITIES.

MACCS

NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

MELCOR Computer Code

NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS. Primer And User's Guides. Version 1.8.3 September 1994.

NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS. Reference Manuals. Version 1.8.3 September 1994.

Maintenance

NUREG/CR-6002: RISK-BASED MAINTENANCE MODELING. Prioritization Of Maintenance Importances And Quantification Of Maintenance Effectiveness.

Maintenance Rule

NUREG-1526: LESSONS LEARNED FROM EARLY IMPLEMENTATION OF THE MAINTENANCE RULE AT NINE NUCLEAR POWER PLANTS.

Management Plan

NUREG-1444 S01: SITE DECOMMISSIONING MANAGEMENT PLAN.

Manual Chapter 2512

NUREG-1528: RECONSTITUTION OF THE MANUAL CHAPTER 2512 CONSTRUCTION INSPECTION PROGRAM FOR WATTS BAR UNIT 1.

Material Control

NUREG-1065 R02: ACCEPTABLE STANDARD FORMAT AND CONTENT FOR THE FUNDAMENTAL NUCLEAR MATERIAL CONTROL (FNMC) PLAN REQUIRED FOR LOW-ENRICHED URANIUM FACILITIES.

NUREG-1280 R01: STANDARD FORMAT AND CONTENT ACCEPTANCE CRITERIA FOR THE MATERIAL CONTROL AND ACCOUNTING (MC&A) REFORM AMENDMENT.

Medical Device

NUREG-1535: INGESTION OF PHOSPHORUS-32 AT MASSACHUSETTS INSTITUTE OF

TECHNOLOGY, CAMBRIDGE, MASSACHUSETTS, IDENTIFIED ON AUGUST 19, 1995.

NUREG/CR-6323: RELATIVE RISK ANALYSIS IN REGULATING THE USE OF RADIATION-EMITTING MEDICAL DEVICES. A Preliminary Application.

Medical Misadministration

NUREG/CR-6277 V01: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Identification Of Problems And Alternative Approaches.

NUREG/CR-6277 V02: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Function And Task Analysis.

NUREG/CR-6277 V03: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Human-System Interfaces And Procedures.

NUREG/CR-6277 V04: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Training And Organizational Analysis.

NUREG/CR-6277 V05: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Literature Review.

Medical Quality Management

NUREG/CR-6277 V01: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Identification Of Problems And Alternative Approaches.

NUREG/CR-6277 V02: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Function And Task Analysis.

NUREG/CR-6277 V03: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Human-System Interfaces And Procedures.

NUREG/CR-6277 V04: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Training And Organizational Analysis.

NUREG/CR-6277 V05: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Literature Review.

Medical Use

NUREG/CP-0144 V01: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Summary.

NUREG/CP-0144 V02: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Supporting Documents.

Microbial Degradation

NUREG/CR-6188 V02: MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE. Annual Report For FY 1994.

Mid-Loop Operation

NUREG/CR-6144 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Evaluation Of Severe Accident Risk During Mid-Loop Operations. Main Report.

NUREG/CR-6144 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Evaluation Of Severe Accident Risk During Mid-Loop Operations. Appendices.

Mine Seismicity

NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.

Moisture Redistribution

NUREG/CR-6348: THERMALLY DRIVEN MOISTURE REDISTRIBUTION IN PARTIALLY SATURATED POROUS MEDIA.

Multiplant Action Issues

NUREG-1435 S04: STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS. TMI Action Plan Requirements, Unresolved Safety Issues, Generic Safety Issues, Other Multiplant Action Issues.

NPRDS

NUREG/CR-5944 V02: A CHARACTERIZATION OF CHECK VALVE DEGRADATION AND FAILURE EXPERIENCE IN THE NUCLEAR POWER INDUSTRY. 1991 Failures.

NURETH-7

NUREG/CP-0142 V01: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7). Sessions 1-5.

NUREG/CP-0142 V02: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7). Sessions 6-11.

NUREG/CP-0142 V03: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7). Sessions 12-16.

NUREG/CP-0142 V04: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7). Sessions 17-24.

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Natural Analog

NUREG/CP-0147: PROCEEDINGS OF THE WORKSHOP ON THE ROLE OF NATURAL ANALOGS IN GEOLOGIC DISPOSAL OF HIGH-LEVEL NUCLEAR WASTE.Held In San Antonio,Texas,July 22-25, 1991.

Natural Circulation

NUREG/CR-6285: SEVERE ACCIDENT NATURAL CIRCULATION STUDIES AT THE INEL.

New Sensor

NUREG/CR-6334: NEW SENSOR FOR MEASUREMENT OF LOW AIR FLOW VELOCITY.Phase I Final Report.

Nozzle

NUREG/CR-6260: APPLICATION OF NUREG/CR-5999 INTERIM FATIGUE CURVES TO SELECTED NUCLEAR POWER PLANT COMPONENTS.

Nuclear Accident

NUREG/CR-6310: AN ANALYSIS OF POTASSIUM IODIDE (KI) PROPHYLAXIS FOR THE GENERAL PUBLIC IN THE EVENT OF A NUCLEAR ACCIDENT.

Nuclear Air Cleaning

NUREG/CP-0141: PROCEEDINGS OF THE 23RD DOE/NRC NUCLEAR AIR CLEANING CONFERENCE.Held In Buffalo,New York,July 25-28,1994.

Nuclear Fuel Facility

NUREG/CR-6287: MANAGEMENT CONCEPTS AND SAFETY APPLICATIONS FOR NUCLEAR FUEL FACILITIES.

Nuclear Materials Licensees

NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.

Nuclear Plant Analyzer

NUREG/CR-6291 V01: NUCLEAR PLANT ANALYZER.Installation Manual.
NUREG/CR-6291 V02: NUCLEAR PLANT ANALYZER.Analyzer Reference Manual.
NUREG/CR-6291 V03: NUCLEAR PLANT ANALYZER.Computer Visual System Reference Manual.
NUREG/CR-6291 V04: NUCLEAR PLANT ANALYZER.Programmer's Manual.

Nuclear Plant Safety System

NUREG/CR-6263 V01: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Executive Summary.
NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Main Report.

Nuclear Plant Structure

NUREG-1522: ASSESSMENT OF INSERVICE CONDITIONS OF SAFETY-RELATED NUCLEAR PLANT STRUCTURES.

Nuclear Power Plant

NUREG/CR-5975 R01: INCENTIVE REGULATION OF INVESTOR-OWNED NUCLEAR POWER PLANTS BY PUBLIC UTILITY REGULATORS.
NUREG/CR-6159: USING MICRO SAINT TO PREDICT PERFORMANCE IN A NUCLEAR POWER PLANT CONTROL ROOM.A Test Of Validity And Feasibility.
NUREG/CR-6172: REVIEWING PSA-BASED ANALYSES TO MODIFY TECHNICAL SPECIFICATIONS AT NUCLEAR POWER PLANTS.

Nuclear Reactor

NUREG/CP-0142 V01: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 1-5.
NUREG/CP-0142 V02: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 6-11.
NUREG/CP-0142 V03: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 12-16.
NUREG/CP-0142 V04: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 17-24.

Nuclear Regulatory Legislation

NUREG-0980 V01 N03: NUCLEAR REGULATORY LEGISLATION.103d Congress.

NUREG-0980 V02 N03: NUCLEAR REGULATORY LEGISLATION.103d Congress.

Nuclear Regulatory Research

NUREG-1266 V09: NRC SAFETY RESEARCH IN SUPPORT OF REGULATION - FY 1994.

Nuclear Safety

NUREG-0700 R01 DFC: HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE.Draft Report For Comment.

Nuclear Safety Research

NUREG/CP-0148: TRANSACTIONS OF THE TWENTY-THIRD WATER REACTOR SAFETY INFORMATION MEETING.

Occupational Exposure

NUREG/CP-0143: PROCEEDINGS OF THE THIRD INTERNATIONAL WORKSHOP ON THE IMPLEMENTATION OF ALARA AT NUCLEAR POWER PLANTS.Held At Hauppauge, Long Island, New York.
NUREG/CR-3469 V08: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.
NUREG/CR-6112: IMPACT OF REDUCED DOSE LIMITS ON NRC LICENSED ACTIVITIES. Major Issues In The Implementation Of ICRP/NCRP Dose Limit Recommendations.Final Report.

Occupational Radiation Exposure

NUREG/CR-5884 V01: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION.Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....
NUREG/CR-5884 V02: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION.Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....

Office Of The Inspector General

NUREG-1415 V07 N02: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report To Congress, October 1, 1994 - March 31, 1995.
NUREG-1415 V08 N01: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report To Congress, April 1, 1995 - September 30, 1995.

On-Line Monitoring

NUREG/CR-6343: ON-LINE TESTING OF CALIBRATION OF PROCESS INSTRUMENTATION CHANNELS IN NUCLEAR POWER PLANTS.Phase II Final Report.

Operating Experience

NUREG-1272 V08 N02: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA.1993 Annual Report - Nuclear Materials.
NUREG/CR-6089: DETECTION OF PUMP DEGRADATION.
NUREG/CR-6192: AGING AND SERVICE WEAR OF SPRING-LOADED PRESSURE RELIEF VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.

Operating Experience Feedback Report

NUREG-1275 V11: OPERATING EXPERIENCE FEEDBACK REPORT - TURBINE-GENERATOR OVERSPEED PROTECTION SYSTEM.Commercial Power Reactors.

Operational Event

NUREG/CR-4674 V21: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Main Report And Appendices A-H.
NUREG/CR-4674 V22: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Appendix I.

Operator Licensing

NUREG-1478 R01: NON-POWER REACTOR OPERATOR LICENSING EXAMINER STANDARDS.

Organization Chart

NUREG-0325 R18: U.S. NUCLEAR REGULATORY COMMISSION ORGANIZATION CHARTS AND FUNCTIONAL STATEMENTS.July 23, 1995.

Overspeed Protection System

NUREG-1275 V11: OPERATING EXPERIENCE FEEDBACK REPORT - TURBINE-GENERATOR OVERSPEED PROTECTION SYSTEM.Commercial Power Reactors.

PRA

NUREG/CR-6116 V06: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Graphical Evaluation Module (GEM) Reference Manual.

PRA Data Loading

NUREG/CR-6116 V10: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Data Loading Manual.

PWR

NUREG-1122 R01: KNOWLEDGE AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS: PRESSURIZED WATER REACTORS.

NUREG/CR-5884 V01: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....

NUREG/CR-5884 V02: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....

NUREG/CR-5954: EFFECT ON AGING ON PWR CHEMICAL AND VOLUME CONTROL SYSTEM.

NUREG/CR-6054: ESTIMATING PRESSURIZED WATER REACTOR DECOMMISSIONING COSTS. A User's Manual For The PWR Cost Estimating Computer Program (CECP) Software.

NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.

NUREG/CR-6266: ANALYSIS OF BORON DILUTION IN A FOUR-LOOP PWR.

Packaging

NUREG-0383 V01 R18: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Report Of NRC Approved Packages.

NUREG-0383 V02 R18: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Certificates Of Compliance.

NUREG-0383 V03 R15: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Report Of NRC Approved Quality Assurance Programs for Radioactive Materials Packages.

Paleoseismicity

NUREG/GR-0014: BALDCYPRESS TREE RING ELEMENTAL CONCENTRATIONS AT REELFOOT LAKE, TENNESSEE, FROM AD 1795 TO AD 1820.

Paper Mill Digester

NUREG/CR-6074 V04: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING. Technical Report On The Findings Of Task 4. Investigation Of Sealed Source for Paper Mill Digester.

Performance

NUREG/CR-6046: ALERTNESS, PERFORMANCE, AND OFF-DUTY SLEEP ON 8-HOUR AND 12-HOUR NIGHT SHIFTS IN A SIMULATED CONTINUOUS OPERATIONS CONTROL ROOM SETTING.

NUREG/CR-6159: USING MICRO SAINT TO PREDICT PERFORMANCE IN A NUCLEAR POWER PLANT CONTROL ROOM. A Test Of Validity And Feasibility.

Performance Assessment

NUREG-1464: NRC ITERATIVE PERFORMANCE ASSESSMENT PHASE 2. Development Of Capabilities For Review Of A Performance Assessment For A High-Level Waste Repository.

NUREG/CR-5927 V02: EVALUATION OF A PERFORMANCE ASSESSMENT METHODOLOGY FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES. Validation Needs.

NUREG/CR-6351: REVIEW OF SCENARIO SELECTION APPROACHES FOR PERFORMANCE ASSESSMENT OF HIGH-LEVEL WASTE REPOSITORIES AND RELATED ISSUES.

Performance Incentive

NUREG/CR-5975 R01: INCENTIVE REGULATION OF INVESTOR-OWNED NUCLEAR POWER PLANTS BY PUBLIC UTILITY REGULATORS.

Performance Testing

NUREG/CR-6354 DRF FC: PERFORMANCE TESTING OF ELECTRONIC PERSONAL DOSIMETERS. Draft Report For Comment.

Petitions For Rulemaking

NUREG-0936 V13 N03: NRC REGULATORY AGENDA. Semiannual Report, July-December 1994.

NUREG-0936 V14 N01: NRC REGULATORY AGENDA. Semiannual Report, January-June 1995.

Phosphorus-32

NUREG-1535: INGESTION OF PHOSPHORUS-32 AT MASSACHUSETTS INSTITUTE OF TECHNOLOGY, CAMBRIDGE, MASSACHUSETTS, IDENTIFIED ON AUGUST 19, 1995.

Pipe

NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING WELDS. Seventh Program Report March 1993 - December 1994.

NUREG/CR-6004: PROBABILISTIC PIPE FRACTURE EVALUATIONS FOR LEAK-RATE-DETECTION APPLICATIONS.

NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES. Experiments And Analysis, March 1990 - December 1994.

NUREG/CR-6251: STAINLESS STEEL SUBMERGED ARC WELD FUSION LINE TOUGHNESS.

NUREG/CR-6297: FRACTURE EVALUATIONS OF FUSION LINE CRACKS IN NUCLEAR PIPE BIMETALLIC WELDS.

NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.

NUREG/CR-6299: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE.

NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

Piping

NUREG/CR-3243: COMPARISON OF ASME CODE FATIGUE EVALUATION METHODS FOR NUCLEAR CLASS 1 PIPING WITH CLASS 2 OR 3 PIPING.

NUREG/CR-6154 V02: EXPERIMENTAL RESULTS FROM CONTAINMENT PIPING BELLWS SUBJECTED TO SEVERE ACCIDENT CONDITIONS. Results From Bellows Tested In Corroded Conditions.

NUREG/CR-6240: APPLICATION OF BOUNDING SPECTRA TO SEISMIC DESIGN OF PIPING BASED ON THE PERFORMANCE OF ABOVE GROUND PIPING IN POWER PLANTS SUBJECTED TO STRONG MOTION EARTHQUAKES.

Piping System

NUREG/CR-6239 V01: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Main Report.

NUREG/CR-6239 V02: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Appendices.

Porous Media

NUREG/CR-6333: BREATH VERSION 1.1 - COUPLED FLOW AND ENERGY TRANSPORT IN POROUS MEDIA. Simulator Description And User Guide.

NUREG/CR-6348: THERMALLY DRIVEN MOISTURE REDISTRIBUTION IN PARTIALLY SATURATED POROUS MEDIA.

Potassium Iodide

NUREG/CR-6310: AN ANALYSIS OF POTASSIUM IODIDE (KI) PROPHYLAXIS FOR THE GENERAL PUBLIC IN THE EVENT OF A NUCLEAR ACCIDENT.

Pressure Locking

NUREG/CP-0146: PROCEEDINGS OF THE WORKSHOP ON GATE VALVE PRESSURE LOCKING AND THERMAL BINDING.

Pressure Sensor

NUREG/CR-6312: ASSESSMENT OF FIBER OPTIC PRESSURE SENSORS.

Pressure Vessel

NUREG/CR-5591 V05 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM. Progress Report For April 1994 Through September 1994.

Pressurized Water Reactor

NUREG-1122 R01: KNOWLEDGE AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS: PRESSURIZED WATER REACTORS.

NUREG/CR-5884 V01: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On

The Financial Insurance Requirements Of The Decommissioning Rule And....

NUREG/CR-5884 V02: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION.Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....

NUREG/CR-5954: EFFECT ON AGING ON PWR CHEMICAL AND VOLUME CONTROL SYSTEM.

NUREG/CR-6054: ESTIMATING PRESSURIZED WATER REACTOR DECOMMISSIONING COSTS. A User's Manual For The PWR Cost Estimating Computer Program (CECP) Software.

NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.

NUREG/CR-6266: ANALYSIS OF BORON DILUTION IN A FOUR-LOOP PWR.

Probabilistic Accident Consequence

NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Main Report.

NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Appendices A And B.

NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment.Appendices C,D,E,F,G,H.

Probabilistic Risk Assessment

NUREG-1464: NRC ITERATIVE PERFORMANCE ASSESSMENT PHASE 2.Development Of Capabilities For Review Of A Performance Assessment For A High-Level Waste Repository.

NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Verification And Validation (V&V) Manual.

NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS.Primer And User's Guides. Version 1.8.3 September 1994.

NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS.Reference Manuals.Version 1.8.3 September 1994.

NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.

NUREG/CR-6143 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF,UNIT 1.Summary Of Results.

NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF,UNIT 1.Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage.Main Report And Appendices.

NUREG/CR-6143 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF,UNIT 1.Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage.Supporting MELCOR Calculations.

Probabilistic Safety Assessment

NUREG/CR-6172: REVIEWING PSA-BASED ANALYSES TO MODIFY TECHNICAL SPECIFICATIONS AT NUCLEAR POWER PLANTS.

Proceeding

NUREG/CP-0140 V01: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING.Plenary Session, Advanced Instrumentation & Control Hardware & Software, Human Factors Research, IPE & PRA.

NUREG/CP-0140 V02: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING.Severe Accident Research, Thermal Hydraulic Research For Advanced Passive LWRs, High-Burnup Fuel Behavior.

NUREG/CP-0140 V03: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING.Primary Systems Integrity, Structural And Seismic Engineering, Aging Research, Products And Applications.

Process Instrumentation

NUREG/CR-6343: ON-LINE TESTING OF CALIBRATION OF PROCESS INSTRUMENTATION CHANNELS IN NUCLEAR POWER PLANTS.Phase II Final Report.

Program Performance

NUREG/CR-5758 V05: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports CY 1994.

Public Information Circular

NUREG-0725 R10: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL.

Public Utility Commission

NUREG/CR-5975 R01: INCENTIVE REGULATION OF INVESTOR-OWNED NUCLEAR POWER PLANTS BY PUBLIC UTILITY REGULATORS.

Pump

NUREG-1482: GUIDELINES FOR INSERVICE TESTING AT NUCLEAR POWER PLANTS.

NUREG/CR-5857: AGING OF TURBINE DRIVES FOR SAFETY-RELATED PUMPS IN NUCLEAR POWER PLANTS.

NUREG/CR-6089: DETECTION OF PUMP DEGRADATION.

Quality Assurance

NUREG/CR-6316 V01: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.

NUREG/CR-6316 V02: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Survey And Assessment Of Conventional Software Verification And Validation Methods.

NUREG/CR-6316 V03: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Survey And Documentation Of Expert System Verification And Validation Methodologies.

NUREG/CR-6316 V04: GUIDELINES FOR THE VERIFICATION AND VALIDATION EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Evaluation Of Knowledge Base Certification Methods.

NUREG/CR-6316 V05: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Rationale And Description Of V&V Guideline Packages And Procedures.

NUREG/CR-6316 V06: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Validation Scenarios.

NUREG/CR-6316 V07: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.User's Manual.

NUREG/CR-6316 V08: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Bibliography.

NUREG/CR-6324: QUALITY ASSURANCE FOR GAMMA KNIVES.

Quality Management

NUREG/CR-6276: A COMPILATION OF CURRENT REGULATIONS, STANDARDS, AND GUIDELINES IN REMOTE AFTERLOADING BRACTHYTHERAPY.

RELAP5 Computer Code

NUREG/CR-6150 V01: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Interface Theory.

NUREG/CR-6150 V02: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Damage Progression Model Theory.

NUREG/CR-6150 V03: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.User's Guide And Input Manual.

NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.

NUREG/CR-6150 V05: SCDAP/RELAP5/MOD 3.1 CODE MANUAL.Developmental Assessment.

NUREG/CR-6325: AN IMPLICIT STEADY-STATE INITIALIZATION PACKAGE FOR THE RELAP5 COMPUTER CODE.

RELAP5 Interface

NUREG/CR-6291 V01: NUCLEAR PLANT ANALYZER.Installation Manual.

NUREG/CR-6291 V02: NUCLEAR PLANT ANALYZER.Analyzer Reference Manual.

NUREG/CR-6291 V03: NUCLEAR PLANT ANALYZER.Computer Visual System Reference Manual.

NUREG/CR-6291 V04: NUCLEAR PLANT ANALYZER.Programmer's Manual.

RELAP5/MOD3 Computer Code

NUREG/CR-5535 V01: RELAP5/MOD3 CODE MANUAL.Code Structure, System Models, And Solution Methods.

NUREG/CR-5535 V02: RELAP5/MOD3 CODE MANUAL.User's Guide And Input Requirements.

NUREG/CR-5535 V04: RELAP5/MOD3 CODE MANUAL.Models And Correlations.

NUREG/CR-5535 V05 R1: RELAP5/MOD3 CODE MANUAL.User's Guideline.

Radiation Dose

NUREG/CP-0143: PROCEEDINGS OF THE THIRD INTERNATIONAL WORKSHOP ON THE IMPLEMENTATION OF ALARA AT NUCLEAR POWER PLANTS. Held At Hauppauge, Long Island, New York.
 NUREG/CR-3469 V08: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.
 NUREG/CR-6112: IMPACT OF REDUCED DOSE LIMITS ON NRC LICENSED ACTIVITIES. Major Issues In The Implementation Of ICRP/ NCRP Dose Limit Recommendations. Final Report.

Radiation Exposure

NUREG-0713 V15: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES. 1993. Twenty-Sixth Annual Report.
 NUREG-1535: INGESTION OF PHOSPHORUS-32 AT MASSACHUSETTS INSTITUTE OF TECHNOLOGY, CAMBRIDGE, MASSACHUSETTS, IDENTIFIED ON AUGUST 19, 1995.
 NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

Radiation Protection

NUREG/CR-3469 V08: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.

Radiation Survey

NUREG-1507 DRFT FC: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS. Draft Report For Comment.

Radiation Therapy

NUREG/CR-6323: RELATIVE RISK ANALYSIS IN REGULATING THE USE OF RADIATION-EMITTING MEDICAL DEVICES. A Preliminary Application.
 NUREG/CR-6324: QUALITY ASSURANCE FOR GAMMA KNIVES.

Radioactive Material

NUREG-0383 V01 R18: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Report Of NRC Approved Packages.
 NUREG-0383 V02 R18: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Certificates Of Compliance.
 NUREG-0383 V03 R15: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Report Of NRC Approved Quality Assurance Programs for Radioactive Materials Packages.
 NUREG/CP-0144 V01: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Summary.
 NUREG/CP-0144 V02: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Supporting Documents.
 NUREG/CR-2907 V13: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1992.
 NUREG/CR-2907 V14: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS.

Radioactive Material Safety Program

NUREG-1516 DRFT FC: MANAGEMENT OF RADIOACTIVE MATERIAL SAFETY PROGRAMS AT MEDICAL FACILITIES. Draft Report For Comment.

Radioactive Release

NUREG/CR-2850 V13: DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1991.

Radiological Criteria

NUREG/CR-6307: SUMMARY OF COMMENTS RECEIVED AT WORKSHOP ON USE OF A SITE SPECIFIC ADVISORY BOARD (SSAB) TO FACILITATE PUBLIC PARTICIPATION IN DECOMMISSIONING CASES.

Radiological Exposure

NUREG-1530: REASSESSMENT OF NRC'S DOLLAR PER PERSON-REM CONVERSION FACTOR POLICY.

Radiological Survey

NUREG-1506 DRFT FC: MEASUREMENT METHODS FOR RADIOLOGICAL SURVEYS IN SUPPORT OF NEW DECOMMISSIONING CRITERIA. Draft Report For Comment.

Radionuclide Characterization

NUREG/CR-6390: RADIOLOGICAL CHARACTERIZATION OF SPENT CONTROL ROD ASSEMBLIES.

Radionuclide Migration

NUREG/CR-6305: BLT-EC (BREACH, LEACH, TRANSPORT, AND EQUILIBRIUM CHEMISTRY), A FINITE-ELEMENT MODEL FOR ASSESSING THE RELEASE OF RADIONUCLIDES FROM LOW-LEVEL WASTE DISPOSAL UNITS. Background, Theory, And Model Description.

Reactive Transport

NUREG/CR-6347: MULTI-PHASE REACTIVE TRANSPORT THEORY.

Reactor

NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Final Report.
 NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Draft Report For Comment.

Reactor Accident

NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.
 NUREG/CR-6349: COST-BENEFIT CONSIDERATIONS IN REGULATORY ANALYSIS.

Reactor Component

NUREG/CR-6002: RISK-BASED MAINTENANCE MODELING. Prioritization Of Maintenance Importances And Quantification Of Maintenance Effectiveness.

Reactor Instrumentation

NUREG/CP-0140 V01: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING. Plenary Session, Advanced Instrumentation & Control Hardware & Software, Human Factors Research, IPE & PRA.

Reactor Operation

NUREG/CR-6172: REVIEWING PSA-BASED ANALYSES TO MODIFY TECHNICAL SPECIFICATIONS AT NUCLEAR POWER PLANTS.

Reactor Operator

NUREG-1122 R01: KNOWLEDGE AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS: PRESSURIZED WATER REACTORS.
 NUREG-1123 R01: KNOWLEDGE AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS: BOILING WATER REACTORS.

Reactor Pressure Vessel

NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.
 NUREG/CR-6273: BIAXIAL LOADING EFFECTS ON FRACTURE TOUGHNESS OF REACTOR PRESSURE VESSEL STEEL.
 NUREG/CR-6327: MODELS FOR EMBRITTLEMENT RECOVERY DUE TO ANNEALING OF REACTOR PRESSURE VESSEL STEELS.

Reactor Safety

NUREG/CP-0140 V01: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING. Plenary Session, Advanced Instrumentation & Control Hardware & Software, Human Factors Research, IPE & PRA.
 NUREG/CP-0140 V02: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING. Severe Accident Research, Thermal Hydraulic Research For Advanced Passive LWRs, High-Burnup Fuel Behavior.
 NUREG/CP-0140 V03: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING. Primary Sys-

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tems Integrity, Structural And Seismic Engineering, Aging Research, Products And Applications.
NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.
NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.
NUREG/CR-6311: EVALUATING PREDICTION UNCERTAINTY.

Reactor Safety Research

NUREG/CP-0148: TRANSACTIONS OF THE TWENTY-THIRD WATER REACTOR SAFETY INFORMATION MEETING.

Reconstitution Program

NUREG-1528: RECONSTITUTION OF THE MANUAL CHAPTER 2512 CONSTRUCTION INSPECTION PROGRAM FOR WATTS BAR UNIT 1.

Reelfoot Lake

NUREG/GR-0014: BALDCYPRESS TREE RING ELEMENTAL CONCENTRATIONS AT REELFOOT LAKE, TENNESSEE, FROM AD 1795 TO AD 1820.

Refueling Outage

NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Main Report And Appendices.
NUREG/CR-6143 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Supporting MELCOR Calculations.

Regulation

NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Final Report.
NUREG/CR-5975 R01: INCENTIVE REGULATION OF INVESTOR-OWNED NUCLEAR POWER PLANTS BY PUBLIC UTILITY REGULATORS.
NUREG/CR-6315: CANDU REACTORS, THEIR REGULATION IN CANADA, AND THE IDENTIFICATION OF RELEVANT NRC SAFETY ISSUES.
NUREG/CR-6323: RELATIVE RISK ANALYSIS IN REGULATING THE USE OF RADIATION-EMITTING MEDICAL DEVICES. A Preliminary Application.

Regulatory Agenda

NUREG-0936 V13 N03: NRC REGULATORY AGENDA. Semiannual Report, July-December 1994.
NUREG-0936 V14 N01: NRC REGULATORY AGENDA. Semiannual Report, January-June 1995.

Regulatory Analysis

NUREG-1530: REASSESSMENT OF NRC'S DOLLAR PER PERSON-REM CONVERSION FACTOR POLICY.
NUREG/CR-6349: COST-BENEFIT CONSIDERATIONS IN REGULATORY ANALYSIS.

Regulatory And Technical Report

NUREG-0304 V19 N03: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Third Quarter 1994, July-September.
NUREG-0304 V19 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1994.
NUREG-0304 V20 N01: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1995, January-March.
NUREG-0304 V20 N02: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Second Quarter 1995, April-June.

Regulatory Document

NUREG/CR-5973 R02: CODES AND STANDARDS AND OTHER GUIDANCE CITED IN REGULATORY DOCUMENTS.

Regulatory Impact Survey

NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.

Reinforced Concrete

NUREG/CR-6184: SEPARATE EFFECTS TESTING AND ANALYSES TO INVESTIGATE LINER TEARING OF THE 1:16-SCALE REINFORCED CONCRETE CONTAINMENT BUILDING.

Relay

NUREG/CR-6220: AN ASSESSMENT OF FIRE VULNERABILITY FOR AGED ELECTRICAL RELAYS.

Release Rate

NUREG/CR-6261: A SUMMARY OF ORNL FISSION PRODUCT RELEASE TESTS WITH RECOMMENDED RELEASE RATES AND DIFFUSION COEFFICIENTS.

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NUREG/CR-6192: AGING AND SERVICE WEAR OF SPRING-LOADED PRESSURE RELIEF VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.

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NUREG/CR-6276: A COMPILATION OF CURRENT REGULATIONS, STANDARDS, AND GUIDELINES IN REMOTE AFTERLOADING BRACHYTHERAPY.

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NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.
NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function And Task Analysis.
NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Supporting Analyses Of Human-System Interfaces, Procedures And Practices, Training And Organizational Practices And Procedures.

Report To Congress

NUREG-0090 V17 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. October-December 1994.
NUREG-0090 V18 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. January-March 1995.

Reporting Requirement

NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.

Requalification Examination

NUREG/CR-6355: A LIMITED ASSESSMENT OF THE ASEP HUMAN RELIABILITY ANALYSIS PROCEDURE USING SIMULATOR EXAMINATION RESULTS.

Resin

NUREG/CR-5229 V07: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1994. Annual Report.
NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Program. Test Results For Fiscal Years 1986, 1987, 1988, And 1989.

Risk Analysis

NUREG/CR-6311: EVALUATING PREDICTION UNCERTAINTY.
NUREG/CR-6323: RELATIVE RISK ANALYSIS IN REGULATING THE USE OF RADIATION-EMITTING MEDICAL DEVICES. A Preliminary Application.

Risk Assessment

NUREG/CP-0144 V01: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Summary.
NUREG/CP-0144 V02: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Supporting Documents.

Robust Laser Interferometer

NUREG/CR-6313 V01: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEM. Summary of Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.
NUREG/CR-6313 V02: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEMS. Supplemental Appendices Presenting Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.

Rules

NUREG-0936 V13 N03: NRC REGULATORY AGENDA. Semiannual Report, July-December 1994.
NUREG-0936 V14 N01: NRC REGULATORY AGENDA. Semiannual Report, January-June 1995.

SAPHIRE

NUREG/CR-6116 V06: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Graphical Evaluation Module (GEM) Reference Manual.
 NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Verification And Validation (V&V) Manual.
 NUREG/CR-6116 V10: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Data Loading Manual.

SCALE

NUREG/CR-0200 V1 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Control Modules.
 NUREG/CR-0200 V2P1R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Functional Modules F1-F8.
 NUREG/CR-0200 V2P2R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Functional Modules F9-F16.
 NUREG/CR-0200 V3 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Miscellaneous.

SCDAP

NUREG/CR-6150 V01: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. Interface Theory.
 NUREG/CR-6150 V02: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. Damage Progression Model Theory.
 NUREG/CR-6150 V03: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. User's Guide And Input Manual.
 NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. MATPRO-A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.
 NUREG/CR-6150 V05: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. Developmental Assessment.

Safeguards Summary Event List

NUREG-0525 V02 R03: SAFEGUARDS SUMMARY EVENT LIST (SSEL). January 1, 1990 Through December 31, 1994.

Safety Application

NUREG/CR-6287: MANAGEMENT CONCEPTS AND SAFETY APPLICATIONS FOR NUCLEAR FUEL FACILITIES.

Safety Criteria

NUREG/CR-6284: CRITICALITY SAFETY CRITERIA FOR LICENSE REVIEW OF LOW-LEVEL WASTE FACILITIES.

Safety Evaluation Report

NUREG-0847 S15: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)
 NUREG-0847 S16: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)
 NUREG-0847 S17: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)
 NUREG-0847 S18: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)
 NUREG-0847 S19: 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)

Safety Issues

NUREG-1435 S04: STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS. TMI Action Plan Requirements, Unresolved Safety Issues, Generic Safety Issues, Other Multiplant Action Issues.
 NUREG-1517: REPORT OF THE SOUTH TEXAS PROJECT ALLEGATIONS REVIEW TEAM.
 NUREG/CR-6315: CANDU REACTORS, THEIR REGULATION IN CANADA, AND THE IDENTIFICATION OF RELEVANT NRC SAFETY ISSUES.

Safety Management

NUREG/CR-6287: MANAGEMENT CONCEPTS AND SAFETY APPLICATIONS FOR NUCLEAR FUEL FACILITIES.

Safety Research

NUREG-1266 V09: NRC SAFETY RESEARCH IN SUPPORT OF REGULATION - FY 1994.

Safety Standard

NUREG/CR-6349: COST-BENEFIT CONSIDERATIONS IN REGULATORY ANALYSIS.

Safety Testing

NUREG/CR-6074 V04: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING. Technical Report On The Findings Of Task 4. Investigation Of Sealed Source for Paper Mill Digester.
 NUREG/CR-6074 V05: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING. Technical Report On The Findings Of Task 4. Investigation Of Failed Radioactive Stainless Steel Troxler Gauges.

Safety-Related System

NUREG/CR-6016: AGING AND SERVICE WEAR OF AIR-OPERATED VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.
 NUREG/CR-6192: AGING AND SERVICE WEAR OF SPRING-LOADED PRESSURE RELIEF VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.

Sealed Source

NUREG/CR-6074 V04: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING. Technical Report On The Findings Of Task 4. Investigation Of Sealed Source for Paper Mill Digester.
 NUREG/CR-6074 V05: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING. Technical Report On The Findings Of Task 4. Investigation Of Failed Radioactive Stainless Steel Troxler Gauges.

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NUREG/CR-6368: EXPERIMENTAL INVESTIGATION OF SEDIMENTATION OF LOCA-GENERATED FIBROUS DEBRIS AND SLUDGE IN BWR SUPPRESSION POOLS.

Seismic

NUREG/CR-6144 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Summary Of Results.

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NUREG/CR-6240: APPLICATION OF BOUNDING SPECTRA TO SEISMIC DESIGN OF PIPING BASED ON THE PERFORMANCE OF ABOVE GROUND PIPING IN POWER PLANTS SUBJECTED TO STRONG MOTION EARTHQUAKES.

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NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
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 NUREG/CR-6311: EVALUATING PREDICTION UNCERTAINTY.

Severe Accident

NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS.
 NUREG-1519: SURFACE INTERACTIONS OF CESIUM AND BORIC ACID WITH STAINLESS STEEL.
 NUREG/CP-0140 V02: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING. Severe Accident Research, Thermal Hydraulic Research For Advanced Passive LWRs, High-Burnup Fuel Behavior.
 NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.
 NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS. Primer And User's Guides. Version 1.8.3 September 1994.
 NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS. Reference Manuals. Version 1.8.3 September 1994.
 NUREG/CR-6143 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Summary Of Results.
 NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Main Report And Appendices.
 NUREG/CR-6143 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Supporting MELCOR Calculations.

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NUREG/CR-6154 V02: EXPERIMENTAL RESULTS FROM CONTAINMENT PIPING BELLOWS SUBJECTED TO SEVERE ACCIDENT CONDITIONS. Results From Bellows Tested In Corroded Conditions.
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Shipment

NUREG-0725 R10: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL.

Shipping Cask

NUREG/CR-5657: AUTOCASK (AUTOMATIC GENERATION OF 3-D CASK MODELS). A Microcomputer Based System For Shipping Cask Design Review Analysis.
NUREG/CR-6322: BUCKLING ANALYSIS OF SPENT FUEL BASKET.

Shippingport Reactor

NUREG/CR-6275: MECHANICAL PROPERTIES OF THERMALLY AGED CAST STAINLESS STEELS FROM SHIPPINGPORT REACTOR COMPONENTS.

Short Crack

NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING WELDS. Seventh Program Report March 1993 - December 1994.

Shutdown Operation

NUREG/CR-6143 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Summary Of Results.
NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Main Report And Appendices.
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NUREG/CR-6144 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Summary Of Results.
NUREG/CR-6144 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Evaluation Of Severe Accident Risk During Mid-Loop Operations. Main Report.
NUREG/CR-6144 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Evaluation Of Severe Accident Risk During Mid-Loop Operations. Appendices.

Simulator

NUREG-1527: NRC'S OBJECT-ORIENTED SIMULATOR INSTRUCTOR STATION.

Simulator Examination

NUREG/CR-6355: A LIMITED ASSESSMENT OF THE ASEP HUMAN RELIABILITY ANALYSIS PROCEDURE USING SIMULATOR EXAMINATION RESULTS.

Site Specific Advisory Board

NUREG/CR-6307: SUMMARY OF COMMENTS RECEIVED AT WORKSHOP ON USE OF A SITE SPECIFIC ADVISORY BOARD (SSAB) TO FACILITATE PUBLIC PARTICIPATION IN DECOMMISSIONING CASES.

Sludge

NUREG/CR-6368: EXPERIMENTAL INVESTIGATION OF SEDIMENTATION OF LOCA-GENERATED FIBROUS DEBRIS AND SLUDGE IN BWR SUPPRESSION POOLS.

Software

NUREG/CP-0145: WORKSHOP ON DEVELOPING SAFE SOFTWARE. Held At Hotel Del Coronado, San Diego, CA, July 22-23, 1992.
NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Main Report.
NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Appendices A-D.
NUREG/CR-6316 V01: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.
NUREG/CR-6316 V02: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Assessment Of Conventional Software Verification And Validation Methods.
NUREG/CR-6316 V03: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Documentation Of Expert System Verification And Validation Methodologies.
NUREG/CR-6316 V04: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Evaluation Of Knowledge Base Certification Methods.
NUREG/CR-6316 V05: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Rationale And Description Of V&V Guideline Packages And Procedures.
NUREG/CR-6316 V06: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Validation Scenarios.
NUREG/CR-6316 V07: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. User's Manual.
NUREG/CR-6316 V08: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Bibliography.

Solid Waste Disposal

NUREG/CR-2907 V13: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1992.
NUREG/CR-2907 V14: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS.

Solidification

NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990, 1991, 1992, And 1993.

Source Term

NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS.

South Texas Project

NUREG-1517: REPORT OF THE SOUTH TEXAS PROJECT ALLEGATIONS REVIEW TEAM.

Special Nuclear Material

NUREG-1065 R02: ACCEPTABLE STANDARD FORMAT AND CONTENT FOR THE FUNDAMENTAL NUCLEAR MATERIAL CONTROL (FNMC) PLAN REQUIRED FOR LOW-ENRICHED URANIUM FACILITIES.

Spent Fuel

NUREG-1323 R01: LICENSE APPLICATION REVIEW PLAN FOR A GEOLOGIC REPOSITORY FOR SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE. Draft Review Plan.

Spent Fuel Basket

NUREG/CR-6322: BUCKLING ANALYSIS OF SPENT FUEL BASKET.

Stainless Steel

NUREG-1519: SURFACE INTERACTIONS OF CESIUM AND BORIC ACID WITH STAINLESS STEEL.
NUREG/CR-6251: STAINLESS STEEL SUBMERGED ARC WELD FUSION LINE TOUGHNESS.
NUREG/CR-6275: MECHANICAL PROPERTIES OF THERMALLY AGED CAST STAINLESS STEELS FROM SHIPPINGPORT REACTOR COMPONENTS.

Standard

NUREG/CR-6385: COMPARISONS OF ANS, ASME, AWS, AND NFPA STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN, NUREG-0800, AND RELATED DOCUMENTS.

NUREG/CR-6386: COMPARISONS OF ANSI STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN, NUREG-0800, AND RELATED DOCUMENTS.

Standard Review Plan

NUREG/CR-5973 R02: CODES AND STANDARDS AND OTHER GUIDANCE CITED IN REGULATORY DOCUMENTS.
NUREG/CR-6382: COMPARISONS OF ASTM STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN, NUREG-0800, AND RELATED DOCUMENTS.
NUREG/CR-6385: COMPARISONS OF ANS, ASME, AWS, AND NFPA STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN, NUREG-0800, AND RELATED DOCUMENTS.
NUREG/CR-6386: COMPARISONS OF ANSI STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN, NUREG-0800, AND RELATED DOCUMENTS.

Standard Technical Specifications

NUREG-1430 V01 R01: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Specifications.
NUREG-1430 V02 R01: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Bases (Sections 2.0 - 3.3).
NUREG-1430 V03 R01: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Bases (Sections 3.4 - 3.9).
NUREG-1431 V01 R01: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Specifications.
NUREG-1431 V02 R01: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Bases (Sections 2.0 - 3.3).
NUREG-1431 V03 R01: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Bases (Sections 3.4 - 3.9).
NUREG-1432 V01 R01: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS. Specifications.
NUREG-1432 V02 R01: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS. Bases (Sections 2.0 - 3.3).
NUREG-1432 V03 R01: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS. Bases (Sections 3.4 - 3.9).
NUREG-1433 V01 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS BWR/4. Specifications.
NUREG-1433 V02 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS BWR/4. Bases (Sections 2.0 - 3.3).
NUREG-1433 V03 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS BWR/4. Bases (Sections 3.4 - 3.10).
NUREG-1434 V01 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6. Specifications.
NUREG-1434 V02 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6. Bases (Sections 2.0 - 3.3).
NUREG-1434 V03 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6. Bases 3.4 - 3.10).

Steady-State Initialization

NUREG/CR-6325: AN IMPLICIT STEADY-STATE INITIALIZATION PACKAGE FOR THE RELAP5 COMPUTER CODE.

Storage Cask

NUREG/CR-6242: CASKS (COMPUTER ANALYSIS OF STORAGE CASKS): A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR STORAGE CASK DESIGN REVIEW. User's Manual To Version 1b (Including Program Reference).

Strainer Blockage

NUREG/CR-6224: PARAMETRIC STUDY OF THE POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS.

Stress Corrosion Cracking

NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, October 1993 - March 1994.
NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, April-September 1994.

Stress Triaxiality

NUREG/CR-6259: CONSTRAINT EFFECTS ON FRACTURE INITIATION LOADS IN HSST WIDE-PLATE TESTS.

Structural Analysis

NUREG/CR-5657: AUTOCASK (AUTOMATIC GENERATION OF 3-D CASK MODELS). A Microcomputer Based System For Shipping Cask Design Review Analysis.

Structural Codes And Standards

NUREG/CR-6358 V01: ASSESSMENT OF UNITED STATES INDUSTRY STRUCTURAL CODES AND STANDARDS FOR APPLICATION TO ADVANCED NUCLEAR POWER REACTORS. Final Report.
NUREG/CR-6358 V02: ASSESSMENT OF UNITED STATES INDUSTRY STRUCTURAL CODES AND STANDARDS FOR APPLICATION TO ADVANCED NUCLEAR POWER REACTORS. Appendices.

Suppression Pool

NUREG/CR-6368: EXPERIMENTAL INVESTIGATION OF SEDIMENTATION OF LOCA-GENERATED FIBROUS DEBRIS AND SLUDGE IN BWR SUPPRESSION POOLS.

Surface Interaction

NUREG-1519: SURFACE INTERACTIONS OF CESIUM AND BORIC ACID WITH STAINLESS STEEL.

Surveillance Requirement

NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.

Surveillance System

NUREG/CR-6420: SELF-MONITORING SURVEILLANCE SYSTEM FOR PRESTRESSING TENDONS.

Survey

NUREG/CR-6239 V01: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Main Report.
NUREG/CR-6239 V02: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Appendices.

Survey Design

NUREG-1505 DRFT FC: A NONPARAMETRIC STATISTICAL METHODOLOGY FOR THE DESIGN AND ANALYSIS OF FINAL STATUS DECOMMISSIONING SURVEYS. Draft Report For Comment.
NUREG-1506 DRFT FC: MEASUREMENT METHODS FOR RADIOLOGICAL SURVEYS IN SUPPORT OF NEW DECOMMISSIONING CRITERIA. Draft Report For Comment.
NUREG-1507 DRFT FC: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS. Draft Report For Comment.

System Safety

NUREG/CP-0145: WORKSHOP ON DEVELOPING SAFE SOFTWARE. Held At Hotel Del Coronado, San Diego, CA, July 22-23, 1992.

Systems Integrity

NUREG/CP-0140 V03: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING. Primary Systems Integrity, Structural And Seismic Engineering, Aging Research, Products And Applications.

TLD

NUREG-0837 V14 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. October-December 1994.
NUREG-0837 V15 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. January-March 1995.
NUREG-0837 V15 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. April-June 1995.
NUREG-0837 V15 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. July-September 1995.

TMI Action Plan

NUREG-1435 S04: STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS. TMI Action Plan Requirements, Unresolved Safety Issues, Generic Safety Issues, Other Multiplant Action Issues.

Tabletop

NUREG-1514: GUIDANCE FOR A LARGE TABLETOP EXERCISE FOR A NUCLEAR POWER PLANT.

Task Procedure

NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Supporting Analyses Of Human-System Interfaces, Procedures And Practices, Training And Organizational Practices And Procedures.

Technical Specifications

NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.
NUREG/CR-6172: REVIEWING PSA-BASED ANALYSES TO MODIFY TECHNICAL SPECIFICATIONS AT NUCLEAR POWER PLANTS.

Teletherapy

NUREG/CR-6277 V01: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Identification Of Problems And Alternative Approaches.
NUREG/CR-6277 V02: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Function And Task Analysis.

84 Subject Index

NUREG/CR-6277 V03: HUMAN FACTORS EVALUATION OF TELETHERAPY.Human-System Interfaces And Procedures.
NUREG/CR-6277 V04: HUMAN FACTORS EVALUATION OF TELETHERAPY.Training And Organizational Analysis.
NUREG/CR-6277 V05: HUMAN FACTORS EVALUATION OF TELETHERAPY.Literature Review.

Tendon

NUREG/CR-6420: SELF-MONITORING SURVEILLANCE SYSTEM FOR PRESTRESSING TENDONS.

Therapy Misadministration

NUREG-1535: INGESTION OF PHOSPHORUS-32 AT MASSACHUSETTS INSTITUTE OF TECHNOLOGY,CAMBRIDGE,MASSACHUSETTS,IDENTIFIED ON AUGUST 19, 1995.

Thermal Aging

NUREG/CR-6275: MECHANICAL PROPERTIES OF THERMALLY AGED CAST STAINLESS STEELS FROM SHIPPINGPORT REACTOR COMPONENTS.

Thermal Binding

NUREG/CP-0146: PROCEEDINGS OF THE WORKSHOP ON GATE VALVE PRESSURE LOCKING AND THERMAL BINDING.

Thermal Power Plant

NUREG/CR-6239 V01: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS.Main Report.
NUREG/CR-6239 V02: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Appendices.

Thermal-Hydraulic

NUREG/CP-0140 V02: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING.Severe Accident Research, Thermal Hydraulic Research For Advanced Passive LWRs, High-Burnup Fuel Behavior.
NUREG/CP-0142 V01: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 1-5.
NUREG/CP-0142 V02: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 6-11.
NUREG/CP-0142 V03: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 12-16.
NUREG/CP-0142 V04: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7).Sessions 17-24.
NUREG/CR-5535 V01: RELAP5/MOD3 CODE MANUAL.Code Structure, System Models, And Solution Methods.
NUREG/CR-5535 V02: RELAP5/MOD3 CODE MANUAL.User's Guide And Input Requirements.
NUREG/CR-5535 V04: RELAP5/MOD3 CODE MANUAL.Models And Correlations.
NUREG/CR-5535 V05 R1: RELAP5/MOD3 CODE MANUAL.User's Guideline.
NUREG/CR-6004: PROBABILISTIC PIPE FRACTURE EVALUATIONS FOR LEAK-RATE-DETECTION APPLICATIONS.
NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS.Primer And User's Guides. Version 1.8.3 September 1994.
NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS.Reference Manuals.Version 1.8.3 September 1994.

Thermocouple

NUREG/CR-6334: NEW SENSOR FOR MEASUREMENT OF LOW AIR FLOW VELOCITY.Phase I Final Report.

Thermohydraulic

NUREG/CR-6257: CANDU 3 TRANSIENT ANALYSIS USING ATOMIC ENERGY OF CANADA LTD CODES.

Thermohydrology

NUREG/CR-6348: THERMALLY DRIVEN MOISTURE REDISTRIBUTION IN PARTIALLY SATURATED POROUS MEDIA.

Thermoluminescent Dosimeter

NUREG-0837 V14 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. October-December 1994.
NUREG-0837 V15 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. January-March 1995.
NUREG-0837 V15 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report.April-June 1995.

NUREG-0837 V15 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. July-September 1995.

Title List

NUREG-0540 V16 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.November 1-30, 1994.
NUREG-0540 V16 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.December 1-31, 1994.
NUREG-0540 V17 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31, 1995.
NUREG-0540 V17 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. February 1-28, 1995.
NUREG-0540 V17 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.March 1-31, 1995.
NUREG-0540 V17 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30, 1995.
NUREG-0540 V17 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.May 1-31, 1995.
NUREG-0540 V17 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.June 1-30, 1995.
NUREG-0540 V17 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.July 1-31, 1995.
NUREG-0540 V17 N08: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.August 1-31, 1995.
NUREG-0540 V17 N09: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.September 1-30,1995.
NUREG-0540 V17 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. October 1-31,1995.

Topical Report

NUREG-0390 V09 N01: TOPICAL REPORT REVIEW STATUS.

Transient Analysis

NUREG/CR-5535 V01: RELAP5/MOD3 CODE MANUAL.Code Structure, System Models, And Solution Methods.
NUREG/CR-5535 V02: RELAP5/MOD3 CODE MANUAL.User's Guide And Input Requirements.
NUREG/CR-5535 V04: RELAP5/MOD3 CODE MANUAL.Models And Correlations.
NUREG/CR-5535 V05 R1: RELAP5/MOD3 CODE MANUAL.User's Guideline.
NUREG/CR-6257: CANDU 3 TRANSIENT ANALYSIS USING ATOMIC ENERGY OF CANADA LTD CODES.

Transportation

NUREG-0383 V01 R18: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Report Of NRC Approved Packages.
NUREG-0383 V02 R18: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Certificates Of Compliance.
NUREG-0383 V03 R15: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Report Of NRC Approved Quality Assurance Programs for Radioactive Materials Packages.

Tree Ring

NUREG/GR-0014: BALDCYPRESS TREE RING ELEMENTAL CONCENTRATIONS AT REELFOOT LAKE,TENNESSEE,FROM AD 1795 TO AD 1820.

Troxler Gauge

NUREG/CR-6074 V05: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING.Technical Report On The Findings Of Task 4.Investigation Of Failed Radioactive Stainless Steel Troxler Gauges.

Turbine Drive

NUREG/CR-5857: AGING OF TURBINE DRIVES FOR SAFETY-RELATED PUMPS IN NUCLEAR POWER PLANTS.

Turbine-Generator

NUREG-1275 V11: OPERATING EXPERIENCE FEEDBACK REPORT - TURBINE-GENERATOR OVERSPEED PROTECTION SYSTEM.Commercial Power Reactors.

Two-Phase Flow

NUREG/CR-6114 V02: AUXILIARY ANALYSES IN SUPPORT OF PERFORMANCE ASSESSMENT OF A HYPOTHETICAL LOW-LEVEL WASTE FACILITY.Two-Phase Flow And Contaminant Transport In Unsaturated Soils With Application To Low-Level Radioactive Waste Disposal.

Uncertainty Analysis

- NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Main Report.
 NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Appendices A And B.
 NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment.Appendices C,D,E,F,G,H.

Unsaturated Rock

- NUREG/CR-6356: HYDRAULIC CHARACTERIZATION OF HYDROTHERMALLY ALTERED NOPAL TUFF.

Unsaturated Soil

- NUREG/CR-6114 V02: AUXILIARY ANALYSES IN SUPPORT OF PERFORMANCE ASSESSMENT OF A HYPOTHETICAL LOW-LEVEL WASTE FACILITY.Two-Phase Flow And Contaminant Transport In Unsaturated Soils With Application To Low-Level Radioactive Waste Disposal.

Uranium

- NUREG/CR-6328: ADEQUACY OF THE 123-GROUP CROSS-SECTION LIBRARY FOR CRITICALITY ANALYSES OF WATER-MODERATED URANIUM SYSTEMS.

VITAMIN-B6

- NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.

Validation

- NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS.Main Report.
 NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS.Appendices A-D.
 NUREG/CR-6316 V01: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.
 NUREG/CR-6316 V02: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Survey And Assessment Of Conventional Software Verification And Validation Methods.
 NUREG/CR-6316 V03: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Survey And Documentation Of Expert System Verification And Validation Methodologies.
 NUREG/CR-6316 V04: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Evaluation Of Knowledge Base Certification Methods.
 NUREG/CR-6316 V05: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Rationale And Description Of V&V Guideline Packages And Procedures.
 NUREG/CR-6316 V06: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Validation Scenarios.
 NUREG/CR-6316 V07: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.User's Manual.
 NUREG/CR-6316 V08: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Bibliography.

Validity Limit

- NUREG/CR-6264 V01: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION.An Assessment Of The J(M) Parameter.
 NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION.A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions.

Value-Impact Analysis

- NUREG-1530: REASSESSMENT OF NRC'S DOLLAR PER PERSON-REM CONVERSION FACTOR POLICY.

Valve

- NUREG-1482: GUIDELINES FOR INSERVICE TESTING AT NUCLEAR POWER PLANTS.

Vendor Inspection

- NUREG-0040 V18 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1994.(White Book)

- NUREG-0040 V19 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1995.(White Book)

- NUREG-0040 V19 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,April-June 1995.(White Book)

- NUREG-0040 V19 N03: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,July-September 1995.(White Book)

Verification

- NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS.Main Report.
 NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS.Appendices A-D.
 NUREG/CR-6316 V01: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.
 NUREG/CR-6316 V02: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Survey And Assessment Of Conventional Software Verification And Validation Methods.
 NUREG/CR-6316 V03: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Survey And Documentation Of Expert System Verification And Validation Methodologies.
 NUREG/CR-6316 V04: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Evaluation Of Knowledge Base Certification Methods.
 NUREG/CR-6316 V05: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Rationale And Description Of V&V Guideline Packages And Procedures.
 NUREG/CR-6316 V06: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Validation Scenarios.
 NUREG/CR-6316 V07: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.User's Manual.
 NUREG/CR-6316 V08: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.Bibliography.

Vessel

- NUREG/CR-6260: APPLICATION OF NUREG/CR-5999 INTERIM FATIGUE CURVES TO SELECTED NUCLEAR POWER PLANT COMPONENTS.

Vibration Measurement System

- NUREG/CR-6313 V01: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEM.Summary of Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.
 NUREG/CR-6313 V02: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEMS.Supplemental Appendices Presenting Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.

Violation

- NUREG-1525: ASSESSMENT OF THE NRC ENFORCEMENT PROGRAM.

Visual Display

- NUREG/CR-6291 V01: NUCLEAR PLANT ANALYZER.Installation Manual.
 NUREG/CR-6291 V02: NUCLEAR PLANT ANALYZER.Analyzer Reference Manual.
 NUREG/CR-6291 V03: NUCLEAR PLANT ANALYZER.Computer Visual System Reference Manual.
 NUREG/CR-6291 V04: NUCLEAR PLANT ANALYZER.Programmer's Manual.

Waste Burial

- NUREG-1307 R05: REPORT ON WASTE BURIAL CHARGES.Escalation Of Decommissioning Waste Disposal Costs At Low-Level Waste Burial Facilities.

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- NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Progress Report Of Field Experiments At A Humid Region Site,Beltsville,Maryland.

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NUREG/CR-6251: STAINLESS STEEL SUBMERGED ARC WELD FUSION LINE TOUGHNESS.

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NUREG/CR-6259: CONSTRAINT EFFECTS ON FRACTURE INITIATION LOADS IN HSST WIDE-PLATE TESTS.

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NUREG/CR-6420: SELF-MONITORING SURVEILLANCE SYSTEM FOR PRESTRESSING TENDONS.

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NUREG-1464: NRC ITERATIVE PERFORMANCE ASSESSMENT PHASE 2. Development Of Capabilities For Review Of A Performance Assessment For A High-Level Waste Repository.

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ADVISORY COMMITTEE(S)

ADVISORY COMMITTEE ON NUCLEAR WASTE
NUREG-1423 V05: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON NUCLEAR WASTE, July 1993 - June 1995.
ACRS - ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUREG-1125 V16: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, 1994 Annual.

ATOMIC SAFETY BOARD(S) & PANEL(S)

ATOMIC SAFETY & LICENSING BOARD PANEL
NUREG-1363 V06: ATOMIC SAFETY AND LICENSING BOARD PANEL BIENNIAL REPORT, Fiscal Years 1993 - 1994.

OFFICE OF EXECUTIVE DIRECTOR FOR OPERATIONS (EDO)

REGION 1 (POST 820201)
NUREG-0837 V14 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK, Progress Report, October-December 1994.
NUREG-0837 V15 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK, Progress Report, January-March 1995.
NUREG-0837 V15 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK, Progress Report, April-June 1995.
NUREG-0837 V15 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK, Progress Report, July-September 1995.

REGION 2 (POST 820201)
NUREG-1528: RECONSTITUTION OF THE MANUAL CHAPTER 2512 CONSTRUCTION INSPECTION PROGRAM FOR WATTS BAR UNIT 1.

WALNUT CREEK FIELD OFC, R4 (POST 940404)
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OFC OF ENFORCEMENT (POST 870413)
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NUREG-0940 V13 N4 P2: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MEDICAL LICENSEES, Quarterly Progress Report, October-December 1994.

NUREG-0940 V13 N4 P3: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MATERIAL LICENSEES (NON-MEDICAL), Quarterly Progress Report, October-December 1994.

NUREG-0940 V14 N1 P1: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, REACTOR LICENSEES, Quarterly Progress Report, January-March 1995.

NUREG-0940 V14 N1 P2: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, MEDICAL LICENSEES, Quarterly Progress Report, January-March 1995.

NUREG-0940 V14 N1 P3: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, MATERIAL LICENSEES (NON-MEDICAL), Quarterly Progress Report, January-March 1995.

NUREG-0940 V14 N2 P1: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, INDIVIDUAL ACTIONS, Quarterly Progress Report, April-June 1995.

NUREG-0940 V14 N2 P2: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED, REACTOR LICENSEES, Quarterly Progress Report, April-June 1995.

NUREG-0940 V14 N2 P3: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MATERIAL LICENSEES, Quarterly Progress Report, April-June 1995.

NUREG-1525: ASSESSMENT OF THE NRC ENFORCEMENT PROGRAM.

NUREG-1600: GENERAL STATEMENT OF POLICY AND PROCEDURE FOR NRC ENFORCEMENT ACTIONS, Enforcement Policy.

OFC OF INVESTIGATIONS (POST 880201)
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OFC OF PERSONNEL (POST 870413)
NUREG-0325 R18: U.S. NUCLEAR REGULATORY COMMISSION ORGANIZATION CHARTS AND FUNCTIONAL STATEMENTS, July 23, 1995.

EDO - OFFICE OF ADMINISTRATION (PRE 870413 & POST 890205)

OFFICE OF ADMINISTRATION, DIRECTOR (POST 940714)
NUREG-1145 V11: U.S. NUCLEAR REGULATORY COMMISSION 1994 ANNUAL REPORT.

DIVISION OF FREEDOM OF INFORMATION & PUBLICATIONS SERVICES (POST 940714)

NUREG-0304 V19 N03: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL), Compilation For Third Quarter 1994, July-September.

NUREG-0304 V19 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL), Annual Compilation For 1994.

NUREG-0304 V20 N01: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL), Compilation For First Quarter 1995, January-March.

NUREG-0304 V20 N02: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL), Compilation For Second Quarter 1995, April-June.

NUREG-0540 V16 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE, November 1-30, 1994.

NUREG-0540 V16 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE, December 1-31, 1994.

NUREG-0540 V17 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE, January 1-31, 1995.

NUREG-0540 V17 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE, February 1-28, 1995.

NUREG-0540 V17 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE, March 1-31, 1995.

NUREG-0540 V17 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE, April 1-30, 1995.

NUREG-0540 V17 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE, May 1-31, 1995.

NUREG-0540 V17 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE, June 1-30, 1995.

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NUREG-0540 V17 N08: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE, August 1-31, 1995.

NUREG-0540 V17 N09: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE, September 1-30, 1995.

NUREG-0540 V17 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE, October 1-31, 1995.

NUREG-0750 V40: NUCLEAR REGULATORY COMMISSION ISSUANCES, Opinions And Decisions Of The Nuclear Regulatory Commission With Selected Orders, July-December 1994.

NUREG-0750 V40 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES, July-December 1994.

NUREG-0750 V40 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1994, Pages 169-318.

NUREG-0750 V40 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1994, Pages 319-387.

NUREG-0750 V41 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES, January-March 1995.

NUREG-0750 V41 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES, January-June 1995.

NUREG-0750 V41 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1995, Pages 1-69.

NUREG-0750 V41 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1995, Pages 71-178.

NUREG-0750 V41 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1995, Pages 179-243.

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NUREG-0750 V41 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1995, Pages 321-380.

NUREG-0750 V41 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1995, Pages 381-496.

NUREG-0750 V42 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES, July-September 1995.

NUREG-0750 V42 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1995. Pages 1-45.
 NUREG-0750 V42 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1995. Pages 47-97.
 NUREG-0750 V42 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1995. Pages 99-110.
 NUREG-0750 V42 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1995. Pages 111-180.
 NUREG-0936 V13 N03: NRC REGULATORY AGENDA. Semiannual Report, July-December 1994.
 NUREG-0936 V14 N01: NRC REGULATORY AGENDA. Semiannual Report, January-June 1995.

EDO - OFFICE OF THE CONTROLLER (PRE 820418 & POST 890205)

OFFICE OF THE CONTROLLER (POST 890205)
 NUREG-1470 V04: FINANCIAL STATEMENT FOR FISCAL YEAR 1994.
 DIVISION OF BUDGET & ANALYSIS (POST 890205)
 NUREG-1100 V11: BUDGET ESTIMATES. Fiscal Years 1996-1997.
 NUREG-1350 V07: NUCLEAR REGULATORY COMMISSION INFORMATION DIGEST. 1995 Edition.

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA

OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA, DIRECTOR
 NUREG-0090 V17 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. July-September 1994.
 NUREG-0090 V17 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. October-December 1994.
 NUREG-0090 V18 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. January-March 1995.
 NUREG-1272 V08 N02: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA. 1993 Annual Report - Nuclear Materials.
 NUREG-1514: GUIDANCE FOR A LARGE TABLETOP EXERCISE FOR A NUCLEAR POWER PLANT.
 TECHNICAL TRAINING CENTER
 NUREG-1527: NRC'S OBJECT-ORIENTED SIMULATOR INSTRUCTOR STATION.
 DIVISION OF SAFETY PROGRAMS (POST 870413)
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 NUREG/CP-0146: PROCEEDINGS OF THE WORKSHOP ON GATE VALVE PRESSURE LOCKING AND THERMAL BINDING.

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

OFFICE OF INFORMATION RESOURCES MANAGEMENT (POST 890205)
 NUREG-0020 V19: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of December 31, 1994. (Gray Book I)

EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS

OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
 NUREG-0383 V01 R18: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Report Of NRC Approved Packages.
 NUREG-0383 V02 R18: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Certificates Of Compliance.
 NUREG-0383 V03 R15: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Report Of NRC Approved Quality Assurance Programs for Radioactive Materials Packages.
 NUREG-0430 V14: LICENSED FUEL FACILITY STATUS REPORT. Inventory Difference Data. July 1, 1993 - June 30, 1994. (Gray Book II)
 NUREG-1464: NRC ITERATIVE PERFORMANCE ASSESSMENT PHASE 2. Development Of Capabilities For Review Of A Performance Assessment For A High-Level Waste Repository.
 DIVISION OF INDUSTRIAL & MEDICAL NUCLEAR SAFETY (POST 870729)
 NUREG-0725 R10: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL.
 NUREG-1516 DRFT FC: MANAGEMENT OF RADIOACTIVE MATERIAL SAFETY PROGRAMS AT MEDICAL FACILITIES. Draft Report For Comment.
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 NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.
 NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function And Task Analysis.

DIVISION OF FUEL CYCLE SAFETY & SAFEGUARDS (POST 930207)
 NUREG-1065 R02: ACCEPTABLE STANDARD FORMAT AND CONTENT FOR THE FUNDAMENTAL NUCLEAR MATERIAL CONTROL (FNMC) PLAN REQUIRED FOR LOW-ENRICHED URANIUM FACILITIES.

NUREG-1280 R01: STANDARD FORMAT AND CONTENT ACCEPTANCE CRITERIA FOR THE MATERIAL CONTROL AND ACCOUNTING (MC&A) REFORM AMENDMENT.

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NUREG-0525 V02 R03: SAFEGUARDS SUMMARY EVENT LIST (SSEL). January 1, 1990 Through December 31, 1994.
 NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Supporting Analyses Of Human-System Interfaces, Procedures And Practices, Training And Organizational Practices And Procedures.
 NUREG/CR-6277 V01: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Identification Of Problems And Alternative Approaches.
 NUREG/CR-6277 V02: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Function And Task Analysis.
 NUREG/CR-6277 V03: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Human-System Interfaces And Procedures.
 NUREG/CR-6277 V04: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Training And Organizational Analysis.
 NUREG/CR-6277 V05: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Literature Review.

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NUREG-1323 R01: LICENSE APPLICATION REVIEW PLAN FOR A GEOLOGIC REPOSITORY FOR SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE. Draft Review Plan.
 NUREG-1444 S01: SITE DECOMMISSIONING MANAGEMENT PLAN.

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OFFICE OF THE GENERAL COUNSEL (POST 860701)
 NUREG-0980 V01 N03: NUCLEAR REGULATORY LEGISLATION. 103D Congress.
 NUREG-0980 V02 N03: NUCLEAR REGULATORY LEGISLATION. 103D Congress.
 OFFICE OF THE INSPECTOR GENERAL (POST 890417)
 NUREG-1415 V07 N02: OFFICE OF THE INSPECTOR GENERAL. Semiannual Report To Congress, October 1, 1994 - March 31, 1995.
 NUREG-1415 V08 N01: OFFICE OF THE INSPECTOR GENERAL. Semiannual Report To Congress, April 1, 1995 - September 30, 1995.
 NRC - NO DETAILED AFFILIATION GIVEN
 NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment. Appendices C,D,E,F,G,H.

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OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 941217)
 NUREG-1266 V09: NRC SAFETY RESEARCH IN SUPPORT OF REGULATION - FY 1994.
 NUREG-1464: NRC ITERATIVE PERFORMANCE ASSESSMENT PHASE 2. Development Of Capabilities For Review Of A Performance Assessment For A High-Level Waste Repository.
 DIVISION OF ENGINEERING TECHNOLOGY (POST 941217)
 NUREG-0933 S18: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
 NUREG-0933 S19: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
 DIVISION OF REGULATORY APPLICATIONS (POST 941217)
 NUREG-0713 V15: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES. 1993. Twenty-Sixth Annual Report.
 NUREG-1307 R05: REPORT ON WASTE BURIAL CHARGES. Escalation Of Decommissioning Waste Disposal Costs At Low-Level Waste Burial Facilities.
 NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Final Report.
 NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Draft Report For Comment.
 NUREG-1505 DRFT FC: A NONPARAMETRIC STATISTICAL METHODOLOGY FOR THE DESIGN AND ANALYSIS OF FINAL STATUS DECOMMISSIONING SURVEYS. Draft Report For Comment.
 NUREG-1506 DRFT FC: MEASUREMENT METHODS FOR RADIOLOGICAL SURVEYS IN SUPPORT OF NEW DECOMMISSIONING CRITERIA. Draft Report For Comment.
 NUREG-1507 DRFT FC: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS. Draft Report For Comment.
 NUREG-1530: REASSESSMENT OF NRC'S DOLLAR PER PERSON-REM CONVERSION FACTOR POLICY.

NUREG/CP-0147: PROCEEDINGS OF THE WORKSHOP ON THE ROLE OF NATURAL ANALOGS IN GEOLOGIC DISPOSAL OF HIGH-LEVEL NUCLEAR WASTE. Held In San Antonio, Texas, July 22-25, 1991.

WASTE MANAGEMENT BRANCH (POST 941217)

NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS. Progress Report Of Field Experiments At A Humid Region Site, Beltsville, Maryland.

NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.

DIVISION OF SYSTEMS TECHNOLOGY (POST 941217)

NUREG-0700 R01 DFC: HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE. Draft Report For Comment.

NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS.

NUREG-1519: SURFACE INTERACTIONS OF CESIUM AND BORIC ACID WITH STAINLESS STEEL.

CONTROL, INSTRUMENTATION & HUMAN FACTORS BRANCH (POST 941217)

NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.

NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function And Task Analysis.

NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Supporting Analyses Of Human-System Interfaces, Procedures And Practices, Training And Organizational Practices And Procedures.

NUREG/CR-6159: USING MICRO SAINT TO PREDICT PERFORMANCE IN A NUCLEAR POWER PLANT CONTROL ROOM. A Test Of Validity And Feasibility.

NUREG/CR-6277 V01: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Identification Of Problems And Alternative Approaches.

NUREG/CR-6277 V02: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Function And Task Analysis.

NUREG/CR-6277 V03: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Human-System Interfaces And Procedures.

NUREG/CR-6277 V04: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Training And Organizational Analysis.

NUREG/CR-6277 V05: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Literature Review.

PROBABILISTIC RISK ANALYSIS BRANCH (POST 941217)

NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Main Report.

NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Appendices A And B.

NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.

REACTOR & PLANT SYSTEMS BRANCH (POST 941217)

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NUREG-1535: INGESTION OF PHOSPHORUS-32 AT MASSACHUSETTS INSTITUTE OF TECHNOLOGY, CAMBRIDGE, MASSACHUSETTS, IDENTIFIED ON AUGUST 19, 1995.

EDO - OFFICE OF NUCLEAR REACTOR REGULATION (POST 800428)

OFFICE OF NUCLEAR REACTOR REGULATION (POST 941001)

NUREG-0040 V18 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, October-December 1994. (White Book)

NUREG-0040 V19 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, January-March 1995. (White Book)

NUREG-0040 V19 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, April-June 1995. (White Book)

NUREG-0040 V19 N03: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, July-September 1995. (White Book)

NUREG-0390 V09 N01: TOPICAL REPORT REVIEW STATUS.

NUREG-0847 S16: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)

NUREG-0847 S17: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)

NUREG-0847 S18: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)

NUREG-0847 S19: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)

NUREG-1122 R01: KNOWLEDGE AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS: PRESSURIZED WATER REACTORS.

NUREG-1123 R01: KNOWLEDGE AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS: BOILING WATER REACTORS.

NUREG-1430 V01 R01: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Specifications.

NUREG-1430 V02 R01: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Bases (Sections 2.0 - 3.3).

NUREG-1430 V03 R01: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Bases (Sections 3.4 - 3.9).

NUREG-1431 V01 R01: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Specifications.

NUREG-1431 V02 R01: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Bases (Sections 2.0 - 3.3).

NUREG-1431 V03 R01: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Bases (Sections 3.4 - 3.9).

NUREG-1432 V01 R01: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS. Specifications.

NUREG-1432 V02 R01: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS. Bases (Sections 2.0 - 3.3).

NUREG-1432 V03 R01: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS. Bases (Sections 3.4 - 3.9).

NUREG-1433 V01 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS BWR/4. Specifications.

NUREG-1433 V02 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS BWR/4. Bases (Sections 2.0 - 3.3).

NUREG-1433 V03 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS BWR/4. Bases (Sections 3.4 - 3.10).

NUREG-1434 V01 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6. Specifications.

NUREG-1434 V02 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6. Bases (Sections 2.0 - 3.3).

NUREG-1434 V03 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6. Bases (Sections 3.4 - 3.10).

NUREG-1435 S04: STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS. TMI Action Plan Requirements, Unresolved Safety Issues, Generic Safety Issues, Other Multiplant Action Issues.

NUREG-1478 R01: NON-POWER REACTOR OPERATOR LICENSING EXAMINER STANDARDS.

NUREG-1517: REPORT OF THE SOUTH TEXAS PROJECT ALLEGATIONS REVIEW TEAM.

NUREG-1522: ASSESSMENT OF INSERVICE CONDITIONS OF SAFETY-RELATED NUCLEAR PLANT STRUCTURES.

NUREG-1526: LESSONS LEARNED FROM EARLY IMPLEMENTATION OF THE MAINTENANCE RULE AT NINE NUCLEAR POWER PLANTS.

DIVISION OF REACTOR PROJECTS I/II (DRPE) (POST 941001)

NUREG-0847 S15: 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)

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NUREG-0498 S01: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority)

OFFICE OF NUCLEAR REACTOR REGULATION, DIRECTOR (POST 870411)

NUREG-1482: GUIDELINES FOR INSERVICE TESTING AT NUCLEAR POWER PLANTS.

NRC Originating Organization Index (International Agreements)

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

There were no NUREG/IA reports during this year.

with the same α and β as in the previous case.

For the case of $\alpha = 0$ and $\beta = 1$, the solution is given by the following theorem.

Theorem 2.1. *If $\alpha = 0$ and $\beta = 1$, then the solution of the problem (1.1)–(1.3) is given by*

$$u(x, y, z) = \frac{1}{2\pi} \int_0^{2\pi} \int_0^\pi \frac{f(\theta, \phi)}{\sin \theta} d\theta d\phi, \quad (2.1)$$

where $f(\theta, \phi)$ is the function defined by

$$f(\theta, \phi) = \frac{1}{2\pi} \int_0^{2\pi} \int_0^\pi \frac{f(\theta, \phi)}{\sin \theta} d\theta d\phi. \quad (2.2)$$

NRC Contract Sponsor Index (Contractor Reports)

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

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA

DIVISION OF SAFETY PROGRAMS (POST 870413)
 NUREG/CR-4674 V21: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Main Report And Appendices A-H.
 NUREG/CR-4674 V22: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Appendix I.
 NUREG/CR-6266: ANALYSIS OF BORON DILUTION IN A FOUR-LOOP PWR.

EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS

OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
 NUREG/CR-6328: ADEQUACY OF THE 123-GROUP CROSS-SECTION LIBRARY FOR CRITICALITY ANALYSES OF WATER-MODERATED URANIUM SYSTEMS.
 DIVISION OF INDUSTRIAL & MEDICAL NUCLEAR SAFETY (POST 870729)
 NUREG/CR-0200 V1 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION.Control Modules.
 NUREG/CR-0200 V2P1R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION.Functional Modules F1-F8.
 NUREG/CR-0200 V2P2R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION.Functional Modules F9-F16.
 NUREG/CR-0200 V3 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Miscellaneous.
 NUREG/CR-5657: AUTOCASK (AUTOMATIC GENERATION OF 3-D CASK MODELS).A Microcomputer Based System For Shipping Cask Design Review Analysis.
 NUREG/CR-6074 V04: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING.Technical Report On The Findings Of Task 4.Investigation Of Sealed Source for Paper Mill Digester.
 NUREG/CR-6074 V05: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING.Technical Report On The Findings Of Task 4.Investigation Of Failed Radioactive Stainless Steel Troxler Gauges.
 NUREG/CR-6276: A COMPILATION OF CURRENT REGULATIONS, STANDARDS, AND GUIDELINES IN REMOTE AFTERLOADING BRACHYTHERAPY.
 NUREG/CR-6322: BUCKLING ANALYSIS OF SPENT FUEL BASKET.
 NUREG/CR-6323: RELATIVE RISK ANALYSIS IN REGULATING THE USE OF RADIATION-EMITTING MEDICAL DEVICES.A Preliminary Application.
 NUREG/CR-6324: QUALITY ASSURANCE FOR GAMMA KNIVES.
 NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.
 DIVISION OF FUEL CYCLE SAFETY & SAFEGUARDS (POST 930207)
 NUREG/CR-6287: MANAGEMENT CONCEPTS AND SAFETY APPLICATIONS FOR NUCLEAR FUEL FACILITIES.
 DIVISION OF WASTE MANAGEMENT (NMSS 940403)
 NUREG/CR-6284: CRITICALITY SAFETY CRITERIA FOR LICENSE REVIEW OF LOW-LEVEL WASTE FACILITIES.

EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)

DIVISION OF SYSTEMS RESEARCH (880717-941217)
 NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.
 OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 941217)
 NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0.Verification And Validation (V&V) Manual.
 NUREG/CR-6224: PARAMETRIC STUDY OF THE POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS.

NUREG/CR-6259: CONSTRAINT EFFECTS ON FRACTURE INITIATION LOADS IN HSST WIDE-PLATE TESTS.
 NUREG/CR-6331: ATMOSPHERIC RELATIVE CONCENTRATIONS IN BUILDING WAKES.
 DIVISION OF ENGINEERING TECHNOLOGY (POST 941217)
 NUREG/CR-4219 V10 N2: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For April-September 1993.
 NUREG/CR-4219 V11 N1: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For October 1993 - March 1994.
 NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING WELDS.Seventh Program Report March 1993 - December 1994.
 NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1993 March 1994.
 NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April-September 1994.
 NUREG/CR-5462: AGING STUDY OF BOILING WATER REACTOR HIGH PRESSURE INJECTION SYSTEMS.
 NUREG/CR-5591 V03: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Progress Report For October 1991 - September 1992.
 NUREG/CR-5591 V04 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For April-September 1993.
 NUREG/CR-5591 V05 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For September 1993 Through March 1994.
 NUREG/CR-5591 V05 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Progress Report For April 1994 Through September 1994.
 NUREG/CR-5591 V06 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For October 1994 Through March 1995.
 NUREG/CR-5857: AGING OF TURBINE DRIVES FOR SAFETY-RELATED PUMPS IN NUCLEAR POWER PLANTS.
 NUREG/CR-5944 V02: A CHARACTERIZATION OF CHECK VALVE DEGRADATION AND FAILURE EXPERIENCE IN THE NUCLEAR POWER INDUSTRY.1991 Failures.
 NUREG/CR-5954: EFFECT ON AGING ON PWR CHEMICAL AND VOLUME CONTROL SYSTEM.
 NUREG/CR-6004: PROBABILISTIC PIPE FRACTURE EVALUATIONS FOR LEAK-RATE-DETECTION APPLICATIONS.
 NUREG/CR-6016: AGING AND SERVICE WEAR OF AIR-OPERATED VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.
 NUREG/CR-6017: FIRE MODELING OF THE HEISS DAMPF REACTOR CONTAINMENT.
 NUREG/CR-6089: DETECTION OF PUMP DEGRADATION.
 NUREG/CR-6100: GATE VALVE AND MOTOR-OPERATOR RESEARCH FINDINGS.
 NUREG/CR-6154 V02: EXPERIMENTAL RESULTS FROM CONTAINMENT PIPING BELLOW SUBJECT TO SEVERE ACCIDENT CONDITIONS.Results From Bellows Tested In Corroded Conditions.
 NUREG/CR-6173: A SUMMARY OF THE FIRE TESTING PROGRAM AT THE GERMAN HDR TEST FACILITY.
 NUREG/CR-6184: SEPARATE EFFECTS TESTING AND ANALYSES TO INVESTIGATE LINER TEARING OF THE 1:16-SCALE REINFORCED CONCRETE CONTAINMENT BUILDING.
 NUREG/CR-6191: SIZE AND DEFORMATION LIMITS TO MAINTAIN CONSTRAINT IN K(IC) AND J(C) TESTING OF BEND SPECIMENS.
 NUREG/CR-6192: AGING AND SERVICE WEAR OF SPRING-LOADED PRESSURE RELIEF VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.
 NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.

- NUREG/CR-6220: AN ASSESSMENT OF FIRE VULNERABILITY FOR AGED ELECTRICAL RELAYS.
- NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES. Experiments And Analysis. March 1990 - December 1994.
- NUREG/CR-6239 V01: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Main Report.
- NUREG/CR-6239 V02: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Appendices.
- NUREG/CR-6240: APPLICATION OF BOUNDING SPECTRA TO SEISMIC DESIGN OF PIPING BASED ON THE PERFORMANCE OF ABOVE GROUND PIPING IN POWER PLANTS SUBJECTED TO STRONG MOTION EARTHQUAKES.
- NUREG/CR-6242: CASKS (COMPUTER ANALYSIS OF STORAGE CASKS): A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR STORAGE CASK DESIGN REVIEW. User's Manual To Version 1b (Including Program Reference).
- NUREG/CR-6251: STAINLESS STEEL SUBMERGED ARC WELD FUSION LINE TOUGHNESS.
- NUREG/CR-6264 V01: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. An Assessment Of The J(M) Parameter.
- NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions.
- NUREG/CR-6273: BIAXIAL LOADING EFFECTS ON FRACTURE TOUGHNESS OF REACTOR PRESSURE VESSEL STEEL.
- NUREG/CR-6275: MECHANICAL PROPERTIES OF THERMALLY AGED CAST STAINLESS STEELS FROM SHIPPINGPORT REACTOR COMPONENTS.
- NUREG/CR-6297: FRACTURE EVALUATIONS OF FUSION LINE CRACKS IN NUCLEAR PIPE BIMETALLIC WELDS.
- NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.
- NUREG/CR-6299: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE.
- NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.
- NUREG/CR-6313 V01: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEM. Summary of Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.
- NUREG/CR-6313 V02: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEMS. Supplemental Appendices Presenting Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.
- NUREG/CR-6327: MODELS FOR EMBRITTLEMENT RECOVERY DUE TO ANNEALING OF REACTOR PRESSURE VESSEL STEELS.
- NUREG/CR-6335: FATIGUE STRAIN-LIFE BEHAVIOR OF CARBON AND LOW-ALLOY STEELS, AUSTENITIC STAINLESS STEELS, AND ALLOY 600 IN LWR ENVIRONMENTS.
- NUREG/CR-6358 V01: ASSESSMENT OF UNITED STATES INDUSTRY STRUCTURAL CODES AND STANDARDS FOR APPLICATION TO ADVANCED NUCLEAR POWER REACTORS. Final Report.
- NUREG/CR-6358 V02: ASSESSMENT OF UNITED STATES INDUSTRY STRUCTURAL CODES AND STANDARDS FOR APPLICATION TO ADVANCED NUCLEAR POWER REACTORS. Appendices.
- NUREG/CR-6368: EXPERIMENTAL INVESTIGATION OF SEDIMENTATION OF LOCA-GENERATED FIBROUS DEBRIS AND SLUDGE IN BWR SUPPRESSION POOLS.
- NUREG/CR-6420: SELF-MONITORING SURVEILLANCE SYSTEM FOR PRESTRESSING TENDONS.
- DIVISION OF REGULATORY APPLICATIONS (POST 941217)
- NUREG/CR-3469 V08: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.
- NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS. Progress Report Of Field Experiments At A Humid Region Site, Beltsville, Maryland.
- NUREG/CR-5229 V07: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1994. Annual Report.
- NUREG/CR-5884 V01: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....
- NUREG/CR-5884 V02: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....
- NUREG/CR-5927 V02: EVALUATION OF A PERFORMANCE ASSESSMENT METHODOLOGY FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES. Validation Needs.
- NUREG/CR-6054: ESTIMATING PRESSURIZED WATER REACTOR DECOMMISSIONING COSTS. A User's Manual For The PWR Cost Estimating Computer Program (CECP) Software.
- NUREG/CR-6112: IMPACT OF REDUCED DOSE LIMITS ON NRC LICENSED ACTIVITIES. Major Issues In The Implementation Of ICRP/NCRP Dose Limit Recommendations. Final Report.
- NUREG/CR-6114 V02: AUXILIARY ANALYSES IN SUPPORT OF PERFORMANCE ASSESSMENT OF A HYPOTHETICAL LOW-LEVEL WASTE FACILITY. Two-Phase Flow And Contaminant Transport In Unsaturated Soils With Application To Low-Level Radioactive Waste Disposal.
- NUREG/CR-6188 V02: MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE. Annual Report For FY 1994.
- NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Program. Test Results For Fiscal Years 1986, 1987, 1988, And 1989.
- NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990, 1991, 1992, And 1993.
- NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.
- NUREG/CR-6305: BLT-EC (BREACH, LEACH, TRANSPORT, AND EQUILIBRIUM CHEMISTRY), A FINITE-ELEMENT MODEL FOR ASSESSING THE RELEASE OF RADIONUCLIDES FROM LOW-LEVEL WASTE DISPOSAL UNITS. Background, Theory, And Model Description.
- NUREG/CR-6307: SUMMARY OF COMMENTS RECEIVED AT WORKSHOP ON USE OF A SITE SPECIFIC ADVISORY BOARD (SSAB) TO FACILITATE PUBLIC PARTICIPATION IN DECOMMISSIONING CASES.
- NUREG/CR-6333: BREATH VERSION 1.1 - COUPLED FLOW AND ENERGY TRANSPORT IN POROUS MEDIA. Simulator Description And User Guide.
- NUREG/CR-6334: NEW SENSOR FOR MEASUREMENT OF LOW AIR FLOW VELOCITY. Phase I Final Report.
- NUREG/CR-6347: MULTI-PHASE REACTIVE TRANSPORT THEORY.
- NUREG/CR-6348: THERMALLY DRIVEN MOISTURE REDISTRIBUTION IN PARTIALLY SATURATED POROUS MEDIA.
- NUREG/CR-6351: REVIEW OF SCENARIO SELECTION APPROACHES FOR PERFORMANCE ASSESSMENT OF HIGH-LEVEL WASTE REPOSITORIES AND RELATED ISSUES.
- NUREG/CR-6354 DRF FC: PERFORMANCE TESTING OF ELECTRONIC PERSONAL DOSIMETERS. Draft Report For Comment.
- NUREG/CR-6356: HYDRAULIC CHARACTERIZATION OF HYDROTHERMALLY ALTERED NOPAL TUFF.
- NUREG/CR-6390: RADIOLOGICAL CHARACTERIZATION OF SPENT CONTROL ROD ASSEMBLIES.
- DIVISION OF SYSTEMS TECHNOLOGY (POST 941217)
- NUREG/CR-5535 V01: RELAP5/MOD3 CODE MANUAL. Code Structure, System Models, And Solution Methods.
- NUREG/CR-5535 V02: RELAP5/MOD3 CODE MANUAL. User's Guide And Input Requirements.
- NUREG/CR-5535 V04: RELAP5/MOD3 CODE MANUAL. Models And Correlations.
- NUREG/CR-5535 V05 R1: RELAP5/MOD3 CODE MANUAL. User's Guideline.
- NUREG/CR-6002: RISK-BASED MAINTENANCE MODELING. Prioritization Of Maintenance Importances And Quantification Of Maintenance Effectiveness.
- NUREG/CR-6046: ALERTNESS, PERFORMANCE, AND OFF-DUTY SLEEP ON 8-HOUR AND 12-HOUR NIGHT SHIFTS IN A SIMULATED CONTINUOUS OPERATIONS CONTROL ROOM SETTING.
- NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.
- NUREG/CR-6116 V06: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Graphical Evaluation Module (GEM) Reference Manual.
- NUREG/CR-6116 V10: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Data Loading Manual.
- NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS. Primer And User's Guides. Version 1.8.3 September 1994.
- NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS. Reference Manuals Version 1.8.3 September 1994.
- NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.

- NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function And Task Analysis.
- NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Supporting Analyses Of Human-System Interfaces, Procedures And Practices, Training And Organizational Practices And Procedures.
- NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
- NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
- NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
- NUREG/CR-6143 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Summary Of Results.
- NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Main Report And Appendices.
- NUREG/CR-6143 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Supporting MELCOR Calculations.
- NUREG/CR-6144 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Summary Of Results.
- NUREG/CR-6144 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Evaluation Of Severe Accident Risk During Mid-Loop Operations. Main Report.
- NUREG/CR-6144 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1. Evaluation Of Severe Accident Risk During Mid-Loop Operations. Appendices.
- NUREG/CR-6150 V01: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. Interface Theory.
- NUREG/CR-6150 V02: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. Damage Progression Model Theory.
- NUREG/CR-6150 V03: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. User's Guide And Input Manual.
- NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. MATPRO-A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.
- NUREG/CR-6150 V05: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. Developmental Assessment.
- NUREG/CR-6159: USING MICRO SAINT TO PREDICT PERFORMANCE IN A NUCLEAR POWER PLANT CONTROL ROOM. A Test Of Validity And Feasibility.
- NUREG/CR-6172: REVIEWING PSA-BASED ANALYSES TO MODIFY TECHNICAL SPECIFICATIONS AT NUCLEAR POWER PLANTS.
- NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Main Report.
- NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Appendices A And B.
- NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment. Appendices C, D, E, F, G, H.
- NUREG/CR-6257: CANDU 3 TRANSIENT ANALYSIS USING ATOMIC ENERGY OF CANADA LTD CODES.
- NUREG/CR-6261: A SUMMARY OF ORNL FISSION PRODUCT RELEASE TESTS WITH RECOMMENDED RELEASE RATES AND DIFFUSION COEFFICIENTS.
- NUREG/CR-6263 V01: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS. Candidate Guidelines, Technical Basis And Research Needs. Executive Summary.
- NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS. Candidate Guidelines, Technical Basis And Research Needs. Main Report.
- NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.
- NUREG/CR-6277 V01: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Identification Of Problems And Alternative Approaches.
- NUREG/CR-6277 V02: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Function And Task Analysis.
- NUREG/CR-6277 V03: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Human-System Interfaces And Procedures.
- NUREG/CR-6277 V04: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Training And Organizational Analysis.
- NUREG/CR-6277 V05: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Literature Review.
- NUREG/CR-6285: SEVERE ACCIDENT NATURAL CIRCULATION STUDIES AT THE INEL.
- NUREG/CR-6291 V01: NUCLEAR PLANT ANALYZER. Installation Manual.
- NUREG/CR-6291 V02: NUCLEAR PLANT ANALYZER. Analyzer Reference Manual.
- NUREG/CR-6291 V03: NUCLEAR PLANT ANALYZER. Computer Visual System Reference Manual.
- NUREG/CR-6291 V04: NUCLEAR PLANT ANALYZER. Programmer's Manual.
- NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Main Report.
- NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Appendices A-D.
- NUREG/CR-6310: AN ANALYSIS OF POTASSIUM IODIDE (KI) PROPHYLAXIS FOR THE GENERAL PUBLIC IN THE EVENT OF A NUCLEAR ACCIDENT.
- NUREG/CR-6311: EVALUATING PREDICTION UNCERTAINTY.
- NUREG/CR-6312: ASSESSMENT OF FIBER OPTIC PRESSURE SENSORS.
- NUREG/CR-6315: CANDU REACTORS, THEIR REGULATION IN CANADA, AND THE IDENTIFICATION OF RELEVANT NRC SAFETY ISSUES.
- NUREG/CR-6316 V01: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.
- NUREG/CR-6316 V02: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Assessment Of Conventional Software Verification And Validation Methods.
- NUREG/CR-6316 V03: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Documentation Of Expert System Verification And Validation Methodologies.
- NUREG/CR-6316 V04: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Evaluation Of Knowledge Base Certification Methods.
- NUREG/CR-6316 V05: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Rationale And Description Of V&V Guideline Packages And Procedures.
- NUREG/CR-6316 V06: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Validation Scenarios.
- NUREG/CR-6316 V07: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. User's Manual.
- NUREG/CR-6316 V08: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Bibliography.
- NUREG/CR-6318: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-7.
- NUREG/CR-6325: AN IMPLICIT STEADY-STATE INITIALIZATION PACKAGE FOR THE RELAP5 COMPUTER CODE.
- NUREG/CR-6343: ON-LINE TESTING OF CALIBRATION OF PROCESS INSTRUMENTATION CHANNELS IN NUCLEAR POWER PLANTS. Phase II Final Report.
- NUREG/CR-6349: COST-BENEFIT CONSIDERATIONS IN REGULATORY ANALYSIS.
- NUREG/CR-6355: A LIMITED ASSESSMENT OF THE ASEP HUMAN RELIABILITY ANALYSIS PROCEDURE USING SIMULATOR EXAMINATION RESULTS.
- NUREG/CR-6398: EVALUATION OF THE COMPUTERIZED PROCEDURES MANUAL II (COPMA-II).
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- DIVISION OF ENGINEERING TECHNOLOGY (PRE 860720)
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NUREG/CR-6386: COMPARISONS OF ANSI STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN, NUREG-0800, AND RELATED DOCUMENTS.

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NUREG/CR-5975 R01: INCENTIVE REGULATION OF INVESTOR-OWNED NUCLEAR POWER PLANTS BY PUBLIC UTILITY REGULATORS.

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ADVANCED SYSTEMS TECHNOLOGY, INC.

NUREG/CR-6307: SUMMARY OF COMMENTS RECEIVED AT WORKSHOP ON USE OF A SITE SPECIFIC ADVISORY BOARD (SSAB) TO FACILITATE PUBLIC PARTICIPATION IN DECOMMISSIONING CASES.

AMERICAN NUCLEAR SOCIETY

NUREG/CP-0142 V01: PROCEEDINGS OF THE 7TH INTERNATIONAL MEETING ON NUCLEAR REACTOR THERMAL-HYDRAULICS (NURETH-7). Sessions 1-5.
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NUREG/CR-6312: ASSESSMENT OF FIBER OPTIC PRESSURE SENSORS.
NUREG/CR-6334: NEW SENSOR FOR MEASUREMENT OF LOW AIR FLOW VELOCITY. Phase I Final Report.
NUREG/CR-6343: ON-LINE TESTING OF CALIBRATION OF PROCESS INSTRUMENTATION CHANNELS IN NUCLEAR POWER PLANTS. Phase II Final Report.

ARGONNE NATIONAL LABORATORY

NUREG/CR-4667 V18: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, October 1993 - March 1994.
NUREG/CR-4667 V19: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, April-September 1994.
NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.
NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990, 1991, 1992, And 1993.
NUREG/CR-6266: ANALYSIS OF BORON DILUTION IN A FOUR-LOOP PWR.
NUREG/CR-6275: MECHANICAL PROPERTIES OF THERMALLY AGED CAST STAINLESS STEELS FROM SHIPPINGPORT REACTOR COMPONENTS.
NUREG/CR-6315: CANDU REACTORS, THEIR REGULATION IN CANADA, AND THE IDENTIFICATION OF RELEVANT NRC SAFETY ISSUES.
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NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
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NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Appendices A And B.

NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment. Appendices C, D, E, F, G, H.

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NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.

BATTELLE HUMAN AFFAIRS RESEARCH CENTERS

NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.

BATTELLE MEMORIAL INSTITUTE, COLUMBUS LABORATORIES

NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Final Report.
NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Draft Report For Comment.
NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING WELDS. Seventh Program Report March 1993 - December 1994.
NUREG/CR-6004: PROBABILISTIC PIPE FRACTURE EVALUATIONS FOR LEAK-RATE-DETECTION APPLICATIONS.
NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES. Experiments And Analysis, March 1990 - December 1994.
NUREG/CR-6251: STAINLESS STEEL SUBMERGED ARC WELD FUSION LINE TOUGHNESS.
NUREG/CR-6264 V01: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. An Assessment Of The J(M) Parameter.
NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions.
NUREG/CR-6297: FRACTURE EVALUATIONS OF FUSION LINE CRACKS IN NUCLEAR PIPE BIMETALLIC WELDS.
NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.
NUREG/CR-6299: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE.
NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

BATTELLE MEMORIAL INSTITUTE, PACIFIC NORTHWEST LABORATORY

NUREG/CR-2850 V13: DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1991.
NUREG/CR-5758 V05: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY. Annual Summary Of Program Performance Reports CY 1994.
NUREG/CR-5884 V01: REVISED ANALYSIS OF DECOMMISSIONING FOR THE REFERENCE PRESSURIZED WATER REACTOR POWER STATION. Effects Of Current Regulatory And Other Considerations On The Financial Insurance Requirements Of The Decommissioning Rule And....
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NUREG/CR-5973 R02: CODES AND STANDARDS AND OTHER GUIDANCE CITED IN REGULATORY DOCUMENTS.

NUREG/CR-5975 R01: INCENTIVE REGULATION OF INVESTOR-OWNED NUCLEAR POWER PLANTS BY PUBLIC UTILITY REGULATORS.

NUREG/CR-6054: ESTIMATING PRESSURIZED WATER REACTOR DE-COMMISSIONING COSTS. A User's Manual For The PWR Cost Estimating Computer Program (CECP) Software.

NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.

NUREG/CR-6331: ATMOSPHERIC RELATIVE CONCENTRATIONS IN BUILDING WAKES.

NUREG/CR-6354 DRF FC: PERFORMANCE TESTING OF ELECTRONIC PERSONAL DOSIMETERS.Draft Report For Comment.

NUREG/CR-6355: A LIMITED ASSESSMENT OF THE ASEP HUMAN RELIABILITY ANALYSIS PROCEDURE USING SIMULATOR EXAMINATION RESULTS.

NUREG/CR-6382: COMPARISONS OF ASTM STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.

NUREG/CR-6385: COMPARISONS OF ANS,ASME,AWS, AND NFPA STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.

NUREG/CR-6386: COMPARISONS OF ANSI STANDARDS CITED IN THE NRC STANDARD REVIEW PLAN,NUREG-0800, AND RELATED DOCUMENTS.

NUREG/CR-6390: RADIOLOGICAL CHARACTERIZATION OF SPENT CONTROL ROD ASSEMBLIES.

BATTELLE SEATTLE RESEARCH CENTER

NUREG/CR-5758 V05: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports CY 1994.

BETA CORP, INTERNATIONAL

NUREG/CR-6351: REVIEW OF SCENARIO SELECTION APPROACHES FOR PERFORMANCE ASSESSMENT OF HIGH-LEVEL WASTE REPOSITORIES AND RELATED ISSUES.

BROOKHAVEN NATIONAL LABORATORY

NUREG-0700 R01 DFC: HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE.Draft Report For Comment.

NUREG-1507 DRFT FC: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS.Draft Report For Comment.

NUREG/CP-0140 V01: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING.Plenary Session, Advanced Instrumentation & Control Hardware & Software, Human Factors Research, IPE & PRA.

NUREG/CP-0140 V02: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING.Severe Accident Research, Thermal Hydraulic Research For Advanced Passive LWRs, High-Burnup Fuel Behavior.

NUREG/CP-0140 V03: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING.Primary Systems Integrity, Structural And Seismic Engineering, Aging Research, Products And Applications.

NUREG/CP-0143: PROCEEDINGS OF THE THIRD INTERNATIONAL WORKSHOP ON THE IMPLEMENTATION OF ALARA AT NUCLEAR POWER PLANTS.Held At Hauppauge, Long Island, New York.

NUREG/CP-0148: TRANSACTIONS OF THE TWENTY-THIRD WATER REACTOR SAFETY INFORMATION MEETING.

NUREG/CR-2907 V13: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1992.

NUREG/CR-2907 V14: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS.

NUREG/CR-3469 V08: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.

NUREG/CR-5954: EFFECT ON AGING ON PWR CHEMICAL AND VOLUME CONTROL SYSTEM.

NUREG/CR-6002: RISK-BASED MAINTENANCE MODELING.Prioritization Of Maintenance Importances And Quantification Of Maintenance Effectiveness.

NUREG/CR-6112: IMPACT OF REDUCED DOSE LIMITS ON NRC LICENSED ACTIVITIES. Major Issues In The Implementation Of ICRP/ NCRP Dose Limit Recommendations.Final Report.

NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.

NUREG/CR-6144 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY,UNIT 1.Summary Of Results.

NUREG/CR-6144 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1.Evaluation Of Severe Accident Risk During Mid-Loop Operations.Main Report.

NUREG/CR-6144 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY, UNIT 1.Evaluation Of Severe Accident Risk During Mid-Loop Operations.Appendices.

NUREG/CR-6172: REVIEWING PSA-BASED ANALYSES TO MODIFY TECHNICAL SPECIFICATIONS AT NUCLEAR POWER PLANTS.

NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS.Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990,1991,1992, And 1993.

NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.

NUREG/CR-6305: BLT-EC (BREACH, LEACH, TRANSPORT, AND EQUILIBRIUM CHEMISTRY), A FINITE-ELEMENT MODEL FOR ASSESSING THE RELEASE OF RADIONUCLIDES FROM LOW-LEVEL WASTE DISPOSAL UNITS.Background, Theory, And Model Description.

NUREG/CR-6349: COST-BENEFIT CONSIDERATIONS IN REGULATORY ANALYSIS.

BROWN UNIV., PROVIDENCE, RI

NUREG/CR-6264 V01: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION.An Assessment Of The J(M) Parameter.

NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION.A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions.

CALIFORNIA STATE UNIVERSITY AT FRESNO

NUREG/CR-6305: BLT-EC (BREACH, LEACH, TRANSPORT, AND EQUILIBRIUM CHEMISTRY), A FINITE-ELEMENT MODEL FOR ASSESSING THE RELEASE OF RADIONUCLIDES FROM LOW-LEVEL WASTE DISPOSAL UNITS.Background, Theory, And Model Description.

CALIFORNIA, UNIV. OF, LOS ANGELES, CA

NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Progress Report Of Field Experiments At A Humid Region Site,Beltsville,Maryland.

CALIFORNIA, UNIV. OF, SAN DIEGO, CA

NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.

NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Function And Task Analysis.

NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Supporting Analyses Of Human-System Interfaces,Procedures And Practices,Training And Organizational Practices And Procedures.

CALIFORNIA, UNIV. OF, SANTA BARBARA, CA

NUREG/CR-6327: MODELS FOR EMBRITTLEMENT RECOVERY DUE TO ANNEALING OF REACTOR PRESSURE VESSEL STEELS.

CENTER FOR NUCLEAR WASTE REGULATORY ANALYSES

NUREG-1464: NRC ITERATIVE PERFORMANCE ASSESSMENT PHASE 2.Development Of Capabilities For Review Of A Performance Assessment For A High-Level Waste Repository.

NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.

NUREG/CR-6347: MULTI-PHASE REACTIVE TRANSPORT THEORY.

NUREG/CR-6348: THERMALLY DRIVEN MOISTURE REDISTRIBUTION IN PARTIALLY SATURATED POROUS MEDIA.

CENTRAL RESEARCH INSTITUTE OF ELECTRIC POWER INDUSTRY

NUREG/CR-6235: ASSESSMENT OF SHORT THROUGH-WALL CIRCUMFERENTIAL CRACKS IN PIPES.Experiments And Analysis,March 1990 - December 1994.

COMPUTER SIMULATION & ANALYSIS, INC.

NUREG/CR-6325: AN IMPLICIT STEADY-STATE INITIALIZATION PACKAGE FOR THE RELAP5 COMPUTER CODE.

CONSTRUCTION ENGINEERING LABORATORY, INC.

NUREG/CR-6420: SELF-MONITORING SURVEILLANCE SYSTEM FOR PRESTRESSING TENDONS.

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NUREG/CR-3243: COMPARISON OF ASME CODE FATIGUE EVALUATION METHODS FOR NUCLEAR CLASS 1 PIPING WITH CLASS 2 OR 3 PIPING.

EARTHLINGS COTTAGE

NUREG/CR-6354 DRF FC: PERFORMANCE TESTING OF ELECTRONIC PERSONAL DOSIMETERS.Draft Report For Comment.

EG&G IDAHO, INC.

NUREG/CR-6276: A COMPILATION OF CURRENT REGULATIONS, STANDARDS, AND GUIDELINES IN REMOTE AFTERLOADING BRACHYTHERAPY.

ENERGY, DEPT. OF, ENVIRONMENTAL MEASUREMENTS LABORATORY

NUREG-1505 DRFT FC: A NONPARAMETRIC STATISTICAL METHODOLOGY FOR THE DESIGN AND ANALYSIS OF FINAL STATUS DECOMMISSIONING SURVEYS. Draft Report For Comment.
NUREG-1506 DRFT FC: MEASUREMENT METHODS FOR RADIOLOGICAL SURVEYS IN SUPPORT OF NEW DECOMMISSIONING CRITERIA. Draft Report For Comment.

EPOCH ENGINEERING, INC.

NUREG/CR-6313 V01: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEM. Summary of Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.
NUREG/CR-6313 V02: ROBUST, ACCURATE, AND NON-CONTACTING VIBRATION MEASUREMENT SYSTEMS. Supplemental Appendices Presenting Comparison Measurements Of The Robust Laser Interferometer And Typical Accelerometer Systems.

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NUREG/CR-6348: THERMALLY DRIVEN MOISTURE REDISTRIBUTION IN PARTIALLY SATURATED POROUS MEDIA.
NUREG/CR-6356: HYDRAULIC CHARACTERIZATION OF HYDROTHERMALLY ALTERED NOPAL TUFF.

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HARRIS CORPORATION INFORMATION SYSTEMS

NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Main Report.
NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Appendices A-D.

HARVARD SCHOOL OF PUBLIC HEALTH, BOSTON, MA

NUREG/CP-0141: PROCEEDINGS OF THE 23RD DOE/NRC NUCLEAR AIR CLEANING CONFERENCE. Held In Buffalo, New York, July 25-28, 1994.

HARVARD UNIV., CAMBRIDGE, MA

NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions.

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NUREG/CR-6277 V03: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Human-System Interfaces And Procedures.
NUREG/CR-6277 V04: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Training And Organizational Analysis.
NUREG/CR-6277 V05: HUMAN FACTORS EVALUATION OF TELE-THERAPY. Literature Review.

IDAHO NATIONAL ENGINEERING LABORATORY

NUREG/CP-0144 V01: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Summary.
NUREG/CP-0144 V02: A WORKSHOP ON DEVELOPING RISK ASSESSMENT METHODS FOR MEDICAL USE OF RADIOACTIVE MATERIAL. Supporting Documents.
NUREG/CR-5229 V07: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1994. Annual Report.
NUREG/CR-5462: AGING STUDY OF BOILING WATER REACTOR HIGH PRESSURE INJECTION SYSTEMS.
NUREG/CR-5535 V01: RELAP5/MOD3 CODE MANUAL. Code Structure, System Models, And Solution Methods.
NUREG/CR-5535 V02: RELAP5/MOD3 CODE MANUAL. User's Guide And Input Requirements.
NUREG/CR-5535 V04: RELAP5/MOD3 CODE MANUAL. Models And Correlations.
NUREG/CR-5535 V05 R1: RELAP5/MOD3 CODE MANUAL. User's Guideline.
NUREG/CR-6100: GATE VALVE AND MOTOR-OPERATOR RESEARCH FINDINGS.
NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.
NUREG/CR-6116 V06: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Graphical Evaluation Module (GEM) Reference Manual.
NUREG/CR-6116 V09: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Verification And Validation (V&V) Manual.
NUREG/CR-6116 V10: SYSTEMS ANALYSIS PROGRAMS FOR HANDS-ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 5.0. Data Loading Manual.
NUREG/CR-6150 V01: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. Interface Theory.
NUREG/CR-6150 V02: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. Damage Progression Model Theory.
NUREG/CR-6150 V03: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. User's Guide And Input Manual.
NUREG/CR-6150 V04: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. MATPRO--A Library Of Materials Properties For Light-Water-Reactor Accident Analysis.
NUREG/CR-6150 V05: SCDAP/RELAP5/MOD 3.1 CODE MANUAL. Developmental Assessment.
NUREG/CR-6188 V02: MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE. Annual Report For FY 1994.
NUREG/CR-6256 V01: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Program. Test Results For Fiscal Years 1986, 1987, 1988, And 1989.
NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990, 1991, 1992, And 1993.
NUREG/CR-6257: CANDU 3 TRANSIENT ANALYSIS USING ATOMIC ENERGY OF CANADA LTD CODES.
NUREG/CR-6260: APPLICATION OF NUREG/CR-5999 INTERIM FATIGUE CURVES TO SELECTED NUCLEAR POWER PLANT COMPONENTS.
NUREG/CR-6285: SEVERE ACCIDENT NATURAL CIRCULATION STUDIES AT THE INEL.
NUREG/CR-6291 V01: NUCLEAR PLANT ANALYZER. Installation Manual.
NUREG/CR-6291 V02: NUCLEAR PLANT ANALYZER. Analyzer Reference Manual.
NUREG/CR-6291 V03: NUCLEAR PLANT ANALYZER. Computer Visual System Reference Manual.
NUREG/CR-6291 V04: NUCLEAR PLANT ANALYZER. Programmer's Manual.

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ILLINOIS, STATE OF

NUREG-1516 DRFT FC: MANAGEMENT OF RADIOACTIVE MATERIAL SAFETY PROGRAMS AT MEDICAL FACILITIES.Draft Report For Comment.

ILLINOIS, UNIV. OF, URBANA, IL

NUREG/CR-6191: SIZE AND DEFORMATION LIMITS TO MAINTAIN CONSTRAINT IN K(I)C AND J(I)C TESTING OF BEND SPECIMENS.
NUREG/CR-6259: CONSTRAINT EFFECTS ON FRACTURE INITIATION LOADS IN HSST WIDE-PLATE TESTS.

INSTITUTE FOR CIRCADIAN PHYSIOLOGY

NUREG/CR-6046: ALERTNESS, PERFORMANCE, AND OFF-DUTY SLEEP ON 8-HOUR AND 12-HOUR NIGHT SHIFTS IN A SIMULATED CONTINUOUS OPERATIONS CONTROL ROOM SETTING.

JOHN WREATHALL & CO., INC.

NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.

KOREA INSTITUTE OF NUCLEAR SAFETY

NUREG/CR-6298: FRACTURE BEHAVIOR OF SHORT CIRCUMFERENTIALLY SURFACE-CRACKED PIPE.
NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

LAWRENCE LIVERMORE NATIONAL LABORATORY

NUREG/CP-0145: WORKSHOP ON DEVELOPING SAFE SOFTWARE.Held At Hotel Del Coronado, San Diego, CA, July 22-23, 1992.
NUREG/CR-5657: AUTOCASK (AUTOMATIC GENERATION OF 3-D CASK MODELS).A Microcomputer Based System For Shipping Cask Design Review Analysis.
NUREG/CR-6242: CASKS (COMPUTER ANALYSIS OF STORAGE CASKS): A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR STORAGE CASK DESIGN REVIEW.User's Manual To Version 1b (Including Program Reference).
NUREG/CR-6322: BUCKLING ANALYSIS OF SPENT FUEL BASKET.
NUREG/CR-6323: RELATIVE RISK ANALYSIS IN REGULATING THE USE OF RADIATION-EMITTING MEDICAL DEVICES.A Preliminary Application.
NUREG/CR-6324: QUALITY ASSURANCE FOR GAMMA KNIVES.

LOS ALAMOS NATIONAL LABORATORY

NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.
NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Main Report.
NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Appendices A And B.
NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment.Appendices C,D,E,F,G,H.
NUREG/CR-6311: EVALUATING PREDICTION UNCERTAINTY.

MARYLAND, UNIV. OF, COLLEGE PARK, MD

NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Progress Report Of Field Experiments At A Humid Region Site,Beltsville,Maryland.

MASSACHUSETTS INSTITUTE OF TECHNOLOGY, CAMBRIDGE, MA

NUREG/CR-6305: BLT-EC (BREACH, LEACH, TRANSPORT, AND EQUILIBRIUM CHEMISTRY), A FINITE-ELEMENT MODEL FOR ASSESSING THE RELEASE OF RADIONUCLIDES FROM LOW-LEVEL WASTE DISPOSAL UNITS.Background, Theory, And Model Description.

MICRO ANALYSIS & DESIGN, INC.

NUREG/CR-6159: USING MICRO SAINT TO PREDICT PERFORMANCE IN A NUCLEAR POWER PLANT CONTROL ROOM.A Test Of Validity And Feasibility.

MITRE CORP.

NUREG/CR-6263 V01: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Executive Summary.
NUREG/CR-6263 V02: HIGH INTEGRITY SOFTWARE FOR NUCLEAR POWER PLANTS.Candidate Guidelines, Technical Basis And Research Needs.Main Report.

MODELING & COMPUTER SERVICES

NUREG/CR-6327: MODELS FOR EMBRITTLEMENT RECOVERY DUE TO ANNEALING OF REACTOR PRESSURE VESSEL STEELS.

NETHERLANDS, GOVT. OF

NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Main Report.
NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Appendices A And B.
NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment.Appendices C,D,E,F,G,H.

NORTH CAROLINA STATE UNIV., RALEIGH, NC

NUREG/CR-6398: EVALUATION OF THE COMPUTERIZED PROCEDURES MANUAL II (COPMA-II).

NOTRE DAME, UNIV. OF, NOTRE DAME, IN

NUREG/CR-6017: FIRE MODELING OF THE HEISS DAMPF REAKTOR CONTAINMENT.

OAK RIDGE ASSOCIATED UNIVERSITIES

NUREG-1507 DRFT FC: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS.Draft Report For Comment.

OAK RIDGE NATIONAL LABORATORY

NUREG-1514: GUIDANCE FOR A LARGE TABLETOP EXERCISE FOR A NUCLEAR POWER PLANT.
NUREG/CR-0200 V1 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION.Control Modules.
NUREG/CR-0200 V2P1R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION.Functional Modules F1-F8.
NUREG/CR-0200 V2P2R4: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION.Functional Modules F9-F16.
NUREG/CR-0200 V3 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Miscellaneous.
NUREG/CR-3243: COMPARISON OF ASME CODE FATIGUE EVALUATION METHODS FOR NUCLEAR CLASS 1 PIPING WITH CLASS 2 OR 3 PIPING.
NUREG/CR-4219 V10 N2: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For April-September 1993.
NUREG/CR-4219 V11 N1: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For October 1993 - March 1994.
NUREG/CR-4674 V21: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Main Report And Appendices A-H.
NUREG/CR-4674 V22: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1994 A STATUS REPORT.Appendix I.
NUREG/CR-5591 V03: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Progress Report For October 1991 - September 1992.
NUREG/CR-5591 V04 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For April-September 1993.
NUREG/CR-5591 V05 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For September 1993 Through March 1994.
NUREG/CR-5591 V05 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Progress Report For April 1994 Through September 1994.
NUREG/CR-5591 V06 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For October 1994 Through March 1995.
NUREG/CR-5857: AGING OF TURBINE DRIVES FOR SAFETY-RELATED PUMPS IN NUCLEAR POWER PLANTS.
NUREG/CR-5944 V02: A CHARACTERIZATION OF CHECK VALVE DEGRADATION AND FAILURE EXPERIENCE IN THE NUCLEAR POWER INDUSTRY.1991 Failures.
NUREG/CR-6016: AGING AND SERVICE WEAR OF AIR-OPERATED VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.
NUREG/CR-6089: DETECTION OF PUMP DEGRADATION.
NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS.Primer And User's Guides. Version 1.8.3 September 1994.
NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS.Reference Manuals.Version 1.8.3 September 1994.

NUREG/CR-6192: AGING AND SERVICE WEAR OF SPRING-LOADED PRESSURE RELIEF VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.

NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.

NUREG/CR-6239 V01: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Main Report.

NUREG/CR-6239 V02: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Appendices.

NUREG/CR-6240: APPLICATION OF BOUNDING SPECTRA TO SEISMIC DESIGN OF PIPING BASED ON THE PERFORMANCE OF ABOVE GROUND PIPING IN POWER PLANTS SUBJECTED TO STRONG MOTION EARTHQUAKES.

NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990, 1991, 1992, And 1993.

NUREG/CR-6259: CONSTRAINT EFFECTS ON FRACTURE INITIATION LOADS IN HSST WIDE-PLATE TESTS.

NUREG/CR-6261: A SUMMARY OF ORNL FISSION PRODUCT RELEASE TESTS WITH RECOMMENDED RELEASE RATES AND DIFFUSION COEFFICIENTS.

NUREG/CR-6273: BIAXIAL LOADING EFFECTS ON FRACTURE TOUGHNESS OF REACTOR PRESSURE VESSEL STEEL.

NUREG/CR-6284: CRITICALITY SAFETY CRITERIA FOR LICENSE REVIEW OF LOW-LEVEL WASTE FACILITIES.

NUREG/CR-6318: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST VI-7.

NUREG/CR-6328: ADEQUACY OF THE 123-GROUP CROSS-SECTION LIBRARY FOR CRITICALITY ANALYSES OF WATER-MODERATED URANIUM SYSTEMS.

PACIFIC SCIENCE & ENGINEERING GROUP, INC.

NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.

NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function And Task Analysis.

NUREG/CR-6125 V03: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Supporting Analyses Of Human-System Interfaces, Procedures And Practices, Training And Organizational Practices And Procedures.

PENNSYLVANIA, UNIV. OF, PHILADELPHIA, PA

NUREG/CR-6349: COST-BENEFIT CONSIDERATIONS IN REGULATORY ANALYSIS.

PLG, INC. (FORMERLY PICKARD, LOWE & GARRICK, INC.)

NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.

PRINCETON UNIV., PRINCETON, NJ

NUREG/CR-6114 V02: AUXILIARY ANALYSES IN SUPPORT OF PERFORMANCE ASSESSMENT OF A HYPOTHETICAL LOW-LEVEL WASTE FACILITY. Two-Phase Flow And Contaminant Transport In Unsaturated Soils With Application To Low-Level Radioactive Waste Disposal.

PURDUE UNIV., WEST LAFAYETTE, IN

NUREG/CR-6256 V02: FIELD LYSIMETER INVESTIGATIONS - TEST RESULTS. Low-Level Waste Data Base Development Program: Test Results For Fiscal Years 1990, 1991, 1992, And 1993.

RUTGERS, NEW JERSEY, STATE UNIV. OF, PISCATAWAY, NJ

NUREG/GR-0014: BALDCYPRESS TREE RING ELEMENTAL CONCENTRATIONS AT REELFOOT LAKE, TENNESSEE, FROM AD 1795 TO AD 1820.

S. COHEN & ASSOCIATES, INC.

NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Final Report.

NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Draft Report For Comment.

NUREG/CR-6310: AN ANALYSIS OF POTASSIUM IODIDE (KI) PROPHYLAXIS FOR THE GENERAL PUBLIC IN THE EVENT OF A NUCLEAR ACCIDENT.

SANDIA NATIONAL LABORATORIES

NUREG/CR-5927 V02: EVALUATION OF A PERFORMANCE ASSESSMENT METHODOLOGY FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES. Validation Needs.

NUREG/CR-6017: FIRE MODELING OF THE HEISS DAMPF REAKTOR CONTAINMENT.

NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.

NUREG/CR-6119 V01: MELCOR COMPUTER CODE MANUALS. Primer And User's Guides. Version 1.8.3 September 1994.

NUREG/CR-6119 V02: MELCOR COMPUTER CODE MANUALS. Reference Manuals. Version 1.8.3 September 1994.

NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

NUREG/CR-6143 V01: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Summary Of Results.

NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Main Report And Appendices.

NUREG/CR-6143 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Supporting MELCOR Calculations.

NUREG/CR-6154 V02: EXPERIMENTAL RESULTS FROM CONTAINMENT PIPING BELLWS SUBJECTED TO SEVERE ACCIDENT CONDITIONS. Results From Bellows Tested In Corroded Conditions.

NUREG/CR-6173: A SUMMARY OF THE FIRE TESTING PROGRAM AT THE GERMAN HDR TEST FACILITY.

NUREG/CR-6184: SEPARATE EFFECTS TESTING AND ANALYSES TO INVESTIGATE LINER TEARING OF THE 1:16-SCALE REINFORCED CONCRETE CONTAINMENT BUILDING.

NUREG/CR-6220: AN ASSESSMENT OF FIRE VULNERABILITY FOR AGED ELECTRICAL RELAYS.

NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Main Report.

NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Appendices A And B.

NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment. Appendices C, D, E, F, G, H.

SCIENCE & ENGINEERING ASSOCIATES, INC.

NUREG/CR-6220: AN ASSESSMENT OF FIRE VULNERABILITY FOR AGED ELECTRICAL RELAYS.

NUREG/CR-6224: PARAMETRIC STUDY OF THE POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS.

NUREG/CR-6368: EXPERIMENTAL INVESTIGATION OF SEDIMENTATION OF LOCA-GENERATED FIBROUS DEBRIS AND SLUDGE IN BWR SUPPRESSION POOLS.

SCIENCE APPLICATIONS INTERNATIONAL CORP. (FORMERLY SCIENCE APPLICATIONS)

NUREG-0713 V15: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES, 1993. Twenty-Sixth Annual Report.

NUREG/CR-4674 V21: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1994 A STATUS REPORT. Main Report And Appendices A-H.

NUREG/CR-4674 V22: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1994 A STATUS REPORT. Appendix I.

NUREG/CR-6002: RISK-BASED MAINTENANCE MODELING. Prioritization Of Maintenance Importances And Quantification Of Maintenance Effectiveness.

NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.

NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Main Report And Appendices.

NUREG/CR-6172: REVIEWING PSA-BASED ANALYSES TO MODIFY TECHNICAL SPECIFICATIONS AT NUCLEAR POWER PLANTS.

NUREG/CR-6265: MULTIDISCIPLINARY FRAMEWORK FOR HUMAN RELIABILITY ANALYSIS WITH AN APPLICATION TO ERRORS OF COMMISSION AND DEPENDENCIES.
 NUREG/CR-6316 V01: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.
 NUREG/CR-6316 V02: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Assessment Of Conventional Software Verification And Validation Methods.
 NUREG/CR-6316 V03: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Documentation Of Expert System Verification And Validation Methodologies.
 NUREG/CR-6316 V04: GUIDELINES FOR THE VERIFICATION AND VALIDATION EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Evaluation Of Knowledge Base Certification Methods.
 NUREG/CR-6316 V05: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Rationale And Description Of V&V Guideline Packages And Procedures.
 NUREG/CR-6316 V06: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Validation Scenarios.
 NUREG/CR-6316 V07: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. User's Manual.
 NUREG/CR-6316 V08: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Bibliography.

SCIENTECH, INC.

NUREG/CR-6310: AN ANALYSIS OF POTASSIUM IODIDE (KI) PROPHYLAXIS FOR THE GENERAL PUBLIC IN THE EVENT OF A NUCLEAR ACCIDENT.

SOHAR, INC.

NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Main Report.
 NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Appendices A-D.

SOUTHWEST POWER CONSULTANTS, INC.

NUREG-1493: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Final Report.
 NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Draft Report For Comment.

SOUTHWEST RESEARCH INSTITUTE

NUREG/CP-0147: PROCEEDINGS OF THE WORKSHOP ON THE ROLE OF NATURAL ANALOGS IN GEOLOGIC DISPOSAL OF HIGH-LEVEL NUCLEAR WASTE. Held In San Antonio, Texas, July 22-25, 1991.
 NUREG/CR-6074 V04: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING. Technical Report On The Findings Of Task 4. Investigation Of Sealed Source for Paper Mill Digester.

NUREG/CR-6074 V05: SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING. Technical Report On The Findings Of Task 4. Investigation Of Failed Radioactive Stainless Steel Troxler Gauges.
 NUREG/CR-6333: BREATH VERSION 1.1 - COUPLED FLOW AND ENERGY TRANSPORT IN POROUS MEDIA. Simulator Description And User Guide.
 NUREG/CR-6237: MULTI-PHASE REACTIVE TRANSPORT THEORY.
 NUREG/CR-6348: THERMALLY DRIVEN MOISTURE REDISTRIBUTION IN PARTIALLY SATURATED POROUS MEDIA.
 NUREG/CR-6351: REVIEW OF SCENARIO SELECTION APPROACHES FOR PERFORMANCE ASSESSMENT OF HIGH-LEVEL WASTE REPOSITORIES AND RELATED ISSUES.
 NUREG/CR-6356: HYDRAULIC CHARACTERIZATION OF HYDROTHERMALLY ALTERED NOPAL TUFF.

STEVENSON & ASSOCIATES

NUREG/CR-6239 V01: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Main Report.
 NUREG/CR-6239 V02: SURVEY OF STRONG MOTION EARTHQUAKE EFFECTS ON THERMAL POWER PLANTS IN CALIFORNIA WITH EMPHASIS ON PIPING SYSTEMS. Appendices.
 NUREG/CR-6240: APPLICATION OF BOUNDING SPECTRA TO SEISMIC DESIGN OF PIPING BASED ON THE PERFORMANCE OF ABOVE GROUND PIPING IN POWER PLANTS SUBJECTED TO STRONG MOTION EARTHQUAKES.
 NUREG/CR-6358 V01: ASSESSMENT OF UNITED STATES INDUSTRY STRUCTURAL CODES AND STANDARDS FOR APPLICATION TO ADVANCED NUCLEAR POWER REACTORS. Final Report.
 NUREG/CR-6358 V02: ASSESSMENT OF UNITED STATES INDUSTRY STRUCTURAL CODES AND STANDARDS FOR APPLICATION TO ADVANCED NUCLEAR POWER REACTORS. Appendices.

SWEDEN, GOVT. OF

NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

THE KEVRIC COMPANY, INC.

NUREG/CR-6287: MANAGEMENT CONCEPTS AND SAFETY APPLICATIONS FOR NUCLEAR FUEL FACILITIES.

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NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Main Report.
 NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Appendices A And B.
 NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment. Appendices C,D,E,F,G,H.

UNIVERSITY OF MEMPHIS, MEMPHIS, TN

NUREG/GR-0014: BALDCYPRESS TREE RING ELEMENTAL CONCENTRATIONS AT REELFOOT LAKE, TENNESSEE, FROM AD 1795 TO AD 1820.

International Organization Index

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

There were no NUREG/IA reports during this year.

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