NUREG-0304
Vol. 23, No. 1

Abstracts for Publications in the NUREG-Series

Semiannual Compilation for January - June 1998

U.S. Nuclear Regulatory Commission

Office of the Chief Information Officer

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L. L. Stevenson, Project Manager

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PREFACE

The U.S. Nuclear Regulatory Commission (NRC) compiles bibliographic data and abstracts for publications in the NUREG-series available to the public. The compilation is published semiannually.

In the first listing, the bibliographic data and abstracts for these publications are sequenced according to their NUREG-series number: publications including reports or brochures prepared by the staff designated (NUREG-XXXX) or (NUREG/BR-XXXX); conference proceedings designated (NUREG/CP-XXXX); reports prepared by an NRC contractor designated (NUREG/CR-XXXX); and publications resulting from international agreements designated (NUREG/IA-XXXX).

After the principal listing, nine other indexes list the reports by--

Secondary Report Number
Personal Author
Subject
NRC Originating Organization Index for Staff Reports
NRC Originating Organization Index for International Agreement Reports
NRC Contractor Index for Contractor Reports
Contractor
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Licensed Facility

Staff-Prepared Publication

(1) NUREG-1552 (report number); (2) Fire Barrier Penetration Seals in Nuclear Power Plants (report title); (3) Bajwa, C. S., West, K. S. (report authors); (4) Office of Nuclear Reactor Regulation (organizational unit of authors); (5) July 1996 (publication date); (6) 55 pp. (number of pages); (7) 9608230207 (NRC Document Control System accession number--for NRC use); (8) 89455:045 (the microfiche address--for NRC use).

Staff-Prepared Brochure

(1) NUREG/BR-0164, Rev. 2 (report number); (2) NRC: Regulator of Nuclear Safety (report title); (3) None (report author); (4) Office of Public Affairs (organizational unit of author); (5) April 1997 (publication date); (6) 24 pp (number of pages); (7) 9705020298 (NRC Document Control System accession number--for NRC use); (8) 92700:001-031 (the microfiche address--for NRC use).

Contractor-Prepared Publication

(1) NUREG/CR-6279 (report number); (2) Application of Fracture Toughness Scaling Models to the Ductile-to-Brittle Transition (report title); (3) Joyce, J. A. (report author); (4) U.S. Naval

Academy (organizational unit of author); (5) January 1996 (publication date); (6) 42 pp (number of pages); (7) 9602220350 (NRC Document Control System accession number--for NRC use); (8) 87234:102 (the microfiche address--for NRC use).

Conference Proceedings

(1) NUREG/CP-0149, V01 (report number); (2) Proceedings of the Twenty-Third Water Reactor Safety Information Meeting: Plenary Session, High Burnup Fuel Behavior, Thermal Hydraulic Research (report title); (3) Monteleone, S. (report author); (4) Brookhaven National Laboratory (organization that compiled the proceedings); (5) March 1996 (publication date); (6) 278 pp. (number of pages); (7) 9604150352 (NRC Document Control System accession number--for NRC use); (8) 87868:001 (the microfiche address--for NRC use).

International Agreement Publication

(1) NUREG/IA-0133 (report number); (2) Development, Implementation and Assessment of Specific Closure Laws for Inverted-Annular Film-Boiling in a Two-Fluid Model (report title); (3) DeCachard, F. (report author); (4) Paul Scherrer Institute (organizational unit of author); (5) October 1996 (publication date); (6) 103 pp (number of pages); (7) 9611190277 (NRC Document Control System accession number--for NRC use); (8) 90823:249 (the microfiche address--for NRC use).

Some NRC reports in the NUREG-series are posted on NRC's World Wide Web site under the Reference Library icon on the home page: <http://www.nrc.gov>.

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Main Citations and Abstracts

The list of publications in this compilation are arranged by number, where NUREG-XXXX is an NRC staff-originated publication, NUREG/CP-XXXX is an NRC-sponsored conference proceedings, NUREG/CR-XXXX is an NRC contractor-prepared publication, NUREG/IA-XXXX is an international agreement publication, and NUREG/BR-XXXX is a staff-originated publication. The bibliographic information (see Preface for details) is followed by a brief abstract of this publication.

NUREG-0040 V21 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1997.(White Book) * Office of Nuclear Reactor Regulation (Post 941001). April 1998. 61pp. 9805180328. A3431:300.

This periodical covers the results of inspections performed by the NRC's Quality Assurance, Vendor Inspection and Maintenance Branch that have been distributed to the inspected organizations during the period from October through December 1997.

NUREG-0090 V20: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.Fiscal Year 1997. * Office for Analysis & Evaluation of Operational Data, Director. April 1998. 23pp. 9805060097. A3320:327.

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 (NRC) determines to be significant from the standpoint of public health or safety. The Federal Reports Elimination and Sunset Act of 1995 requires that AOs be reported to Congress on an annual basis. This report includes those events that NRC has determined to be AOs during fiscal year 1997. This report addresses two AOs at NRC-licensed facilities. One involved an event at a nuclear power plant, and one involved materials overexposure. The report also addresses four Agreement State AOs. Two of these AOs involved overexposures and two involved radiopharmaceutical misadministrations. In addition, Appendix C of the report includes five events of loss of control of licensed materials.

NUREG-0304 V22 N03: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Third Quarter 1997,July-September. * NRC - No Detailed Affiliation Given. January 1998. 41pp. 9802100108. A2079:175.

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

NUREG-0304 V22 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1997. * NRC - No Detailed Affiliation Given. April 1998. 92pp. 9805180333. A3435:187.

See NUREG-0304,V22,N03 abstract.

NUREG-0430 V16: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.July 1, 1995 - June 30, 1996.(Gray Book II) PHAM,T.N. Office of Nuclear Material Safety & Safeguards. February 1998. 19pp. 9802250133. A2282:176.

NRC is committed to the periodic publication of licensed fuel cycle facility inventory difference data, following Agency review of the information and completion of any related 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-0540 V19 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.November 1-30, 1997. * NRC - No Detailed Affiliation Given. January 1998. 284pp. 9802100156. A2078:022.

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 V19 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.December 1-31, 1997. * NRC - No Detailed Affiliation Given. February 1998. 315pp. 9803030348. A2413:001.

See NUREG-0540,V19,N11 abstract.

NUREG-0540 V20 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.January 1-31, 1998. MORRIS,E.B. NRC - No Detailed Affiliation Given. March 1998. 327pp. 9803240332. A2683:001.

See NUREG-0540,V19,N11 abstract.

NUREG-0540 V20 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.February 1-28, 1998. * NRC - No Detailed Affiliation Given. April 1998. 300pp. 9805050440. A3318:039.

See NUREG-0540,V19,N11 abstract.

NUREG-0540 V20 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.March 1-31, 1998. * NRC - No Detailed Affiliation Given. May 1998. 390pp. 9806010321. A3571:044.

See NUREG-0540,V19,N11 abstract.

NUREG-0540 V20 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.April 1-30, 1998. * NRC - No Detailed Affiliation Given. June 1998. 352pp. 9807060350. A4010:001.

See NUREG-0540,V19,N11 abstract.

NUREG-0713 V18: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES,1996.Twenty-Ninth Annual Report. THOMAS,M.L. Division of Regulatory Applications (Post 941217). HAGEMEYER,D. Science Applications International Corp. (formerly Science Applications, Inc.). February 1998. 300pp. 9803180118. A2609:001.

This report summarizes the occupational radiation exposure information that has been reported to the NRC's Radiation Exposure Information Reporting System (REIRS). The bulk of the data presented in the report was obtained from the 1996 annual radiation exposure reports submitted in accordance with the requirements of 10 CFR 20.2206. The 1996 annual reports submitted by about 284 licensees indicated that approximately 138,310 individuals were monitored, 75,139 of whom were monitored by nuclear power facilities. They incurred an average individual dose of 0.1 rem (cSv) and an average measurable dose of about 0.29 rem (cSv). Analyses of transient worker data indicate that 22,348 individuals completed work assignments at two or more licensees during the monitoring year. The dose distributions are adjusted each year to account for the duplicate reporting of transient workers by multiple licensees. In 1996, the average measurable dose calculated from reported data was 0.24 cSv (rem). The corrected dose distribution resulted in an average measurable dose of 0.29 cSv (rem).

NUREG-0750 CI04: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January 1, 1991 through December 31, 1995. * NRC - No Detailed Affiliation Given. November 1997. 452pp. 9803260270. A2725:001.

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

NUREG-0750 V45: NUCLEAR REGULATORY COMMISSION ISSUANCES.Opinions And Decisions Of The Nuclear Regulatory Commission With Selected Orders.January-June 1997. * NRC - No Detailed Affiliation Given. December 1997. 526pp. 9802200041. A2245:001.

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

NUREG-0750 V46 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-September 1997. * NRC - No Detailed Affiliation Given. March 1998. 31pp. 9803270317. A2767:305.

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

NUREG-0750 V46 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-December 1997. * NRC - No Detailed Affiliation Given. April 1998. 49pp. 9805050382. A3320:196.

See NUREG-0750,V46,I01 abstract.

NUREG-0750 V46 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1997. Pages 49-193. * NRC - No Detailed Affiliation Given. January 1998. 151pp. 9802180098. A2195:001.

See NUREG-0750,V45 abstract.

NUREG-0750 V46 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1997. Pages 195-256. * NRC - No Detailed Affiliation Given. February 1998. 69pp. 9802180103. A2195:155.

See NUREG-0750,V45 abstract.

NUREG-0750 V46 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1997. Pages 257-285. * NRC - No Detailed Affiliation Given. February 1998. 35pp. 9802180107. A2195:224.

See NUREG-0750,V45 abstract.

NUREG-0750 V46 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1997. Pages 287-319. * NRC - No Detailed Affiliation Given. March 1998. 40pp. 9803270320. A2755:195.

See NUREG-0750,V45 abstract.

NUREG-0750 V47 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January-March 1998. * NRC - No Detailed Affiliation Given. June 1998. 17pp. 9806190291. A3903:323.

See NUREG-0750,V46,I01 abstract.

NUREG-0750 V47 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1998. Pages 1-12. * NRC - No Detailed Affiliation Given. March 1998. 18pp. 9803270335. A2755:146.

See NUREG-0750,V45 abstract.

NUREG-0750 V47 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1998. Pages 13-56. * NRC - No Detailed Affiliation Given. April 1998. 51pp. 9805060094. A3321:277.

See NUREG-0750,V45 abstract.

NUREG-0750 V47 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1998.Pages 57-75. * NRC - No Detailed Affiliation Given. April 1998. 25pp. 9805180293. A3428:328.

See NUREG-0750,V45 abstract.

NUREG-0750 V47 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1998.Pages 77-260. * NRC - No Detailed Affiliation Given. June 1998. 191pp. 9807060212. A4009:044.

See NUREG-0750,V45 abstract.

NUREG-0837 V17 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. July-September 1997. STRUCKMEYER,R. Region 1 (Post 820201). January 1998. 229pp. 9801260119. A1892:001.

This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facilities throughout the country for the third quarter of 1997.

NUREG-0910 R03: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE. * NRC - No Detailed Affiliation Given. February 1998. 380pp. 9803190155. A2629:090.

Title 44 United States Code, "Public Printing and Documents," regulations issued by the General Service Administration (GSA) in 41 CFR Chapter 101, Subchapter B, "Management and Use of Information and Records," and regulations issued by the National Archives and Records Administration (NARA) in 36 CFR Chapter XII, Subchapter B, "Records Management," require each agency to prepare and issue a comprehensive records disposition schedule that contains the NARA approved records disposition schedules for records unique to the agency and contains the NARA's General Records Schedules for records common to several or all agencies. The approved records disposition schedules specify the appropriate duration of retention and the final disposition for records created or maintained by the NRC. NUREG-0910, Rev. 3, contains "NRC's Comprehensive Records Disposition Schedule," and the original authorized approved citation numbers issued by NARA. Rev. 3 incorporates NARA approved changes and additions to the NRC schedules that have been implemented since the last revision dated March, 1992, reflects recent organizational changes implemented at the NRC, and includes the latest version of NARA's General Records Schedule (dated August 1995).

NUREG-0933 S22: A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R. Division of Engineering Technology (Post 941217). March 1998. 398pp. 9804080010. A2923:001.

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 V16 N02: NRC REGULATORY AGENDA.Semiannual Report.July-December 1997. * Office of Administration, Director (Post 940714). February 1998. 68pp. 9803050116. A2424:115.

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-0940 V16 N2 P1: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED INDIVIDUAL ACTIONS.Semiannual Progress Report,July-December 1997. * Ofc of Enforcement (Post 870413). April 1998. 416pp. 9805210430. A3499:001.

This compilation summarizes significant enforcement actions that have been resolved during the period (July - December 1997) and includes copies of Orders and Notices of Violations 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 V16 N2 P2: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED REACTOR LICENSEES.Semiannual Progress Report,July-December 1997. * Ofc of Enforcement (Post 870413). April 1998. 320pp. 9805180409. A3433:027.

This compilation summarizes significant enforcement actions that have been resolved during the period (July - December 1997) 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 V16 N2 P3: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MATERIAL LICENSEES.Semiannual Progress Report,July-December 1997. * Ofc of Enforcement (Post 870413). April 1998. 407pp. 9806010324. A3570:001.

This compilation summarizes significant enforcement actions that have been resolved during the period (July - December 1997) 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 N04: NUCLEAR REGULATORY LEGISLATION.104th Congress. * Office of the General Counsel (Post 860701). December 1997. 594pp. 9804160178. A2992:160.

This document is a compilation of nuclear regulatory legislation and other relevant material through the 104th 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 N04: NUCLEAR REGULATORY LEGISLATION.104th Congress. * Office of the General Counsel (Post 860701). December 1997. 523pp. 9804160185. A2991:001.

See NUREG-0980,V01,N04 abstract.

NUREG-1022 R01: EVENT REPORTING GUIDELINES 10 CFR 50.72 AND 50.73. ALLISON,D.P.; HARPER,M.R.; JONES,W.R.; et al. Office for Analysis & Evaluation of Operational Data, Director. January 1998. 175pp. 9802100113. A2079:001.

Revision 1 to NUREG-1022 clarifies the immediate notification requirements of Title 10 of the Code of Federal Regulations, Part 50, Section 50.72 (10 CFR 50.72), and the 30-day written licensee event report (LER) requirements of 10 CFR 50.73 for nuclear power plants. This revision was initiated to improve the reporting guidelines related to 10 CFR 50.72 and 50.73 and to consolidate these guidelines into a single reference document. A first draft of this document was noticed for public comment in the Federal Register on October 7, 1991 (56 FR 50598). A second draft was noticed for comment in the Federal Register on February 7, 1994 (59 FR 5614). This document updates and supersedes NUREG-1022 and its Supplements 1 and 2 (published in September 1983, February 1984, and September 1985, respectively). It does not change the reporting requirements of 10 CFR 50.72 and 50.73.

NUREG-1100 V14: BUDGET ESTIMATES.Fiscal Year 1999. * Division of Budget & Analysis (Post 890205). February 1998. 176pp. 9802250137. A2282:001.

This report contains the fiscal year budget justification to Congress. The budget provides estimates for salaries and expenses and for the Office of the Inspector General for fiscal year 1999.

NUREG-1125 V19: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS.1997 Annual. * ACRS - Advisory Committee on Reactor Safeguards. April 1998. 222pp. 9806010317. A3572:070.

This compilation contains 67 ACRS reports submitted to the Commission, or to the Executive Director for Operations, during calendar year 1997. 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, the U. S. Library of Congress, and the Internet at http://www.nrc.gov/ACRSACNW. The reports are categorized by the most appropriate generic subject area and by chronological order within the subject area.

NUREG-1187 V01: PERFORMANCE INDICATORS FOR OPERATING COMMERCIAL NUCLEAR POWER REACTORS.Data Through September 1997. * Office for Analysis & Evaluation of Operational Data, Director. January 1998. 485pp. 9802110145. A2094:001.

This Nuclear Regulatory Commission (NRC) report provides performance indicator data, accounting for the different operational conditions, through September 1997 for 109 reactors. There are eight NRC Performance Indicators for Operating Commercial Nuclear Power Plants. (1) automatic scrams while critical, (2) safety system actuations, (3) significant events, (4) safety system failures, (5) forced outage rate, (6) equipment forced outages per 1000 commercial critical hours, (7) collective radiation exposure, and (8) cause codes. This report is based on data extracted from Licensee Event Reports (LERs) submitted in accordance with 10 CFR 50.73, immediate notifications to the NRC Operations Center in accordance with 10 CFR 50.72, monthly operating reports in accordance with plant technical specifications, and screening of operating experience by NRC staff. Radiation exposure data are obtained from the Institute of Nuclear Power Operations (INPO). Graphical presentations of each plant's data, including trends and deviations analyses are provided, as well as tabulated summaries of the data. The trends and deviations analyses and tabulated summaries have been presented and calculated accounting for the plants operational conditions.

NUREG-1272 V10 N01: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA. 1996 Annual Report. * Office for Analysis & Evaluation of Operational Data, Director. December 1997. 265pp. 9804080062. A2920:001.

This annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) describes activities conducted during 1996. The report is published in three parts. NUREG-1272, Vol. 10, 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 trends of some 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 and reports to the NRC's Operations Center. NUREG-1272, Vol. 10, No. 2, covers nuclear materials and presents a review of the events and concerns during 1996 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations. Both reports also contain a discussion of the Incident Investigation Team program and summarize both the Incident Investigation Team and Augmented Inspection Team reports. Each volume contains a list of the AEOD reports issued from CY 1980 through 1996. NUREG-1272, Vol 10, No. 3, covers technical training and presents the activities of the Technical Training Center in support of the NRC's mission in 1996.

NUREG-1272 V10 N02: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA.1996 Annual Report. * Office for Analysis & Evaluation of Operational Data, Director. December 1997. 136pp. 9805050445. A3319:255.

See NUREG-1272,V10,N01 abstract.

NUREG-1272 V10 N03: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA.1996 Annual Report. * Office for Analysis & Evaluation of Operational Data, Director. December 1997. 42pp. 9805060124. A3321:069.

See NUREG-1272,V10,N01 abstract.

NUREG-1363 V07: ATOMIC SAFETY AND LICENSING BOARD BIENNIAL REPORT.Fiscal Years 1995 - 1996. * Atomic Safety & Licensing Board Panel. June 1998. 51pp. 9806290363. A3958:240.

The Panel handled 33 cases in fiscal Year 1995 and 29 cases in fiscal Year 1996. This report summarizes, highlights, and analyzes how the wide-ranging issues raised in these cases were addressed by the Panel's licensing boards and presiding officers during this period. This report also describes the Panel's other responsibilities, addresses the status of Panel activities, and reports on present and projected future caseloads.

NUREG-1415 V10 N02: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report To Congress,October 1, 1997 - March 31, 1998. * Office of the Inspector General (Post 890417). June 1998. 36pp. 9807060272. A4009:318.

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 to Congress no later than November 30 and May 31, respectively.

NUREG-1507: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS. ABELQUIST,E.W. Oak Ridge Associated Universities. BROWN,W.S. Brookhaven National Laboratory. POWERS,G.E.; et al. Division of Regulatory Applications (Post 941217). June 1998. 194pp. 9806190288. A3903:157.

This document 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. The U.S. Nuclear Regulatory Commission (NRC) has amended its regulations to establish residual radioactivity criteria for decommissioning of licensed nuclear facilities. In support of that rulemaking, the Commission has prepared a Generic Environmental Impact Statement (GEIS), consistent with the National Environmental Policy Act (NEPA). 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 radiological surveys is the minimum detectable concentration (MDC) of field survey instruments in relation to the residual radioactivity criteria. The purpose of this study was two-fold. First, the data were used to determine the validity of the theoretical minimum detectable concentrations (MDCs) used in the 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. The types of instruments commonly used in field radiological surveys that were evaluated included, in part, gas proportional, Geiger-Mueller (GM), zinc sulfide (ZnS), and sodium iodide (NaI) detectors.

NUREG-1542 V03: ACCOUNTABILITY REPORT FISCAL YEAR 1997. CONNELLY,S.R. Office of the Controller (Post 890205). March 1998. 92pp. 9804200258. A3031:011.

The U.S. Nuclear Regulatory Commission (NRC) is one of several Federal agencies participating in a pilot project to streamline financial management reporting. The goal of this pilot is to consolidate performance-related reporting into a single accountability report in accordance with the Government Management Reform Act (GMRA) of 1994. The NRC's third accountability report consolidates the information previously reported in the NRC's annual financial statement required by the Chief Financial Officers Act of 1990, as amended; the chairman's annual report to the President and the Congress, required by the Federal Managers Financial Integrity Act of 1982; and the Chairman's semiannual report to the Congress on management decisions and final actions on Office of Inspector General (OIG) audit recommendations, required by the Inspector General Act of 1978, as amended. This report also includes performance measures, as required by the Chief Financial Officers Act, the Government Performance Results Act of 1993, and the Chairman's statement on the compliance of the agency's financial management systems with the Federal Financial Management Improvement Act of 1996.

NUREG-1560 V01 P1: INDIVIDUAL PLANT EXAMINATION PROGRAM: PERSPECTIVES ON REACTOR SAFETY AND PLANT PERFORMANCE.Summary Report. * Division of Systems Technology (Post 941217). December 1997. 257pp. 9802200064. A2254:001.

This report provides perspectives gained by reviewing 75 Individual Plant Examination (IPE) submittals pertaining to 108 nuclear power plant units. IPEs are probabilistic analyses that estimate the core damage frequency (CDF) and containment performance for accidents initiated by internal events (including internal floods, but excluding internal fire). The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, reviewed the IPE submittals with the objective of gaining perspectives in three major areas: (1) improvements made to individual plants as a result of their IPEs and the collective results of the IPE program, (2) plant-specific design and operational features and modeling assumptions that significantly affect the estimates of CDF and containment performance, and (3) strengths and weaknesses of the models and methods used in the IPEs. These perspectives are gained by assessing the core damage and containment performance results, including overall CDF, accident sequences, dominant contributions to the design and operational characteristics of the various reactor and containment types, and by comparing the IPEs to probabilistic risk assessment characteristics. Methods, data, boundary conditions, and assumptions used in the IPEs are considered in understanding the difference and similarities observed among the various types of plants.

NUREG-1560 V02 P2-5: INDIVIDUAL PLANT EXAMINATION PROGRAM: PERSPECTIVES ON REACTOR SAFETY AND PLANT PERFORMANCE. * Division of Systems Technology (Post 941217). December 1997. 546pp. 9802200072. A2246:163.

See NUREG-1560,V01,P1 abstract.

NUREG-1560 V03 P6: INDIVIDUAL PLANT EXAMINATION PROGRAM: PERSPECTIVES ON REACTOR SAFETY AND PLANT PERFORMANCE.Appendices. * Division of Systems Technology (Post 941217). December 1997. 46pp. 9802200077. A2249:302.

See NUREG-1560,V01,P1 abstract.

NUREG-1570: RISK ASSESSMENT OF SEVERE ACCIDENT-INDUCED STEAM GENERATOR TUBE RUPTURE. * Office of Nuclear Reactor Regulation (Post 941001). March 1998. 218pp. 9803310390. A2839:001.

This report describes the basis, results, and related risk implications of an analysis performed by an ad hoc working group to assess the containment bypass potential attributable to steam generator tube rupture (SGTR) induced by severe accident conditions. The SGTR Severe Accident Working Group, comprised of staff members from the NRC's Offices of Nuclear Reactor Regulation (NRR) and Nuclear Regulatory Research (RES), undertook the analysis beginning in December 1995 to support a proposed steam generator integrity rule. The work drew upon previous risk and thermal-hydraulic analyses of core damage sequences, with a focus on the Surry plant as a representative example. This analysis yielded new results, however, derived by predicting thermal-hydraulic conditions of selected severe accident scenarios using the SCDAP/RELAP5 computer code, flawed tube failure modeling, and tube failure probability estimates. These results, in terms of containment bypass probability, form the basis for the findings presented in this report.

NUREG-1575: MULTI-AGENCY RADIATION SURVEY AND SITE INVESTIGATION MANUAL (MARSSIM).Final Report. * NRC - No Detailed Affiliation Given. * Defense, Dept. of. *; et al. Energy, Dept. of. December 1997. 665pp. 9802200046. EPA-402R-97-016. A2248:001.

The MARSSIM provides information on planning, conducting, evaluating, and documenting building and surface soil final status radiological surveys for demonstrating compliance with dose or risk-based regulations or standards. The MARSSIM is a multi-agency consensus document that was developed collaboratively by four Federal agencies having authority and control over radioactive materials: Department of Defense (DOD), Department of Energy (DOE), Environmental Protection Agency (EPA), and Nuclear Regulatory Commission (NRC). The MARSSIM's objective is to describe a consistent approach for building and surface soil final status surveys to meet established dose or risk-based release criteria, while at the same time encouraging an effective use of resources.

NUREG-1600 R01: GENERAL STATEMENT OF POLICY AND PROCEDURE FOR NRC ENFORCEMENT ACTIONS.Enforcement Policy. * Ofc of Enforcement (Post 870413). May 1998. 32pp. 9806030386. A3636:272.

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 May 13, 1998 (63 FR 26630). 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 matters involving 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, Rev. 1 to provide wide-spread 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-1622: NRC ENFORCEMENT POLICY REVIEW.July 1995 - July 1997. LIEBERMAN,J.; PEDERSEN,R.M. Ofc of Enforcement (Post 870413). April 1998. 66pp. 9805060112. A3321:001.

On June 30, 1995, the Nuclear Regulatory Commission (NRC) issued a complete revision of its General Statement of Policy and Procedure for Enforcement Actions (Enforcement Policy) (60 FR 3481). In approving the 1995 revision to the Enforcement Policy, the Commission directed the staff to perform a review of its implementation of the Policy after approximately 2 years of experience and to consider public comments. This report represents the results of that review.

NUREG-1624 DRFT FC: TECHNICAL BASIS AND IMPLEMENTATION GUIDELINES FOR A TECHNIQUE FOR HUMAN EVENT ANALYSIS (ATHEANA).Draft Report For Comment. * Probabilistic Risk Analysis Branch (Post 941217). May 1998. 404pp. 9806080242. A3687:001.

This report introduces a next-generation HRA method called "A Technique for Human Event Analysis," (ATHEANA). ATHEANA was developed to address limitations identified in current HRA approaches by: (1) addressing errors of commission and dependencies; (2) more realistically representing the human-system interactions that have played important roles in accident response; and (3) integrating advances in psychology with engineering, human factors, and PRA disciplines. This report is the step-by-step guidebook for applying the method. It describes how to: (1) select and organize the ATHEANA team, (2) perform and control the structured search processes for human failure events and unsafe acts, including a discussion of the reasons that such events occur (i.e., the elements of error-forcing context), (3) use the knowledge encoded in the PRA along with the specialized knowledge and experience of the ATHEANA team to focus the searches on those events and reasons that are most likely to affect the risk, and (4) quantify the error-forcing contexts and probability of each unsafe act, given its context.

NUREG-1626: FINAL ENVIRONMENTAL IMPACT STATEMENT FOR THE CONSTRUCTION AND OPERATION OF AN INDEPENDENT SPENT FUEL STORAGE INSTALLATION TO STORE THE THREE MILE ISLAND UNIT 2 SPENT FUEL AT THE IDAHO NATIONAL ENGINEERING AND ENVIRONMENTAL.... * Office of Nuclear Material Safety & Safeguards. March 1998. 219pp. 9803180129. A2611:303.

This Final Environmental Impact Statement (FEIS) was prepared by the U.S. Nuclear Regulatory Commission in accordance with the requirements of 10 CFR Part 51. The FEIS contains an assessment of the potential environmental impacts of the construction and operation of an Independent Spent Fuel Storage Installation (ISFSI) for the Three Mile Island Unit 2 (TMI-2) fuel debris at the Idaho National Engineering and Environmental Laboratory (INEEL). The NRC proposes to issue a license to the U.S. Department of Energy-Idaho Operations Office (DOE-ID) which will authorize DOE-ID to store the TMI-2 fuel debris in an ISFSI. DOE-ID is proposing to design, construct, and operate at the Idaho Chemical Processing Plant (ICPP). The TMI-2 fuel debris would be removed from wet storage at the Test Area North pool, transported to the ISFSI at the ICPP, and placed in storage modules on a concrete basemat.

NUREG-1627 V01: PERFORMANCE PLAN FY 1999. FUNCHES,J.L. NRC - No Detailed Affiliation Given. February 1998. 91pp. 9805200007. A3468:006.

The NRC's performance plan complements the agency's strategic plan by setting annual goals with measurable target levels of performance for FY 1999, as required by the Government Performance and Results Act.

NUREG-1629: THE CHARACTERIZATION OF VICKER'S MICROHARDNESS INDENTATIONS AND PILE-UP PROFILES AS A STRAIN-HARDENING MICROPROBE. SANTOS,C. Division of Engineering Technology (Post 941217). ODETTE,G.R.; LUCAS,G.E.; et al. California, Univ. of, Santa Barbara, CA. April 1998. 153pp. 9805180231. A3428:008.

Microhardness measurements have long been used to examine strength properties and changes in strength properties in metals, for example, as induced by irradiation. Microhardness affords a relatively simple test that can be applied to very small volumes of material. Microhardness is nominally related to the flow stress of the material at a fixed level of plastic strain. Further, the geometry of the pile-up of material around the indentation is related to the strain-hardening behavior of the material; steeper pile-ups correspond to smaller strain hardening rates. In this study the relationship between pile-up profiles and strain hardening is examined using both experimental and analytical methods. Vicker's microhardness tests have been performed on a variety of metal alloys including low alloy, high Cr and austenitic stainless steels. The pile-up topology around the indentations has been quantified using confocal microscopy techniques. In addition, the indentation and pile-up geometry has been simulated using finite element method techniques. These results have been used to develop improved quantification of the relationship between pile-up geometry and the strain hardening constitutive behavior of the test material.

NUREG/CP-0162 V01: PROCEEDINGS OF THE TWENTY-FIFTH WATER REACTOR SAFETY INFORMATION MEETING.Plenary Sessions,Pressure Vessel Research,BWR Strainer Blockage And Other Generic Safety Issues,Environmentally Assisted Degradation Of LWR.... MONTELEONE,S. Brookhaven National Laboratory. March 1998. 370pp. 9805180401. A3427:001.

This three-volume report contains papers presented at the Twenty-Fifth Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 20-22, 1997. 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 France, Japan, Norway, and Russia. 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-0162 V02: PROCEEDINGS OF THE TWENTY-FIFTH WATER REACTOR SAFETY INFORMATION MEETING.Human Reliability Analysis And Human Performance Evaluation, Technical Issues Related To Rulemakings, Risk-Informed, Performance-Based Initiatives... MONTELEONE,S. Brookhaven National Laboratory. March 1998. 235pp. 9805180394. A3425:048.

See NUREG/CP-0162,V01 abstract.

NUREG/CP-0162 V03: PROCEEDINGS OF THE TWENTY-FIFTH WATER REACTOR SAFETY INFORMATION MEETING.Thermal-Hydraulic Research And Codes, Digital Instrumentation And Control, Structural Performance. MONTELEONE,S. Brookhaven National Laboratory. April 1998. 358pp. 9805180351. A3423:001.

See NUREG/CP-0162,V01 abstract.

NUREG/CP-0163: PROCEEDINGS OF THE WORKSHOP ON REVIEW OF DOSE MODELING METHODS FOR DEMONSTRATION OF COMPLIANCE WITH THE RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION. NICHOLSON,T.J. Division of Regulatory Applications (Post 941217). PARROTT,J.D. Division of Waste Management (NMSS 940403). May 1998. 123pp. 9806080229. A3688:041.

The public "Workshop on Review of Dose Modeling Methods for Demonstration of Compliance with the Radiological Criteria for License Termination" was held at the NRC Headquarters Auditorium, Rockville, Maryland, on November 13-14, 1997. The workshop was one in a series to support NRC staff development of guidance for implementing the final rule on "Radiological Criteria for License Termination." The workshop topics included discussion of: dose models used for decommissioning reviews; identification of criteria for evaluating the acceptability of dose models; and selection of parameter values for demonstrating compliance with the final rule. The 2-day public workshop was jointly organized by RES and NMSS staff responsible for reviewing dose modeling methods used in decommissioning reviews. The workshop was noticed in the Federal Register (62 FR 51706). The workshop presenters included: NMSS and RES staff, who discussed both dose modeling needs for licensing reviews, and development of guidance related to dose modeling and parameter selection needs; DOE national laboratory scientists, who provided responses to earlier NRC staff-developed questions and discussed their various Federally-sponsored dose models (i.e., DandD, RESRAD, and MEPAS codes); and an EPA scientist, who presented details on the EPA dose assessment model (i.e., PRESTO code). The workshop was formatted to provide opportunities for the attendees to observe computer demonstrations of the dose codes presented. More than 120 workshop attendees from NRC Headquarters and the Regions, Agreement States; as well as industry representatives and consultants; scientists from EPA, DOD, DNFSB, DOE, and the national laboratories; and interested members of the public participated. A complete transcript of the workshop, including viewgraphs and attendance lists, is available in the NRC Public Document Room. This NUREG/CP documents the formal presentations made during the workshop, and provides a preface outlining the workshop's focus, objectives, background, topics and questions provided to the invited speakers, and those raised during the panel discussion. NUREG/CP-0163 also provides technical bases supporting the development of decommissioning guidance.

NUREG/CR-4554 V01 R2: SCANS (SHIPPING CASK ANALYSIS SYSTEM) A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR SHIPPING CASK DESIGN REVIEW.User's Manual to Version 3a. MOK,G.C.; THOMAS,G.R.; GERHARD,M.A.; et al. Lawrence Livermore National Laboratory. March 1998. 219pp. 9803260397. UCID-20674. A2727:001.

SCANS (Shipping Cask ANalysis System) is a microcomputer based system of computer programs and databases developed at the Lawrence Livermore National Laboratory (LLNL) for evaluating safety analysis reports on spent fuel shipping casks. SCANS is an easy-to-use system that calculates the global response to impact loads, pressure loads and thermal conditions, providing reviewers with an independent check on analyses submitted by licensees. SCANS 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. Analysis options are based on regulatory cases described in the Code of Federal Regulations 10 CFR 71 and Regulatory Guides published by the U.S. Nuclear Regulatory Commission in 1977 and 1978.

NUREG/CR-4667 V24: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT-WATER REACTORS. Semiannual Report,January-June 1997. CHOPRA,O.K.; CHUNG,H.M.; GRUBER,E.E.; et al. Argonne National Laboratory. April 1998. 115pp. 9805180239. ANL-98/6. A3428:161.

This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors from January 1997 to June 1997. Topics that have been investigated include (a) fatigue of carbon, low-alloy, and austenitic stainless steels (SSs) used in reactor piping and pressure vessels, b) irradiation-assisted stress corrosion cracking of Types 304 and 304L SS, and (c) EAC of Alloys 600 and 690. Fatigue tests were conducted on ferritic and austenitic SSs in water that contained various concentrations of dissolved oxygen (DO) to determine whether a slow strain rate applied during various portions of a tensile-loading cycle is equally effective in decreasing fatigue life. Slow-strain-rate-tensile tests were conducted in simulated boiling water reactor (BWR) water at 288 degrees C on SS specimens irradiated to a low fluence in the Halden reactor and the results were compared with similar data from a control-blade sheath and neutron-absorber tubes irradiated in BWRs to the same fluence level. Crack-growth-rate tests were conducted on compact-tension specimens from several heats of Alloys 600 and 690 in low-DO, simulated pressurized water reactor environments.

NUREG/CR-4674 V25: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1996. A Status Report. BELLES,R.J.; CLETCHER,J.W.; COPINGER,D.A.; et al. Oak Ridge National Laboratory. December 1997. 271pp. 9802200043. ORNL/NOAC-232. A2250:001.

This report describes the 14 operational events in 1996 that affected 13 commercial light-water reactors and that are considered to be precursors to potential severe core damage accidents. All these events had conditional probabilities of subsequent severe core damage greater than or equal to 1.0 x 10(-6). These events were identified by first computer-screening the 1996 licensee event reports from commercial light-water reactors to identify those events that could potentially be precursors. Candidate precursors were selected and evaluated in a process similar to that used in previous assessments. Selected events underwent engineering evaluation that identified, analyzed, and documented the precursors. Other events designated by the Nuclear Regulatory Commission (NRC) also underwent a similar evaluation. Finally, documented precursors were submitted for review by licensees and NRC headquarters and regional offices to ensure the plant design and its response to the precursor were correctly characterized. This study is a continuation of earlier work, which evaluated 1969-1995 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 the events.

NUREG/CR-5361: SEISMIC ANALYSIS OF PIPING.Final Program Report. JAQUAY,K. . June 1998. 400pp. 9807060324. A4008:001.

This report provides a summary of the work conducted by the Energy Technology Engineering Center (ETEC) under the U.S. Nuclear Regulatory Commission Seismic Analysis of Piping Program. ETEC was contracted by the NRC to review the technical bases for new rules in the ASME Boiler and Pressure Vessel Code, Section III related to seismic analysis of piping systems in nuclear power plants, and evaluate the cumulative impact of these changes in design criteria on overall safety margins of these piping systems. The ETEC effort is documented in this report.

NUREG/CR-5502: ENGINEERING DRAWINGS FOR 10 CFR PART 71 PACKAGE APPROVALS. SHEAFFER,M.K.; THOMAS,G.R.; DANN,R.K.; et al. Lawrence Livermore National Laboratory. May 1998. 20pp. 9806100427. URCL-ID-130438. A3728:158.

This report provides information for preparing drawings of transportation packages submitted in an application for approval under 10 CFR Part 71. It discusses the purpose of these drawings and describes the recommended format and technical content appropriate for package applications. Examples of frequently used drawing symbols are also provided.

NUREG/CR-5562: DATING AND EARTHQUAKES: REVIEW OF QUATERNARY GEOCHRONOLOGY AND ITS APPLICATION TO PALEOSEISMOLOGY. SOWERS,J.M.; LETTIS,W.R. Affiliation Not Assigned. NOLLER,J.S. Vanderbilt Univ., Nashville, TN. March 1998. 850pp. 9807100109. A4103:001.

Quaternary geochronology, or the dating of Quaternary deposits and landforms, is critical to paleoseismology; it provides the means of assessing rates of deformation and the timing of past displacements. This report provides: (1) reviews of twenty-two Quaternary geochronologic methods, each authored by an active researcher, (2) a discussion of the application of geochronology to paleoseismology, including twelve separately authored case studies, and (3) the results of four original field and laboratory studies. Quaternary geochronology is a growing field in which new methods are being developed and existing methods are being improved, resulting in a large selection of methods and greater accuracy and applicability for most methods. In addition, most dating methods are undergoing continued testing to better understand their limitations and applicability. This has led to more effective application, and occasionally, decreased use of specific methods. Despite the many dating methods available and these new advances, obtaining accurate and precise age estimates of Quaternary deposits and landforms remains a challenge. Best results are obtained when the paleoseismologist and geochronologist closely collaborate, when age estimates are verified by the application of multiple dating methods, when error analysis accounts for all sources of uncertainty, and when studies undergo technical peer review.

NUREG/CR-5591 V04 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For October 1992 Through March 1993. CORWIN,W.R. Oak Ridge National Laboratory. April 1998. 50pp. 9805180222. ORNL/TM-11568. A3428:277.

The primary 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. During this reporting period, irradiated crack-arrest specimens were tested; charpy V-notch specimens of high-cooper weld metal were annealed and tested; a fracture mechanics evaluation of the unirradiated Midland low upper-shelf weld was nearly completed; irradiation of the first large Midland capsule was completed; refined calculations and detailed experimental measurements of the exposure parameters in the High Flux Isotope Reactor were evaluated; in-cavity irradiation of vessel support materials were completed; unirradiated microstructural characterization of a Russian reactor vessel steel was completed; collaborative investigations of in-cascade point-defect generation experiments and investigations of a very wide range of flux levels on low-temperature embrittlement were initiated; baseline testing for an ASTM round robin on reconstituted Charpy V-notch specimens was completed; informal agreement was reached on collaboration on pressure vessel material from the Japan Power Demonstration Reactor; impact and tensile specimens of two U.S. reactor vessels materials were encapsulated and irradiation begun in a Russian reactor; and tensile and impact specimens were tested for three stainless steel welds aged for up to 20,000 h.

NUREG/CR-5591 V08 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For October 1996 Through March 1997. ROSSEEL,T.M. Oak Ridge National Laboratory. February 1998. 66pp. 9803180030. ORNL/TM-11568. A2612:158.

Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents that have the potential for major contamination release. Because the RPV is the only key safety-related component of the plant for which a redundant backup system does not exist, it is imperative to fully understand the degree of irradiation-induced degradation of the RPV's fracture resistance that occurs during service. For this reason, the Heavy-Section Steel Irradiation (HSSI) Program has been established. Its primary goal 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 RPV integrity. Effects of specimen size; material chemistry; product form and microstructure; irradiation fluence, flux, temperature, and spectrum; and postirradiation annealing are being examined on a wide range of fracture properties. The HSSI Program is arranged into seven tasks: (1) program management, (2) irradiation effects in engineering materials, (3) annealing, (4) microstructural analysis of radiation effects, (5) in-service irradiated and aged material evaluations, (6) fracture toughness curve shift method, (7) special technical assistance, and (8) foreign research interactions. The work is performed by the Oak Ridge National Laboratory.

NUREG/CR-6119 V01 R1: MELCOR COMPUTER CODE MANUALS.Primer And Users' Guides, Version 1.8.4,July 1997. GAUNTT,R.O.; COLE,R.K. Sandia National Laboratories. HODGE,S.A.; et al. Oak Ridge National Laboratory. May 1998. 625pp. 9806100430. SAND97-2398. A3726:001.

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 behavior. Current uses of MELCOR include estimation of severe accident source terms and their sensitivities and uncertainties in a variety of applications. This publication of the MELCOR computer code manuals corresponds to MELCOR 1.8.4, released to users in July 1997. Volume 1 contains a primer that describes MELCOR's phenomenological scope, organization (by package), and documentation. The remainder of Volume 1 contains the MELCOR User's Guides, which provide the input instructions and guidelines for each package. Volume 2 contains the MELCOR Reference Manuals, which describe the phenomenological models that have been implemented in each package.

NUREG/CR-6119 V02 R1: MELCOR COMPUTER CODE MANUALS.Reference Manuals,Version 1.8.4,July 1997. GAUNTT,R.O.; COLE,R.K. Sandia National Laboratories. HODGE,S.A.; et al. Oak Ridge National Laboratory. May 1998. 800pp. 9806100437. SAND97-2398. A3723:001.

See NUREG/CR-6119,V01,R01 abstract.

NUREG/CR-6359 V01: RAMONA-4B: A COMPUTER CODE WITH THREE-DIMENSIONAL NEUTRON KINETICS FOR BWR AND SBWR SYSTEM TRANSIENTS.Models And Correlations. ROHATGI,U.S.; CHENG,H.S.; KHAN,H.J.; et al. Brookhaven National Laboratory. March 1998. 453pp. 9803190143. BNL-NUREG-52471. A2628:001.

Ramona-4B is a systems transient code for application to different versions of Boiling Water (BWR) such as the current BWR, the Advanced Boiling Water Reactor (ABWR), and the Simplified Boiling Water Reactor (SBWR). This code uses a three-dimensional neutron kinetics model coupled with a multichannel, nonequilibrium, drift-flux, two-phase flow formulation of the thermal hydraulics of the reactor vessel. The code is designed to analyze a wide spectrum of BWR core and system transients and instability issues. Chapter 1 is an overview of the code's capability and limitations; Chapter 2 discusses the neutron kinetics modeling and the implementation of reactivity edits. Chapter 3 is an overview of the heat conduction calculations. Chapter 4 presents modifications to the thermal hydraulics model of the vessel, recirculation loop, steam separators, boron transport and SBWR specific components. Chapter 5 describes modeling of the plant control and safety systems. Chapter 6 presents the modeling of Balance of Plant (BOP). Chapter 7 describes the mechanistic containment model in the code. The content of this report is complementary to the RAMONA-3B code description and assessment document.

NUREG/CR-6359 V02: RAMONA-4B: A COMPUTER CODE WITH THREE-DIMENSIONAL NEUTRON KINETICS FOR BWR AND SBWR SYSTEM TRANSIENTS.User's Manual. ROHATGI,U.S.; CHENG,H.S.; KHAN,H.J.; et al. Brookhaven National Laboratory. March 1998. 350pp. 9803190151. BNL-NUREG-52471. A2630:106.

This document is the User's Manual for the Boiling Water Reactor(BWR), and Simplified Boiling Water Reactor (SBWR) systems transient code RAMONA-4B. The code uses a three-dimensional neutron-kinetics model coupled with a multichannel nonequilibrium, drift-flux, two-phase flow model of the thermal hydraulics of the reactor pressure vessel. The code is designed to analyze a wide spectrum of BWR and SBWR core and system transients. Chapter 1 gives an overview of the code's capabilities and limitations. Chapter 2 describes the code's structure, lists major subroutines, and discusses the computer requirements. Chapter 3 provides the instructions for installing and running the RAMONA-4B code on sun SPARC and IBM workstations. Chapter 4 contains component descriptions and detailed card-by-card input instructions. Chapter 5 gives samples of the tabulated output for the steady-state and transient calculations and discusses the plotting procedures for the steady-state and transient results. Three appendices contain important user and programmer information: lists of plot variables (Appendix A), listings of input deck for sample problem (Appendix B), and a description of the plotting program PAD (Appendix C).

NUREG/CR-6364: HUMAN PERFORMANCE IN RADIOLOGICAL SURVEY SCANNING. BROWN,W.S. Brookhaven National Laboratory. ABELQUIST,E.W. Oak Ridge Associated Universities. March 1998. 54pp. 9803180087. BNL-NUREG-52474. A2609:303.

The probability of detecting residual contamination in the field using portable radiological survey instruments depends not only on the sensitivity of the instrumentation used in scanning, but also on the surveyor's performance. This report provides a basis for taking human performance into account in determining of the minimum level of activity detectable by scanning. A theoretical framework was developed (based on signal detection theory) which allows influences on surveyors to be anticipated and understood, and supports a quantitative assessment of performance. The performance of surveyors under controlled yet realistic field conditions was examined to gain insight into the task and to develop means of quantifying performance. Then, their performance was assessed under laboratory conditions to quantify more precisely their ability to make the required discriminations. The information was used to characterize surveyors' performance in the scanning task and to provide a basis for predicting levels of radioactivity that are likely to be detectable under various conditions by surveyors using portable survey instruments.

NUREG/CR-6377: EFFECTS ON RADIONUCLIDE CONCENTRATIONS BY CEMENT/GROUND-WATER INTERACTIONS IN

SUPPORT OF PERFORMANCE ASSESSMENT OF LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES. KRUPKA,K.M.; SERNE,R.J. Battelle Memorial Institute, Pacific Northwest National Laboratory. May 1998. 154pp. 9806010310. PNNL-11408. A3572:292.

The U.S. Nuclear Regulatory Commission is developing a technical position docur~nt that provides guidance regarding the performance assessment of low-level radioactive waste disposal facilities. This guidance considers the effects that the chemistry of the vault disposal system may have on radionuclide release. The geochemistry of pore waters buffered by cementitious materials in the disposal system will be different from the local ground water. Therefore, the cement-buffered environment needs to be considered within the source term calculations if credit is taken for solubility limits and/or sorption of dissolved radionuclides within disposal units. A literature review was conducted on methods to model pore-water compositions resulting from reactions with cement, experimental studies of cement/water systems, natural analogue studies of cement and concrete, and radionuclide solubilities experimentally determined in cement pore waters. Based on this review, geochemical modeling was used to calculate maximum concentrations for americium, neptunium, nickel, plutonium, radium, strontium, thorium, and uranium for pore-water compositions buffered by cement and local ground-water. Another literature review was completed on radionuclide sorption behavior onto "fresh" cement/concrete where the pore water pH will be ** 10. Based on this review, a database was developed of preferred minimum distribution coefficient (K(d)) values for these radionuclides in cement/concrete environments.

NUREG/CR-6410: NUCLEAR FUEL CYCLE FACILITY ACCIDENT ANALYSIS HANDBOOK. * Science Applications International Corp. (formerly Science Applications, Inc.). March 1998. 637pp. 9804060094. A2880:001.

The purpose of this Handbook is to provide guidance on how to calculate the characteristics of releases of radioactive materials and/or hazardous chemicals from nonreactor nuclear facilities. In addition, the Handbook provides guidance on how to calculate the consequences of those releases. There are four major chapters: Hazard Evaluation and Scenario Development; Source Term Determination; Transport Within Containment/Confinement; and Atmospheric Dispersion and Consequence Modeling. These chapters are supported by Appendices, including: a summary of chemical and nuclear information that contains descriptions of various fuel cycle facilities; details on how to calculate the characteristics of source terms for releases of hazardous chemicals; a comparison on NRC, EPA, and OSHA programs that address chemical safety; a summary of the performance of HEPA and other filters; and a discussion of uncertainties. Several sample problems are presented: a free-fall spill of powder; an explosion with radioactive releases; a fire with radioactive releases; filter failure; hydrogen fluoride release from a tankcar; a uranium hexafluoride cylinder rupture; a liquid spill in a vitrification plant; and a criticality incident. Finally, this Handbook includes a computer model, LPF#1b, that is intended for use in calculating leakpath factors.

NUREG/CR-6412: AGING AND LOSS-OF-COOLANT ACCIDENT (LOCA) TESTING OF ELECTRICAL CONNECTIONS. NELSON,C.F. Sandia National Laboratories. January 1998. 109pp. 9803180097. SAND97-3170. A2611:196.

This report presents the results of an experimental program to determine the aging and loss-of-coolant accident (LOCA) behavior of electrical connections in order to obtain an initial scoping of their performance. Ten types of connections commonly used in nuclear power plants were tested. These included 3 types of conduit seals, 2 types of cable-to-device connectors, 3 types of cable-to-cable connectors, and 2 types of in-line splices. The connections were aged for 6 months under simultaneous thermal (99 degrees C) and radiation (46 Gy/hr) conditions. A simulated LOCA consisting of sequential high dose-rate irradiation (3 kGy/hr) and high-temperature steam exposures followed the aging. Connection functionality was monitored using insulation resistance measurements during the aging and LOCA exposures. Because only 5 of the 10 connection types passed a post-LOCA, submerged dielectric withstand test, further detailed investigation of electrical connections and the effects of cable jacket integrity on the cable-connection system is warranted.

NUREG/CR-6447: RESULTS OF CRACK-ARREST TESTS ON IRRADIATED A 508 CLASS 3 STEEL. ISKANDER,S.K.; MILELLA,P.P.; PINI,A. Oak Ridge National Laboratory. February 1998. 97pp. 9802270188. ORNL-6894. A2342:122.

Crack-arrest specimens of irradiated A 508 class 3 forging steel were tested and evaluated according to the American Society for Testing and Materials Standard Test Method for Determining Plain-Strain Crack-Arrest Fracture Toughness, K(Ia), of Ferritic Steels, E 1221-88. The irradiation-induced shifts while small, averaging only about 10 K, are approximately the same as the Charpy 41-J temperature shifts. The specimens were irradiated at temperatures ranging from 243 to 280 degrees C to fluences varying from 1.7 to 2.7 x 10(19) neutrons/cm(2)(>1 MeV).

NUREG/CR-6453: H. B. ROBINSON-2 PRESSURE VESSEL BENCHMARK. REMEC,I.; KAM,F.B. Oak Ridge National Laboratory. February 1998. 58pp. 9803050078. ORNL/TM-13204. A2424:056.

The HBR-2 benchmark is specified and analyzed in this report. Analysis of the HBR-2 benchmark can be used as partial fulfillment of the requirements for the qualification of the methodology for calculating neutron fluence in pressure vessels, as required by the U.S. Nuclear Regulatory Commission Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Section 1 of this report provides all the dimensions, material compositions, and neutron source data necessary for the analysis. The measured quantities, to be compared with the calculated values, are the specific activities of the neutron dosimeters, on both sides of the pressure vessel: in the surveillance capsule attached to the thermal shield and in the reactor cavity. Section 2 describes the analysis of the HBR-2 benchmark with the computer code DORT and three ENDF/B-VI based multigroup libraries. The average ratio of the calculated-to-measured specific activities (C/M) for the six dosimeters in the surveillance capsule was 0.90 xx 0.04 for all three libraries. The average C/Ms for the cavity dosimeters (without neptunium dosimeter) were 0.89 xx 0.10, 0.91 xx 0.10, and 0.90 xx 0.09 for the BUGLE-93, SAILOR-95, and BUGLE-96 libraries, respectively.

NUREG/CR-6472: PRELIMINARY PHENOMENA IDENTIFICATION AND RANKING TABLES FOR SIMPLIFIED BOILING WATER REACTOR LOSS-OF-COOLANT ACCIDENT SCENARIOS. KROGER,P.G.; ROHATGI,U.S.; JO,J.H.; et al. Brookhaven National Laboratory. April 1998. 162pp. 9805050390. BNL-NUREG-52501. A3317:239.

A set of Phenomena Identification and Ranking Tables (PIRT) for three potential Loss-of-Coolant Accident (LOCA) scenarios in the General Electric Simplified Boiling Water Reactor is presented. The selected LOCA scenarios are typical for the class of small and large breaks generally considered in Safety Analysis Reports. The method used to develop the PIRTs is described. Following a discussion of the transient scenarios, the PIRTs are presented and discussed in detailed and summarized form. A procedure for future validation of the PIRTs, to enhance their value, is outlined.

NUREG/CR-6479: TECHNICAL BASIS FOR ENVIRONMENTAL QUALIFICATION OF MICROPROCESSOR-BASED SAFETY-RELATED EQUIPMENT IN NUCLEAR POWER PLANTS. KORSAH,K. Oak Ridge National Laboratory. HASSAN,M. Brookhaven National Laboratory. TANAKA,T.J.; et al. Sandia National Laboratories. January 1998. 128pp. 9803180022. ORNL/TM-13264. A2610:197.

This document presents the results of studies sponsored by the Nuclear Regulatory Commission (NRC) to provide the technical basis for environmental qualification of computer-based safety equipment in nuclear power plants. The studies were conducted by Oak Ridge National Laboratory (ORNL), Sandia National Laboratories (SNL), and Brookhaven National Laboratory (BNL). The studies address the following: (1) adequacy of the present test methods for qualification of digital I&C systems; (2) preferred (i.e., Regulatory Guide-endorsed) standards; (3) recommended stressors to be included in the qualification process during type testing; (4) resolution of need for accelerated aging for equipment to be located in a benign environment; and (5) determination of an appropriate approach for addressing the impact of smoke in digital equipment qualification programs. Significant findings from the studies form the technical basis for a recommended approach to the environmental qualification of microprocessor-based safety-related equipment in nuclear power plants.

NUREG/CR-6509: THE EFFECT OF INITIAL TEMPERATURE ON FLAME ACCELERATION AND DEFLAGRATION-TO-DETONATION TRANSITION PHENOMENON. CICCARELLI,G.; BOCCIO,J.L.; GINSBERG,T.; et al. Brookhaven National Laboratory. May 1998. 75pp. 9806080235. BNL-NUREG-52515. A3688:162.

The High-Temperature Combustion Facility at BNL was used to conduct deflagration-to-detonation transition (DDT) experiments. Periodic orifice plates were installed inside the entire length of the detonation tube in order to promote flame acceleration. The orifice plates are 27.3-cm outer diameter, which is equivalent to the inner diameter of the tube, and 20.6-cm-inner diameter. The detonation tube length is 21.3-meters long, and the spacing of the orifice plates is one tube diameter. A standard automobile diesel engine glow plug was used to ignite the test mixture at one end of the tube. Hydrogen-air-steam mixtures were tested at a range of temperatures up to 650K and at an initial pressure of 0.1 MPa. It was also observed that the distance required for the flame to accelerate to the point of detonation initiation, referred to as the run-up distance, was found to be a function of both the hydrogen mole fraction and the mixture initial temperature. Decreasing the hydrogen mole fraction or increasing the initial mixture temperature resulted in longer run-up distances. The density ratio across the flame and the speed of sound in the unburned mixture were found to be two parameters which influence the run-up distance.

NUREG/CR-6511 V02: STEAM GENERATOR TUBE INTEGRITY PROGRAM.Annual Report,August 1995 - September 1996. DIERCKS,D.R.; BAKHTIARI,S.; KASZA,K.E.; et al. Argonne National Laboratory. February 1998. 193pp. 9803180026. ANL-97/3. A2611:001.

This report summarizes work performed by Argonne National Laboratory on the Steam Generator Tube Integrity Program from the inception of the program in August 1995 through September 1996. The program is divided into five tasks: (1) Assessment of Inspection Reliability, (2) Research on ISI (inservice-inspection) Technology, (3) Research on Degradation Modes and Integrity, (4) Tube Removals from Steam Generators, and (5) Program Management. Under Task 1, progress is reported on the preparation of facilities and evaluation of nondestructive evaluation techniques for inspecting a mock-up steam generator for round-robin testing, the development of better ways to correlate failure pressure and leak rate with eddy current (EC) signals, the inspection of sleeved tubes, workshop and training activities, and the evaluation of emerging NDE technology. Results are reported in Task 2 on closed-form solutions and finite-element electromagnetic modeling of EC probe responses for various probe designs and flaw characteristics. In Task 3, facilities are being designed and built for the production of cracked tubes under aggressive and near-prototypical conditions and for the testing of flawed and unflawed tubes under normal operating, accident, and severe-accident conditions. Crack behavior and stability are also being modeled to provide guidance for test facility design, develop an improved understanding of the expected rupture behavior of tubes with circumferential cracks, and predict the behavior of flawed and unflawed tubes under severe accident conditions. Task 4 is concerned with the acquisition of tubes and tube sections from retired steam generators for use in the other research tasks. Progress on the acquisition of tubes from the Salem and McGuire I nuclear plants is reported.

NUREG/CR-6534 V02: FRAPCON-3: A COMPUTER CODE FOR THE CALCULATION OF STEADY-STATE, THERMAL-MECHANICAL BEHAVIOR OF OXIDE FUEL RODS FOR HIGH BURNUP. BEYER,C.E.; LANNING,D.D. Battelle Memorial Institute, Pacific Northwest National Laboratory. DAVIS,K.L.; et al. Idaho National Engineering & Environmental Laboratory. December 1997. 111pp. 9803050101. PNNL-11513. A2423:211.

FRAPCON-3 is a FORTRAN IV computer code that calculates the steady-state response of light water reactor fuel rods during long-term burnup. The code calculates the temperature, pressure, and deformation of a fuel rod as functions of time-dependent fuel rod power and coolant boundary conditions. The phenomena modeled by the code include 1) heat conduction through the fuel and cladding, 2) cladding elastic and plastic deformation, 3) fuel-cladding mechanical interaction, 4) fission gas release, 5) fuel rod internal gas pressure, 6) heat transfer between fuel and cladding, 7) cladding oxidation, and 8) heat transfer from cladding to coolant. The code contains necessary material properties, water properties, and heat- transfer correlations. The codes' integral predictions of mechanical behavior have not been assessed against a data base, e.g., cladding strain or failure data. Therefore, it is recommended that the code not be used for analyses of cladding stress or strain. FRAPCON-3 is programmed for use on both mainframe computers and UNIX-based workstations such as DEC 5000 or SUN Sparcstation 10. It is also programmed for personal computers with FORTRAN compiler software and at least 8 to 10 megabytes of random access memory (RAM).

NUREG/CR-6534 V03: FRAPCON-3: INTEGRAL ASSESSMENT. LANNING,D.D.; BEYER,C.E. Battelle Memorial Institute, Pacific Northwest National Laboratory. BERNA,G.A. Affiliation Not Assigned. December 1997. 210pp. 9803050091. PNNL-11513. A2423:001.

Fuel rod material properties and performance models have been updated for the FRAPCON steady-state fuel rod performance code to account for changes in behavior due to extended fuel burnup. The updated code is named FRAPCON-3 and is intended to replace the earlier codes FRAPCON-2 and GAPCON-THERMAL-2. The property and model updates are described in Volume 1 of this report. Volume 2 of this report constitutes the code description document and includes the input instructions. This document (Volume 3) provides the results of the assessment of the integral code predictions to measured data for various performance parameters. In the case of fuel temperature and fission gas release (FGR) predictions, comparison is made to both benchmark data sets and independent benchmark data sets. The benchmark data sets are described in Section 2.0. Appendix A describes each individual set of benchmark data and gives the code input for each data comparison. The data are drawn from a wide range of burnup levels and operating conditions for both PWR and BWR type rods.

NUREG/CR-6536: VERIFICATION OF THE LWRARC CODE FOR LIGHT-WATER-REACTOR AFTERHEAT RATE CALCULATIONS. MURPHY,B.D. Oak Ridge National Laboratory. February 1998. 15pp. 9802270183. ORNL/TM-13396. A2342:107.

This report describes verification studies carried out on the LWRARC (Light-Water-Reactor Afterheat Rate Calculations) computer code. The LWRARC code is proposed for automating the implementation of procedures specified in Draft Revision 1 of the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.54, "Spent-Fuel Heat Generation in an Independent Spent-Fuel Storage Installation," which gives guidelines on the calculation of decay heat for spent nuclear fuel. Draft Regulatory Guide 3.54 allows one to estimate decay-heat values by means of a table lookup procedure with interpolation performed between table-entry values. The tabulated values of the relevant parameters span ranges that are appropriate for spent fuel from a boiling-water reactor (BWR) or a pressurized-water reactor (PWR), as the case may be, and decay-heat rates are obtained for spent fuel whose properties are within those parameter limits. In some instances, where these limits are either exceeded or where they approach critical regions, adjustments are invoked following table lookup. The LWRARC computer code is intended to replicate this manual process.

NUREG/CR-6537: INFLUENCE OF LONG-TERM THERMAL AGING ON THE MICROSTRUCTURAL EVOLUTION OF NUCLEAR REACTOR PRESSURE VESSEL MATERIALS.An Atom Probe Study. PAREIGE,P. France. RUSSELL,K.F.; STOLLER,R.E.; et al. Oak Ridge National Laboratory. March 1998. 28pp. 9803180083. ORNL/TM-13406. A2610:324.

Atom probe field ion microscopy (APFIM) investigations of the microstructure of unaged (as-fabricated) and long-term thermally aged (xx100,000 h at 280 degrees C) surveillance materials from commercial reactor pressure vessel steels were performed. This combination of materials and conditions permitted the investigation of potential thermal-aging effects. This microstructural study focused on the quantification of the compositions of the matrix and carbides. The APFIM results indicate that there was no significant microstructural evolution after a long-term thermal exposure in weld, plate, or forging materials. The matrix depletion of copper that was observed in weld materials was consistent with the copper concentration in the matrix after the stress-relief heat treatment. The compositions of cementite carbides aged for 100,000 h were compared with the Thermocalc(TM) prediction. The APFIM comparisons of materials under these conditions are consistent with the measured change in mechanical properties such as the Charpy transition temperature.

NUREG/CR-6540: STATE-OF-THE-ART REPORT ON PIPING FRACTURE MECHANICS. WILKOWSKI,G.M.; OLSON,R.J.; SCOTT,P.M. Battelle Memorial Institute, Columbus Laboratories. January 1998. 385pp. 9802100139. BMI-2196. A2077:001.

This report is an in-depth summary of the state-of-the-art in nuclear piping fracture mechanics. It represents the culmination of 20 years of work done primarily in the U.S., but also attempts to include important aspects from other international efforts. Although the focus of this work was for the nuclear industry, the technology is also applicable in many cases to fossil plants, petrochemical/refinery plants, and the oil and gas industry. In compiling this detailed summary report, all of the equations and details of the analysis procedure or experimental results are not necessarily included. Rather, the report describes the important aspects and limitations, tells the reader where he can go for further information, and more importantly, describes the accuracy of the models. Nevertheless, the report still contains over 150 equations and over 400 references. The main sections of this report describe: (1) the evolution of piping fracture mechanics history relative to the developments of the nuclear industry, (2) technical developments in stress analyses, material property aspects, and fracture mechanics analyses, (3) unresolved issues and technically evolving areas, and (4) a summary of conclusions of major developments to date.

NUREG/CR-6544: METHODOLOGY FOR ANALYZING PRECURSORS TO EARTHQUAKE-INITIATED AND FIRE-INITIATED ACCIDENT SEQUENCES. BUDNITZ,R.J.; LAMBERT,H.E. Future Resources Associates, Inc. APOSTOLAKI,G.A.; et al. Massachusetts Institute of Technology, Cambridge, MA. April 1998. 149pp. 9805050400. A3320:027.

This report covers work to develop a methodology for analyzing precursors to both earthquake-initiated and internal fire-initiated accidents at commercial nuclear power plants. Currently, the U.S. Nuclear Regulatory Commission sponsors a large ongoing project, the Accident Sequence Precursor project, to analyze the safety significance of other types of accident precursors, such as those arising from internally-initiated transients and pipe breaks, but earthquakes and fires are not within the current scope. The results of this project are that (i) an overall step-by-step methodology has been developed for precursors to both fire-initiated and seismic-initiated potential accidents; (ii) some stylized case-study examples are provided to demonstrate how the fully-developed methodology works in practice, and (iii) a generic seismic-fragility data base for equipment is provided for use in seismic-precursor analyses.

NUREG/CR-6545 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Early Health Effects Uncertainty Assessment.Main Report. HASKIN,F.E. New Mexico, Univ. of, Albuquerque, NM. HARPER,F.T. Sandia National Laboratories. GOOSSENS,L.H.J.; et al. Delft University of Technology. December 1997. 64pp. 9804240196. EUR 16775. A3158:001.

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 cosponsoring 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. 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 results of the study to develop distribution for variables related to the MACCS and COSYMA early health effects models.

NUREG/CR-6545 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Early Health Effects Uncertainty Assessment.Appendices. HASKIN,F.E. New Mexico, Univ. of, Albuquerque, NM. HARPER,F.T. Sandia National Laboratories. GOOSSENS,L.H.J.; et al. Delft University of Technology. December 1997. 350pp. 9804240244. EUR 16775. A3158:068.

See NUREG/CR-6545,V01 abstract.

NUREG/CR-6546: A DAMAGE MECHANICS BASED APPROACH TO STRUCTURAL DETERIORATION AND RELIABILITY. BHATTACHARYA,B.; ELLINGWOOD,B. Johns Hopkins Univ., Baltimore, MD. * Oak Ridge National Laboratory. February 1998. 200pp. 9803180018. ORNLSUB96-SP638. A2610:001.

Structural deterioration often occurs without perceptible manifestation. Continuum damage mechanics defines structural damage in terms of the material microstructure, and relates the damage variable to the macroscopic strength or stiffness of the structure. This enables one to predict the state of damage prior to the initiation of a macroscopic flaw, and allows one to estimate residual strength/service life of an existing structure. The accumulation of damage is a dissipative process that is governed by the laws of thermodynamics. Partial differential equations for damage growth in terms of the Helmholtz free energy are derived from fundamental thermodynamical conditions. Closed-form solutions to the equations are obtained under uniaxial loading for ductile deformation damage as a function of plastic strain, for creep damage as a function of time, and for fatigue damage as function of number of cycles. The proposed damage growth model is extended into the stochastic domain by considering fluctuations in the free energy, and closed-form solutions of the resulting stochastic differential equation are obtained in each of the three cases mentioned above. A reliability analysis of a ring-stiffened cylindrical steel shell subjected to corrosion, accidental pressure, and temperature is performed.

NUREG/CR-6554: FINITE ELEMENT ANALYSES FOR SEISMIC SHEAR WALL INTERNATIONAL STANDARD PROBLEM. PARK,Y.J.; HOFMAYER,C.H. Brookhaven National Laboratory. April 1998. 276pp. 9805050413. BNL-NUREG-52530. A3318:343.

Two identical reinforced concrete (RC) shear walls, which consist of web, flanges and massive top and bottom slabs, were tested up to ultimate failure under earthquake motions at the Nuclear Power Engineering Corporation's (NUPEC) Tadotsu Engineering Laboratory, Japan. NUPEC provided the dynamic test results to the OECD (Organization for Economic Cooperation and Development), Nuclear Energy Agency (NEA) for use as an International Standard Problem (ISP). The shear walls were intended to be part of a typical reactor building. One of the major objectives of the Seismic Shear Wall ISP (SSWISP) was to evaluate various seismic analysis methods for concrete structures used for design and seismic margin assessment. It also offered a unique opportunity to assess the state-of-the-art in nonlinear dynamic analysis of reinforced concrete shear wall structures under severe earthquake loadings. As a participant of the SSWISP workshops, Brookhaven National Laboratory (BNL) performed finite element analyses under the sponsorship of the U.S. Nuclear Regulatory Commission (USNRC). Three types of analysis were performed, i.e., monotonic static (push-over), cyclic static and dynamic analyses. Additional monotonic static analyses were performed by two consultants, F. Vecchio of the University of Toronto (UT) and F. Filippou of the University of California at Berkeley (UCB). The analysis results by BNL and the consultants were presented during the second workshop in Yokohama, Japan in 1996. A total of 55 analyses were presented during the workshop by 30 participants from 11 different countries. The major findings on the presented analysis methods, as well as engineering insights regarding the applicability and reliability of the FEM codes are described in detail in this report.

NUREG/CR-6555 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS.Late Health Effects Uncertainty Assessment.Main Report. LITTLE,M.P.; MUIRHEAD,C.R. United Kingdom. GOOSSENS,L.H.J.; et al. Delft University of Technology. December 1997. 64pp. 9802230110. EUR 16774. A2257:293.

The development of two new probabilistic accident consequence codes, MACCS and COSYMA, was developed in 1990. These codes estimate the consequence 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 cosponsoring 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. 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 results of the study to develop distribution for variables related to the MACCS and COSYMA late health effects models.

NUREG/CR-6555 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS.Late Health Effects Uncertainty Assessment.Appendices. LITTLE,M.P.; MUIRHEAD,C.R. United Kingdom. GOOSSENS,L.H.J.; et al. Delft University of Technology. December 1997. 223pp. 9802230116. EUR 16774. A2257:070.

See NUREG/CR-6555,V01 abstract.

NUREG/CR-6564: ANALYSES OF SOURCE SPECTRA, ATTENUATION, AND SITE EFFECTS FROM CENTRAL AND EASTERN UNITED STATES EARTHQUAKES. LINDLEY,G. California, Univ. of, Santa Barbara, CA. February 1998. 100pp. 9802250142. A2281:189.

Results from 27 previous studies were used to analyze stress drop vs. magnitude in eastern North America. Stress drop was not constant, but increased approximately with the square root of the seismic moment from 3 bars at 10(20) dyne-cm to 690 bars at 10(25) dyne-cm. Q(lg) as a function of frequency was analyzed in five regions of the contiguous United States. Simultaneous inversions using Fourier amplitude spectra were computed to determine attenuation, site responses, and source spectra. Unlike some previous studies, Q(lg) in the central and northeastern U.S. was found to be nearly identical from 2 to 10 Hz. Q(lg) in the southeastern U.S. is about 20% lower. Anelastic attenuation of four regional phases, and source parameters of 27 earthquakes, including the 1995 West Texas earthquake (M(b) 5.6) were also estimated. L(g) attenuation is in good agreement with previous estimates for the central and eastern U.S. Assuming a single corner frequency source model, stress drops range from about 1 to 100 bars. The West Texas earthquake has lower values of a few bars to a few tens of bars.

NUREG/CR-6571 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Uncertainty Assessment For Internal Dosimetry.Main Report. GOOSSENS,L.H.J.; KRAAN,B.C.P.; et al. Delft University of Technology. HARRISON,J.D. National Radiological Protection Board. April 1998. 70pp. 9805180257. EUR 16773. A3425:279.

The development of two new probabilistic accident consequence codes, MACCS and COSYMA, was completed in 1990. These codes estimate the consequence 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 cosponsoring 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. 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 results of the study to develop distribution for variables related to the MACCS and COSYMA internal dosimetry models.

NUREG/CR-6571 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Uncertainty Assessment For Internal Dosimetry.Appendices. GOOSSENS,L.H.J.; KRAAN,B.C.P.; et al. Delft University of Technology. HARRISON,J.D. National Radiological Protection Board. April 1998. 317pp. 9805180287. EUR 16773. A3426:001.

See NUREG/CR-6571,V01 abstract.

NUREG/CR-6573: "INVESTIGATING SEISMOTECTONICS IN THE EASTERN UNITED STATES USING A GEOGRAPHIC INFORMATION SYSTEM." EBEL,J.E.; LAZAREWICZ,A.R.; KAFKA,A.L. Boston College, Weston, MA. February 1998. 109pp. 9803030332. A2379:209.

A Geographic Information System (GIS) database has been assembled to use in regional analyses looking for seismotectonically active features in the central and eastern U.S. (CEUS). Included in the database for the region are topography, earthquakes, stress measurements, gravity residual field, magnetic residual field, major rivers and regional geology, especially faults. Observables from this database were extracted for the seismically active areas of the northeastern, southeastern and central U.S. for use in multivariate statistical analyses. These analyses indicate that the earthquakes of the CEUS do tend to associate with faults and other deformation structures, but that the geologic characteristics are not very similar between earthquakes in different regions. The discriminant function analysis shows some ability to differentiate between seismic and nonseismic areas.

NUREG/CR-6575: FAILURE BEHAVIOR OF INTERNALLY PRESSURIZED FLAWED AND UNFLAWED STEAM GENERATOR TUBING AT HIGH TEMPERATURE -EXPERIMENTS AND COMPARISON WITH MODEL PREDICTIONS. MAJUMDAR,S.; SHACK,W.J.; DIERCKS,D.R.; et al. Argonne National Laboratory. March 1998. 102pp. 9803260391. ANL-97/17. A2738:221.

This report summarizes experimental work performed at Argonne National Laboratory on the failure of internally pressurized steam generator tubing at high temperatures (** 700 degrees C). A model was developed for predicting failure of flawed and unflawed steam generator tubes under internal pressure and temperature histories postulated to occur during severe accidents. The model was validated by failure tests on specimens with part-through-wall axial and circumferential flaws of various lengths and depths, conducted under various constant and ramped internal pressure and temperature conditions. The failure temperatures predicted by the model for two temperature and pressure histories, calculated for severe accidents initiated by a station blackout, agree very well with tests performed on both flawed and unflawed specimens.

NUREG/CR-6577: U.S. NUCLEAR POWER PLANT OPERATING COST AND EXPERIENCE SUMMARIES. KOHN,W.E.; REID,R.L.; WHITE,V.S. Oak Ridge National Laboratory. February 1998. 433pp. 9802230118. ORNL/TM-13494. A2256:001.

This report has been prepared to provide historical operating cost and experience information on U.S. commercial power plants. Costs incurred after initial construction are characterized as annual production costs, representing fuel and plant operating and maintenance expenses, and capital expenditures related to facility additions/modifications which are included in the plant capital asset base. As discussed in the report, annual data for these two cost categories were obtained from publicly available reports and must be accepted as having different degrees of accuracy and completeness. Treatment of inconclusive and incomplete data is discussed. As an aid to understanding the fluctuations in the cost histories, operations summaries for each nuclear unit are provided. The intent of these summaries is to identify important operating events; refueling, major maintenance, and other significant outages; operating milestones; and significant licensing or enforcement actions. Information used in the summaries is condensed from annual operating reports submitted by the licensees, plant histories contained in Nuclear Power Experience, trade press articles, and the Nuclear Regulatory Commission (NRC) web site (www.nrc.gov).

NUREG/CR-6579: DIGITAL I&C SYSTEMS IN NUCLEAR POWER PLANTS.Risk-Screening Of Environmental Stressors And A Comparison Of Hardware Unavailability With An Existing Analog System. HASSAN,M. Brookhaven National Laboratory. VESELY,W.E. Science Applications International Corp. (formerly Science Applications, Inc.). January 1998. 128pp. 9802130002. BNL-NUREG-52536. A2144:068.

In this report, we present a screening study to identify environmental stressors for digital instrumentation and control (I&C) systems in a nuclear power plant (NPP) which can be potentially risk-significant, and compare the hardware unavailability of such a system with that of its existing analog counterpart. The stressors evaluated are temperature, humidity, vibration, radiation, electro- magnetic interference (EMI), and smoke. The results of risk-screening for an example plant, subject to some bounding assumptions and based on relative changes in plant risk (core damage frequency impacts of the stressors), indicate that humidity, EMI from lightning, and smoke can be potentially risk-significant. Risk from other sources of EMI could not be evaluated for a lack of data. Risk from temperature appears to be insignificant as that from the assumed levels of vibrations. A comparison of the hardware unavailability of the existing analog Safety Injection Actuation System (SIAS) in the example plant with that of an assumed digital upgrade of the system indicates that system unavailability may be more sensitive to the level of redundancy in elements of the digital system than to the environmental and operational variations involved. The findings of this study can be used to focus activities relating to the regulatory basis for digital I&C upgrades in NPPs, including identification of dominant stressors, data-gathering, equipment qualification, and requirements to limit the effects of environmental stressors.

NUREG/CR-6583: EFFECTS OF LWR COOLANT ENVIRONMENTS ON FATIGUE DESIGN CURVES OF CARBON AND LOW-ALLOY STEELS. CHOPRA,O.K.; SHACK,W.J. Argonne National Laboratory. March 1998. 128pp. 9803260384. ANL-97/18. A2727:220.

The ASME Boiler and Pressure Vessel Code provides rules for the construction of nuclear power plant components. Figures I-9.1 through I-9.6 of Appendix I to Section III of the Code specify fatigue design curves for structural materials. While effects of reactor coolant environments are not explicitly addressed by the design curves, test data indicate that the Code fatigue curves may not always be adequate in coolant environments. This report summarizes work performed by Argonne National Laboratory on fatigue of carbon and low-alloy steels in light water reactor (LWR) environments. The existing fatigue S-N data have been evaluated to establish the effects of various material and loading variables such as steel type, dissolved oxygen level, strain range, strain rate, temperature, orientation, and sulfur content on the fatigue life of these steels. Statistical models have been developed for estimating the fatigue S-N curves as a function of material, loading, and environmental variables. The results have been used to estimate the probability of fatigue cracking of reactor components. The different methods for incorporating the effects of LWR coolant environments on the ASME Code fatigue design curves are presented.

NUREG/CR-6589: THE EFFECTS OF SURFACE CONDITION ON AN ULTRASONIC INSPECTION: ENGINEERING STUDIES USING VALIDATED COMPUTER MODEL. GREENWOOD,M.S. Battelle Memorial Institute, Pacific Northwest National Laboratory. April 1998. 154pp. 9805200010. PNNL-11751. A3467:217.

This report documents work performed at Pacific Northwest National Laboratory (PNNL) on the effects of surface roughness on the reliability of an ultrasonic inservice inspection. The primary objective of this research is to develop ASME Code recommendations in order to limit the adverse effects of a rough surface and thereby increase the reliability of ultrasonic inservice inspections. In order to achieve this objective, engineering studies were conducted that included experimental validation of computer codes, developed at the Center for Nondestructive Evaluation (CNDE) at Iowa State University as a result of a cooperative effort between the Electric Power Research Institute (EPRI) and the Nuclear Regulatory Commission. The basic problem associated with a rough surface in an inservice inspection is that as the transducer rotates slightly to accommodate the rough surface, the beam direction in the metal changes and the time-of-flight of the echo changes as well. One problem is the excessive weld crown, where weld material protrudes above the adjoining surfaces. In this research this condition is modeled by considering a step discontinuity on the top surface. CNDE developed several models of increasing complexity in order to model an inservice inspection. This report describes the validation of four computer codes. These codes were used to mimic an inservice inspection in order to understand effects associated with rotation of the transducer as it traverses a step discontinuity. Studies resulted in ASME Section XI Code recommendations.

NUREG/CR-6598: AN INVESTIGATION OF TENDON SHEATHING FILLER MIGRATION INTO CONCRETE. NAUS,D.J.; OLAND,C.B. Oak Ridge National Laboratory. March 1998. 81pp. 9807060348. ORNL/TM-13554. A4009:236.

During some of the inspections at nuclear power plants with prestressed concrete containments, it was observed that the containments had experienced leakage of the tendon sheathing filler (i.e., streaks). The objective of this activity was to provide an indication of the extent of tendon sheathing filler leakage into the concrete and its affects on concrete properties. Literature was reviewed and concrete core samples were obtained from the Trojan Nuclear Plant and tested. The literature primarily addressed effects of crude or lubricating oils that are known to cause concrete damage. However, these materials have significantly different characteristics relative to the materials used as tendon sheathing fillers. Examination and testing of the concrete cores indicated that the appearance of tendon sheathing filler on the concrete surface was due to leakage from the conduits and its subsequent migration through cracks that were present. Migration of the tendon sheathing filler was confined to the cracks and there was no perceptible movement into the concrete. Results of compressive strength testing indicated that the concrete quality was consistent in the containment and that the strength had increased over 40% in 25.4 years relative to the average compressive strength at 28-days age.

NUREG/CR-6604: RADTRAD: A SIMPLIFIED MODEL FOR RADIONUCLIDE TRANSPORT AND REMOVAL AND DOSE ESTIMATION. HUMPHREYS,S.L.; MILLER,L.A.; et al. Sandia National Laboratories. HEAMES,T.J. . April 1998. 408pp. 9805180342. SAND98-0272. A3424:001.

This report documents the RADTRAD computer code developed for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation to estimate transport and removal of radionuclides and dose at selected receptors. The document includes a users' guide to the code, a description of the technical basis for the code, the quality assurance and code acceptance testing documentation, and a programmers' guide. The RADTRAD code can be used to estimate the containment release using either the TID-14844 or NUREG-1465 source terms, and assumptions, or a user-specified table. In addition, the code can account for the reduction in the quantity of radioactive material due to containment sprays, natural deposition, filters, and other natural and engineered safety features. The RADTRAD code uses a combination of tables and/or numerical models of source term reduction phenomena to determine the time dependent dose at user specified locations for a given accident scenario. The code also provides the inventory, decay chain, and dose conversion factors needed for the dose calculation. The RADTRAD code can be used for occupational radiation exposure assessments, typically in the control room, for site boundary dose estimates, and for dose attenuation estimates due to facility or accident sequence modifications.

NUREG/CR-6605: AN EVALUATION OF HUMAN FACTORS RESEARCH FOR ULTRASONIC INSERVICE INSPECTION. POND,D.J.; DONOHOO,D.T.; HARRIS,R.V. Battelle Memorial Institute, Pacific Northwest National Laboratory. March 1998. 41pp. 9803260380. PNNL-11797. A2726:291.

This work was undertaken to determine if human factors research has yielded information applicable to upgrading requirements in ASME Boiler and Pressure Vessel Code Section XI, improving methods and techniques in Section V, and/or suggesting relevant research. A preference was established for information and recommendations which have become accepted and standard practice. Manual Ultrasonic Testing/Inservice Inspection (UT/ISI) is a complex task subject to influence by dozens of variables. This review frequently revealed equivocal findings regarding effects of environmental variables as well as repeated indications that inspection performance may be more, and more reliability, influenced by the workers' social environment, including managerial practices, than by other situational variables. Also of significance are each inspectors relevant knowledge, skills, and abilities, and determination of these is seen as a necessary first step in upgrading requirements, methods, and techniques as well as in focusing research in support of such programs. While understanding the effects and mediating mechanisms of the variables impacting inspection performance is a worthwhile pursuit for researchers, initial improvements in industrial UT/ISI performance may be achieved by implementing practices already known to mitigate the effects of potentially adverse conditions.

NUREG/CR-6606: INVESTIGATION OF TECHNIQUES FOR THE DEVELOPMENT OF SEISMIC DESIGN BASIS USING THE PROBABILISTIC SEISMIC HAZARD ANALYSIS. BERNREUTER,D.L.; BOISSONNADE,A.; SHORT,C.M. Lawrence Livermore National Laboratory. April 1998. 168pp. 9805060107. UCRL-ID-128920. A3321:113.

The Nuclear Regulatory Commission asked Lawrence Livermore Laboratory to form a group of experts to assist them in revising the seismic and geologic siting criteria for nuclear power plants, Appendix A to 10 CFR Part 100. This document describes a deterministic approach for determining a safe Shutdown Earthquake (SSE) Ground Motion for a nuclear power plant site. One disadvantage of this approach is the difficulty of integrating differences of opinions and differing interpretations into seismic hazard characterization. In answer to this, probabilistic seismic hazard assessment methodologies incorporate differences of opinions and interpretations among earth science experts. For this reason, probabilistic hazard methods were selected for determining SSEs for the revised regulation, 10 CFR Part 100.23. However, because these methodologies provide a composite analysis of all possible earthquakes that may occur, they do not provide the familiar link between seismic design loading requirements and engineering design practice. Therefore, approaches used to characterize seismic events (magnitude and distance) which best represent the ground motion level determined with the probabilistic hazard analysis were investigated. This report summarizes investigations conducted at 69 nuclear reactor sites in the central and eastern U.S. for determining SSEs using probabilistic analyses. Alternative techniques are presented along with justification for key choices.

NUREG/CR-6608: SUMMARY AND EVALUATION OF LOW-VELOCITY IMPACT TEST OF SOLID STEEL BILLET ONTO CONCRETE PADS. WITTE,M.C.; HOVINGH,J.; MOK,G.C.; et al. Lawrence Livermore National Laboratory. February 1998. 184pp. 9802250129. UCRL-129211. A2281:001.

Spent fuel storage casks intended for use at independent spent fuel storage installations are evaluated during the application and review process for low-velocity impacts representative of possible handling accidents. In the past, the analyses involved in these evaluations have assumed that the casks dropped or tipped onto an unyielding surface-a conservative and simplifying assumption. Applicants are currently seeking a more realistic model for the analyses to predict the effect of a cask dropping onto a reinforced concrete pad, including energy absorbing aspects such as cracking and flexure. To develop data suitable for benchmarking these analyses, the NRC has conducted several series of drop-test studies of a solid steel billet and of a near-full-scale empty cask. This report contains a summary and evaluation of all steel billet testing conducted by Sandia National Laboratories and Lawrence Livermore National Laboratory. A series of finite element analyses of the billet testing is described and benchmarked against the test data. A method to apply the benchmarked finite element model of the soil and concrete pad to an analysis of a full-size storage cask is provided. In addition, an application to a "generic" full-size cask is presented for side and end drops, and tipover events.

NUREG/CR-6611: RESULTS OF PRESSURE LOCKING AND THERMAL BINDING TESTS OF GATE VALVES. DEWALL,K.G.; WATKINS,J.C.; MCKELLAR,M.G.; et al. Idaho National Engineering & Environmental Laboratory. May 1998. 69pp. 9806080225. INEELEXT9800161. A3688:237.

The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, is funding the Idaho National Engineering and Environmental Laboratory (INEEL) in performing research investigating the performance of gate valves subjected to pressure locking and thermal binding conditions. Pressure locking and thermal binding are phenomena that make a closed gate valve difficult to open. If the loads associated with pressure locking or thermal binding are very high, the actuator might not have the capacity to open the valve. We tested a flexible-wedge gate valve and a double-disc gate valve under pressure locking and thermal binding conditions. The results show that these valves are susceptible to pressure locking; however, they are not significantly affected by thermal binding. The results also show that seat leakage affects the bonnet pressurization rate when the valve is subjected to thermally induced pressure locking conditions.

NUREG/CR-6613 V01: CODE MANUAL FOR MACCS2.User's Guide. CHANIN,D. Technadyne Engineering Consultants, Inc. YOUNG,M.L. Sandia National Laboratories. May 1998. 300pp. 9806150058. SAND97-0594. A3823:001.

This report describes the use of the MACCS2 code. The document is primarily a user's guide, though some model description information is included. MACCS2 represents a major enhancement of its predecessor MACCs, the MELCOR Accident Consequence Code System. MACCS2, distributed by government code centers since 1990, was developed to evaluate the impacts of severe accidents at nuclear power plants on the surrounding public. The principal phenomena considered are atmospheric transport and deposition under time-variant meteorology, short and long-term mitigative actions and exposure pathways, deterministic and stochastic health effects, and economic costs. No other U.S. code that is publicly available at present offers all these capabilities. MACCS2 was developed as a general-purpose tool applicable to diverse reactor and nonreactor facilities licensed by the Nuclear Regulatory Commission or operated by the Department of Energy or the Department of Defense. The MACCS2 package includes three primary enhancements: (1) a more flexible emergency-response model, (2) an expanded library of radionuclides, and (3) a semidynamic food chain model. Other improvements are in the areas of phenomenological modeling and new output options. Initial installation of the code, written in FORTRAN77, requires a 486 or higher IBM-compatible PC with 8 MB of RAM.

NUREG/CR-6613 V02: CODE MANUAL FOR MACCS2.Preprocessor Codes COMIDA2, FGRDCF, IDCF2. CHANIN,D. Technadyne Engineering Consultants, Inc. YOUNG,M.L. Sandia National Laboratories. May 1998. 102pp. 9806150062. SAND97-0594. A3799:129.

This report is a user's guide for the preprocessors developed for the MACCS2 code. MACCS2 represents a major enhancement of its predecessor MACCS, the MELCOR Accident Consequence Code System. MACCS, distributed by government code centers since 1990, was developed to evaluate the impacts of severe accidents at nuclear power plants on the surrounding public. The principal phenomena considered are atmospheric transport and deposition under time-variant meteorology, short and long-term mitigative actions and exposure pathways,deterministic and stochastic health effects, and economic costs. MACCS2 was developed as a general-purpose tool applicable to diverse reactor and nonreactor facilities licensed by the Nuclear Regulatory Commission or operated by the Department of Energy or the Department of Defense. The preprocessors available for use with the MACCS2 code are COMIDA2, DOSFAC2, FGRDCF, and IDCF2. The COMIDA2 code contains a semidynamic food chain model and generates a file of dose-to-source conversion factors that are used by MACCS2 in calculations of ingestion doses. DOSFAC2, FGRDCF, and IDCF2 generate a file of dose conversion factors that are required for MACCS2 dose calculations. The preprocessors, written in FORTRAN 77, require a 486 or higher IBM-compatible PC.

NUREG/CR-6615: A SURVEY OF REPAIR PRACTICES FOR NUCLEAR POWER PLANT CONTAINMENT METALLIC PRESSURE BOUNDARIES. OLAND,C.B.; NAUS,D.J. Oak Ridge National Laboratory. May 1998. 128pp. 9806080222. ORNL/TM-13601. A3686:178.

The Nuclear Regulatory Commission has initiated a program at the Oak Ridge National Laboratory to provide assistance in their assessment of the effects of potential degradation on the structural integrity and leaktightness of metal containment vessels and steel liners of concrete containments in nuclear power plants. One of the program objectives is to identify repair practices for restoring metallic containment pressure boundary components that have been damaged or degraded in service. This report presents issues associated with inservice condition assessments and continued service evaluations and identifies the rules and requirements for the repair and replacement of nonconforming containment pressure boundary components by welding or metal removal. Discussion topics include base and welding materials, welding procedure and performance qualifications, inspection techniques, testing methods, acceptance criteria, and documentation requirements necessary for making repairs and replacements so that the plant can be returned to a safe operating condition.

NUREG/GR-0016: THE ROLE OF TIME-DEPENDENT DEFORMATION IN INTERGRANULAR CRACK INITIATION OF ALLOY 600 STEAM GENERATOR TUBING MATERIAL. WAS,G.S.; LIAN,K. Michigan, Univ. of, Ann Arbor, MI. March 1998. 41pp. 9804200284. A3031:101.

Intergranular stress corrosion cracking (IGSCC) of two commercial alloy 600 conditions and controlled-purity Ni-18Cr-9Fe alloys were investigated using constant extension rate tensile (CERT) tests in primary water with 1 bar hydrogen overpressure at 360 degrees C and 320 degrees C. Heat treatments produced two types of microstructures in both commercial and controlled-purity alloys: one dominated by grain boundary carbides and one dominated by intragranular carbides. CERT tests conducted over a range of strain rates and at two temperatures showed that in all samples, IGSCC was the dominant failure mode. For both the commercial alloy and the controlled-purity alloys, the microstructure with grain boundary carbides showed delayed crack initiation and shallower crack depths than did the intragranular carbide microstructure under all experimental conditions, indicating that a grain boundary carbide microstructure is more resistant to IGSCC than an intragranular carbide microstructure. Observations support both the film rupture/slip dissolution mechanism and enhanced localized plasticity. Crack growth rates increased with increasing strain rate according to a power law relation with a strain rate exponent between 0.40 and 0.64. However, crack growth rate measured in m/unit strain decreased with increasing strain rate indicating an effect of either the environment or creep. The temperature dependence of the crack growth rate was consistent with the literature.

NUREG/IA-0024: APPLICATION OF RELAP5/MOD3.1 CODE TO THE LOFT TEST L3-6. PYLEV,S.S.; ROGINSKAGA,V.L. Russia. February 1998. 66pp. 9802100124. A2059:287.

A calculation of LOFT Experiment L3-6, a small-break equivalent to a 4-inch diameter rupture in the cold leg of a four-loop commercial pressurized water reactor, has been performed to help validate RELAP5/MOD3.1 for this application. The version of the code to be used is SCDAP/RELAP5/MOD3.1.8d0. Three calculations were carried out in order to study the sensitivity to change of the break nozzle superheated discharge coefficient. Conducted comparative analysis of the LOFT L3-6 experiment shows on the whole a reasonable agreement between calculated and measured data. Some discrepancies in the system pressure do not distort a picture of the transient.

NUREG/IA-0025: RELAP5/MOD3 SUBCOOLED BOILING MODEL ASSESSMENT. DEVKIN,A.S.; PODOSENOV,A.S. Russian Research Center (Kurchatov Institute). May 1998. 83pp. 9805200009. A3468:097.

This report presents the assessment of the RELAP5/Mod3 (5m5 version) code subcooled boiling process model, which is based on a variety of experiments. The accuracy of the model is confirmed for a wide range of regime parameters for the case of uniform heating along the channel. The condensation rate is rather underpredicted, which may lead to considerable errors in void fraction behavior prediction in subcooled boiling regimes for nonuniformly or unheated channels.

NUREG/BR-0050: ADVISORY COMMITTEE ON NUCLEAR WASTE - 1998 STRATEGIC PLAN AND PRIORITY ISSUES AND ACTIVITIES. * ACNW - Advisory Committee on Nuclear Waste. March 1998. 12pp. 9804200219. A3031:001.

The Advisory Committee on Nuclear Waste (ACNW) has developed a strategic plan that establishes a framework to guide it in providing independent and timely technical advice to the Nuclear Regulatory Commission on nuclear waste disposal and management issues. The plan includes near-term priority issues the Committee will consider in 1998, as well as longer term issues the Committee plans to consider in 1999 and beyond. The ACNW's strategic plan is anchored to the NRC's Strategic Plan for Fiscal years 1997 - 2002 and supports the mission, vision, and relevant goals, strategies, and substrategies identified by the agency.

NUREG/BR-0117N97-4: NMSS LICENSEE NEWSLETTTER. * Office of Nuclear Material Safety and Safeguards. February 1998. 12pp. 9804200291. A3031:143.

This newsletter contains articles that discuss recent regulatory issues and provide administrative information. It includes descriptions of recent Federal Register notices, generic communications, significant enforcement actions, and significant operational events.

NUREG/BR-0249: THE ATOMIC SAFETY AND LICENSING BOARD PANEL. * Office of Public Affairs. February 1998. 6pp. 9803190405. A2670:015.

Through the Atomic Energy Act, Congress made it possible for the public to get a full and fair hearing on civilian nuclear matters. Individuals who are directly affected by licensing action involving a facility producing or utilizing nuclear materials may participate in a formal hearing, on the record, before independent judges on the Atomic Safety and Licensing Board Panel. Hearings, routinely involve difficult interrelated questions, often at the cutting edge of science and technology, confronting highly technical and scientific theories, opinions, and research findings. In addition, NRC hearings air local concerns about the consequences of severe accidents and continue the national debate over the role nuclear power should play in meeting the nation's energy needs.

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Secondary Report Number Index

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NUREG/CR-6511 V02: STEAM GENERATOR TUBE INTEGRITY PROGRAM.Annual Report,August 1995 - September 1996.

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NUREG-1629: THE CHARACTERIZATION OF VICKER'S MICROHARDNESS INDENTATIONS AND PILE-UP PROFILES AS A STRAIN-HARDENING MICROPROBE.

Strainer Blockage
NUREG/CP-0162 V01: PROCEEDINGS OF THE TWENTY-FIFTH WATER REACTOR SAFETY INFORMATION MEETING.Plenary Sessions,Pressure Vessel Research,BWR Strainer Blockage And Other Generic Safety Issues,Environmentally Assisted Degradation Of LWR....

Stress Corrosion Cracking
NUREG/CR-6511 V02: STEAM GENERATOR TUBE INTEGRITY PROGRAM.Annual Report,August 1995 - September 1996.
NUREG/GR-0016: THE ROLE OF TIME-DEPENDENT DEFORMATION IN INTERGRANULAR CRACK INITIATION OF ALLOY 600 STEAM GENERATOR TUBING MATERIAL.

Stress Drop
NUREG/CR-6564: ANALYSES OF SOURCE SPECTRA, ATTENUATION, AND SITE EFFECTS FROM CENTRAL AND EASTERN UNITED STATES EARTHQUAKES.

Structural Engineering
NUREG/CR-6546: A DAMAGE MECHANICS BASED APPROACH TO STRUCTURAL DETERIORATION AND RELIABILITY.

Structural Performance
NUREG/CP-0162 V03: PROCEEDINGS OF THE TWENTY-FIFTH WATER REACTOR SAFETY INFORMATION MEETING.Thermal-Hydraulic Research And Codes, Digital Instrumentation And Control, Structural Performance.

Subcooled Boiling Regime
NUREG/IA-0025: RELAP5/MOD3 SUBCOOLED BOILING MODEL ASSESSMENT.

Surface Condition
NUREG/CR-6589: THE EFFECTS OF SURFACE CONDITION ON AN ULTRASONIC INSPECTION: ENGINEERING STUDIES USING VALIDATED COMPUTER MODEL.

Surface Crack
NUREG/CR-6540: STATE-OF-THE-ART REPORT ON PIPING FRACTURE MECHANICS.

Survey Instrument
NUREG-1507: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS.
NUREG/CR-6364: HUMAN PERFORMANCE IN RADIOLOGICAL SURVEY SCANNING.

System Transient
NUREG/CR-6359 V01: RAMONA-4B: A COMPUTER CODE WITH THREE-DIMENSIONAL NEUTRON KINETICS FOR BWR AND SBWR SYSTEM TRANSIENTS.Models And Correlations.
NUREG/CR-6359 V02: RAMONA-4B: A COMPUTER CODE WITH THREE-DIMENSIONAL NEUTRON KINETICS FOR BWR AND SBWR SYSTEM TRANSIENTS.User's Manual.

TLD
NUREG-0837 V17 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. July-September 1997.

TMI-2
NUREG-1626: FINAL ENVIRONMENTAL IMPACT STATEMENT FOR THE CONSTRUCTION AND OPERATION OF AN INDEPENDENT SPENT FUEL STORAGE INSTALLATION TO STORE THE THREE MILE ISLAND UNIT 2 SPENT FUEL AT THE IDAHO NATIONAL ENGINEERING AND ENVIRONMENTAL....

Tendon Sheathing
NUREG/CR-6598: AN INVESTIGATION OF TENDON SHEATHING FILLER MIGRATION INTO CONCRETE.

Thermal Aging
NUREG/CR-6537: INFLUENCE OF LONG-TERM THERMAL AGING ON THE MICROSTRUCTURAL EVOLUTION OF NUCLEAR REACTOR PRESSURE VESSEL MATERIALS.An Atom Probe Study.

Thermal Binding
NUREG/CR-6611: RESULTS OF PRESSURE LOCKING AND THERMAL BINDING TESTS OF GATE VALVES.

Thermal Hydraulic Research
NUREG/CP-0162 V03: PROCEEDINGS OF THE TWENTY-FIFTH WATER REACTOR SAFETY INFORMATION MEETING.Thermal-Hydraulic Research And Codes, Digital Instrumentation And Control, Structural Performance.

Thermal-Hydraulic
NUREG/CR-6359 V02: RAMONA-4B: A COMPUTER CODE WITH THREE-DIMENSIONAL NEUTRON KINETICS FOR BWR AND SBWR SYSTEM TRANSIENTS.User's Manual.

Thermoluminescent Dosimeter
NUREG-0837 V17 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. July-September 1997.

Title List
NUREG-0540 V19 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.November 1-30, 1997.
NUREG-0540 V19 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.December 1-31, 1997.
NUREG-0540 V20 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.January 1-31, 1998.
NUREG-0540 V20 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.February 1-28, 1998.
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NUREG-0540 V20 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.April 1-30, 1998.

Tube
NUREG/CR-6511 V02: STEAM GENERATOR TUBE INTEGRITY PROGRAM.Annual Report,August 1995 - September 1996.

Tube Rupture
NUREG-1570: RISK ASSESSMENT OF SEVERE ACCIDENT-INDUCED STEAM GENERATOR TUBE RUPTURE.

Tubing
NUREG/CR-6575: FAILURE BEHAVIOR OF INTERNALLY PRESSURIZED FLAWED AND UNFLAWED STEAM GENERATOR TUBING AT HIGH TEMPERATURE -EXPERIMENTS AND COMPARISON WITH MODEL PREDICTIONS.

Ultrasonic Inservice Inspection
NUREG/CR-6605: AN EVALUATION OF HUMAN FACTORS RESEARCH FOR ULTRASONIC INSERVICE INSPECTION.

Ultrasonic Inspection
NUREG/CR-6589: THE EFFECTS OF SURFACE CONDITION ON AN ULTRASONIC INSPECTION: ENGINEERING STUDIES USING VALIDATED COMPUTER MODEL.

Uncertainty Analysis
NUREG/CR-6545 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Early Health Effects Uncertainty Assessment.Main Report.
NUREG/CR-6545 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Early Health Effects Uncertainty Assessment.Appendices.
NUREG/CR-6555 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS.Late Health Effects Uncertainty Assessment.Main Report.
NUREG/CR-6555 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS.Late Health Effects Uncertainty Assessment.Appendices.
NUREG/CR-6571 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Uncertainty Assessment For Internal Dosimetry.Main Report.
NUREG/CR-6571 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Uncertainty Assessment For Internal Dosimetry.Appendices.

Vendor Inspection
NUREG-0040 V21 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1997.(White Book)

Vicker
NUREG-1629: THE CHARACTERIZATION OF VICKER'S MICROHARDNESS INDENTATIONS AND PILE-UP PROFILES AS A STRAIN-HARDENING MICROPROBE.

Welding
NUREG/CR-6615: A SURVEY OF REPAIR PRACTICES FOR NUCLEAR POWER PLANT CONTAINMENT METALLIC PRESSURE BOUNDARIES.

NRC Originating Organization Index (Staff Reports)

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

ADVISORY COMMITTEE(S)
ACRS - ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUREG-1125 V19: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS.1997 Annual.

ATOMIC SAFETY BOARD(S) & PANEL(S)
ATOMIC SAFETY & LICENSING BOARD PANEL
NUREG-1363 V07: ATOMIC SAFETY AND LICENSING BOARD BIENNIAL REPORT.Fiscal Years 1995 - 1996.

OFFICE OF EXECUTIVE DIRECTOR FOR OPERATIONS (EDO)
REGION 1 (POST 820201)
NUREG-0837 V17 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. July-September 1997.
OFC OF ENFORCEMENT (POST 870413)
NUREG-0940 V16 N2 P1: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED INDIVIDUAL ACTIONS.Semiannual Progress Report,July-December 1997.
NUREG-0940 V16 N2 P2: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED REACTOR LICENSEES.Semiannual Progress Report,July-December 1997.
NUREG-0940 V16 N2 P3: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MATERIAL LICENSEES.Semiannual Progress Report,July-December 1997.
NUREG-1600 R01: GENERAL STATEMENT OF POLICY AND PROCEDURE FOR NRC ENFORCEMENT ACTIONS.Enforcement Policy.
NUREG-1622: NRC ENFORCEMENT POLICY REVIEW.July 1995 - July 1997.

EDO - OFFICE OF ADMINISTRATION (PRE 870413 & POST 890205)
OFFICE OF ADMINISTRATION, DIRECTOR (POST 940714)
NUREG-0936 V16 N02: NRC REGULATORY AGENDA.Semiannual Report.July-December 1997.

EDO - OFFICE OF THE CONTROLLER (PRE 820418 & POST 890205)
OFFICE OF THE CONTROLLER (POST 890205)
NUREG-1542 V03: ACCOUNTABILITY REPORT FISCAL YEAR 1997.
DIVISION OF BUDGET & ANALYSIS (POST 890205)
NUREG-1100 V14: BUDGET ESTIMATES.Fiscal Year 1999.

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA
OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA, DIRECTOR
NUREG-0090 V20: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.Fiscal Year 1997.
NUREG-1022 R01: EVENT REPORTING GUIDELINES 10 CFR 50.72 AND 50.73.
NUREG-1187 V01: PERFORMANCE INDICATORS FOR OPERATING COMMERCIAL NUCLEAR POWER REACTORS.Data Through September 1997.
NUREG-1272 V10 N01: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA. 1996 Annual Report.
NUREG-1272 V10 N02: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA.1996 Annual Report.
NUREG-1272 V10 N03: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA.1996 Annual Report.

EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
NUREG-0430 V16: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.July 1, 1995 - June 30, 1996.(Gray Book II)
NUREG-1626: FINAL ENVIRONMENTAL IMPACT STATEMENT FOR THE CONSTRUCTION AND OPERATION OF AN INDEPENDENT SPENT FUEL STORAGE INSTALLATION TO STORE THE THREE MILE ISLAND UNIT 2 SPENT FUEL AT THE IDAHO NATIONAL ENGINEERING AND ENVIRONMENTAL....
DIVISION OF WASTE MANAGEMENT (NMSS 940403)
NUREG/CP-0163: PROCEEDINGS OF THE WORKSHOP ON REVIEW OF DOSE MODELING METHODS FOR DEMONSTRATION OF COMPLIANCE WITH THE RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION.

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF THE GENERAL COUNSEL (POST 860701)
NUREG-0980 V01 N04: NUCLEAR REGULATORY LEGISLATION.104th Congress.
NUREG-0980 V02 N04: NUCLEAR REGULATORY LEGISLATION.104th Congress.
OFFICE OF THE INSPECTOR GENERAL (POST 890417)
NUREG-1415 V10 N02: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report To Congress,October 1, 1997 - March 31, 1998.
NRC - NO DETAILED AFFILIATION GIVEN
NUREG-0304 V22 N03: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Third Quarter 1997,July-September.
NUREG-0304 V22 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1997.
NUREG-0540 V19 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.November 1-30, 1997.
NUREG-0540 V19 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.December 1-31, 1997.
NUREG-0540 V20 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.January 1-31, 1998.
NUREG-0540 V20 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.February 1-28, 1998.
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NUREG-0540 V20 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.April 1-30, 1998.
NUREG-0750 CI04: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January 1, 1991 through December 31, 1995.
NUREG-0750 V45: NUCLEAR REGULATORY COMMISSION ISSUANCES.Opinions And Decisions Of The Nuclear Regulatory Commission With Selected Orders.January-June 1997.
NUREG-0750 V46 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-September 1997.
NUREG-0750 V46 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-December 1997.
NUREG-0750 V46 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1997. Pages 49-193.
NUREG-0750 V46 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1997. Pages 195-256.
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NUREG-0750 V47 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January-March 1998.
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NUREG-0750 V47 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1998.Pages 77-260.
NUREG-0910 R03: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE.
NUREG-1575: MULTI-AGENCY RADIATION SURVEY AND SITE INVESTIGATION MANUAL (MARSSIM).Final Report.
NUREG-1627 V01: PERFORMANCE PLAN FY 1999.

EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)
DIVISION OF ENGINEERING TECHNOLOGY (POST 941217)
NUREG-0933 S22: A PRIORITIZATION OF GENERIC SAFETY ISSUES.
NUREG-1629: THE CHARACTERIZATION OF VICKER'S MICRO-HARDNESS INDENTATIONS AND PILE-UP PROFILES AS A STRAIN-HARDENING MICROPROBE.
DIVISION OF REGULATORY APPLICATIONS (POST 941217)
NUREG-0713 V18: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES,1996.Twenty-Ninth Annual Report.
NUREG-1507: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS.
NUREG/CP-0163: PROCEEDINGS OF THE WORKSHOP ON REVIEW OF DOSE MODELING METHODS FOR DEMONSTRATION OF COMPLIANCE WITH THE RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION.
DIVISION OF SYSTEMS TECHNOLOGY (POST 941217)
NUREG-1560 V01 P1: INDIVIDUAL PLANT EXAMINATION PROGRAM: PERSPECTIVES ON REACTOR SAFETY AND PLANT PERFORMANCE.Summary Report.
NUREG-1560 V02 P2-5: INDIVIDUAL PLANT EXAMINATION PROGRAM: PERSPECTIVES ON REACTOR SAFETY AND PLANT PERFORMANCE.
NUREG-1560 V03 P6: INDIVIDUAL PLANT EXAMINATION PROGRAM: PERSPECTIVES ON REACTOR SAFETY AND PLANT PERFORMANCE.Appendices.
NUREG/CR-6509: THE EFFECT OF INITIAL TEMPERATURE ON FLAME ACCELERATION AND DEFLAGRATION-TO-DETONATION TRANSITION PHENOMENON.
PROBABILISTIC RISK ANALYSIS BRANCH (POST 941217)
NUREG-1624 DRFT FC: TECHNICAL BASIS AND IMPLEMENTATION GUIDELINES FOR A TECHNIQUE FOR HUMAN EVENT ANALYSIS (ATHEANA).Draft Report For Comment.

EDO - OFFICE OF NUCLEAR REACTOR REGULATION (POST 800428)
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NUREG-0040 V21 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1997.(White Book)
NUREG-1570: RISK ASSESSMENT OF SEVERE ACCIDENT-INDUCED STEAM GENERATOR TUBE RUPTURE.

ADVISORY COMMITTEE(s)
ACNW - ADVISORY COMMITTEE ON NUCLEAR WASTE
NUREG/BR-0050: ADVISORY COMMITTEE ON NUCLEAR WASTE - 1996 STRATEGIC PLAN AND PRIORITY ISSUES AND ACTIVITIES.

EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
NUREG/BR-0117N97-4: NMSS LICENSEE NEWSLETTER.

U.S. NUCLEAR REGULATORY COMMISSION
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EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)
DIVISION OF SYSTEMS TECHNOLOGY (POST 941217)
NUREG/IA-0024: APPLICATION OF RELAP5/MOD3.1 CODE TO THE LOFT TEST L3-6.
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EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA
DIVISION OF SAFETY PROGRAMS (POST 870413)
NUREG/CR-4674 V25: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1996. A Status Report.

EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
NUREG/CR-4554 V01 R2: SCANS (SHIPPING CASK ANALYSIS SYSTEM) A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR SHIPPING CASK DESIGN REVIEW.User's Manual to Version 3a.
NUREG/CR-5502: ENGINEERING DRAWINGS FOR 10 CFR PART 71 PACKAGE APPROVALS.
NUREG/CR-6608: SUMMARY AND EVALUATION OF LOW-VELOCITY IMPACT TEST OF SOLID STEEL BILLET ONTO CONCRETE PADS.
DIVISION OF FUEL CYCLE SAFETY & SAFEGUARDS (POST 930207)
NUREG/CR-6410: NUCLEAR FUEL CYCLE FACILITY ACCIDENT ANALYSIS HANDBOOK.
DIVISION OF WASTE MANAGEMENT (NMSS 940403)
NUREG/CR-6377: EFFECTS ON RADIONUCLIDE CONCENTRATIONS BY CEMENT/GROUND-WATER INTERACTIONS IN SUPPORT OF PERFORMANCE ASSESSMENT OF LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES.

EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)
DIVISION OF ENGINEERING TECHNOLOGY (POST 941217)
NUREG/CR-4667 V24: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT-WATER REACTORS. Semiannual Report,January-June 1997.
NUREG/CR-5361: SEISMIC ANALYSIS OF PIPING.Final Program Report.
NUREG/CR-5562: DATING AND EARTHQUAKES: REVIEW OF QUATERNARY GEOCHRONOLOGY AND ITS APPLICATION TO PALEOSEISMOLOGY.
NUREG/CR-5591 V04 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For October 1992 Through March 1993.
NUREG/CR-5591 V08 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For October 1996 Through March 1997.
NUREG/CR-6412: AGING AND LOSS-OF-COOLANT ACCIDENT (LOCA) TESTING OF ELECTRICAL CONNECTIONS.
NUREG/CR-6447: RESULTS OF CRACK-ARREST TESTS ON IRRADIATED A 508 CLASS 3 STEEL.
NUREG/CR-6453: H. B. ROBINSON-2 PRESSURE VESSEL BENCHMARK.
NUREG/CR-6511 V02: STEAM GENERATOR TUBE INTEGRITY PROGRAM.Annual Report,August 1995 - September 1996.
NUREG/CR-6537: INFLUENCE OF LONG-TERM THERMAL AGING ON THE MICROSTRUCTURAL EVOLUTION OF NUCLEAR REACTOR PRESSURE VESSEL MATERIALS.An Atom Probe Study.
NUREG/CR-6540: STATE-OF-THE-ART REPORT ON PIPING FRACTURE MECHANICS.
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NUREG/CR-6554: FINITE ELEMENT ANALYSES FOR SEISMIC SHEAR WALL INTERNATIONAL STANDARD PROBLEM.
NUREG/CR-6564: ANALYSES OF SOURCE SPECTRA, ATTENUATION, AND SITE EFFECTS FROM CENTRAL AND EASTERN UNITED STATES EARTHQUAKES.
NUREG/CR-6573: "INVESTIGATING SEISMOTECTONICS IN THE EASTERN UNITED STATES USING A GEOGRAPHIC INFORMATION SYSTEM."
NUREG/CR-6575: FAILURE BEHAVIOR OF INTERNALLY PRESSURIZED FLAWED AND UNFLAWED STEAM GENERATOR TUBING AT HIGH TEMPERATURE -EXPERIMENTS AND COMPARISON WITH MODEL PREDICTIONS.
NUREG/CR-6583: EFFECTS OF LWR COOLANT ENVIRONMENTS ON FATIGUE DESIGN CURVES OF CARBON AND LOW-ALLOY STEELS.
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NUREG/CR-6598: AN INVESTIGATION OF TENDON SHEATHING FILLER MIGRATION INTO CONCRETE.
NUREG/CR-6605: AN EVALUATION OF HUMAN FACTORS RESEARCH FOR ULTRASONIC INSERVICE INSPECTION.
NUREG/CR-6606: INVESTIGATION OF TECHNIQUES FOR THE DEVELOPMENT OF SEISMIC DESIGN BASIS USING THE PROBABILISTIC SEISMIC HAZARD ANALYSIS.
NUREG/CR-6611: RESULTS OF PRESSURE LOCKING AND THERMAL BINDING TESTS OF GATE VALVES.
NUREG/CR-6615: A SURVEY OF REPAIR PRACTICES FOR NUCLEAR POWER PLANT CONTAINMENT METALLIC PRESSURE BOUNDARIES.
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NUREG/CR-6364: HUMAN PERFORMANCE IN RADIOLOGICAL SURVEY SCANNING.
NUREG/CR-6536: VERIFICATION OF THE LWRARC CODE FOR LIGHT-WATER-REACTOR AFTERHEAT RATE CALCULATIONS.
DIVISION OF SYSTEMS TECHNOLOGY (POST 941217)
NUREG/CR-6119 V01 R1: MELCOR COMPUTER CODE MANUALS.Primer And Users' Guides, Version 1.8.4,July 1997.
NUREG/CR-6119 V02 R1: MELCOR COMPUTER CODE MANUALS.Reference Manuals,Version 1.8.4,July 1997.
NUREG/CR-6359 V01: RAMONA-4B: A COMPUTER CODE WITH THREE-DIMENSIONAL NEUTRON KINETICS FOR BWR AND SBWR SYSTEM TRANSIENTS.Models And Correlations.
NUREG/CR-6359 V02: RAMONA-4B: A COMPUTER CODE WITH THREE-DIMENSIONAL NEUTRON KINETICS FOR BWR AND SBWR SYSTEM TRANSIENTS.User's Manual.
NUREG/CR-6472: PRELIMINARY PHENOMENA IDENTIFICATION AND RANKING TABLES FOR SIMPLIFIED BOILING WATER REACTOR LOSS-OF-COOLANT ACCIDENT SCENARIOS.
NUREG/CR-6479: TECHNICAL BASIS FOR ENVIRONMENTAL QUALIFICATION OF MICROPROCESSOR-BASED SAFETY-RELATED EQUIPMENT IN NUCLEAR POWER PLANTS.
NUREG/CR-6509: THE EFFECT OF INITIAL TEMPERATURE ON FLAME ACCELERATION AND DEFLAGRATION-TO-DETONATION TRANSITION PHENOMENON.
NUREG/CR-6534 V02: FRAPCON-3: A COMPUTER CODE FOR THE CALCULATION OF STEADY-STATE, THERMAL-MECHANICAL BEHAVIOR OF OXIDE FUEL RODS FOR HIGH BURNUP.
NUREG/CR-6534 V03: FRAPCON-3: INTEGRAL ASSESSMENT.
NUREG/CR-6544: METHODOLOGY FOR ANALYZING PRECURSORS TO EARTHQUAKE-INITIATED AND FIRE-INITIATED ACCIDENT SEQUENCES.
NUREG/CR-6545 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Early Health Effects Uncertainty Assessment.Main Report.
NUREG/CR-6545 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Early Health Effects Uncertainty Assessment.Appendices.
NUREG/CR-6555 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS.Late Health Effects Uncertainty Assessment.Main Report.
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NUREG/CR-6571 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Uncertainty Assessment For Internal Dosimetry.Appendices.
NUREG/CR-6579: DIGITAL I&C SYSTEMS IN NUCLEAR POWER PLANTS.Risk-Screening Of Environmental Stressors And A Comparison Of Hardware Unavailability With An Existing Analog System.
NUREG/CR-6613 V01: CODE MANUAL FOR MACCS2.User's Guide.
NUREG/CR-6613 V02: CODE MANUAL FOR MACCS2.Preprocessor Codes COMIDA2, FGRDCF, IDCF2.

EDO - OFFICE OF NUCLEAR REACTOR REGULATION (POST 800428)
OFFICE OF NUCLEAR REACTOR REGULATION (POST 941001)
NUREG/CR-6577: U.S. NUCLEAR POWER PLANT OPERATING COST AND EXPERIENCE SUMMARIES.
NUREG/CR-6604: RADTRAD: A SIMPLIFIED MODEL FOR RADIONUCLIDE TRANSPORT AND REMOVAL AND DOSE ESTIMATION.

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

ADVANCED SYSTEMS CONCEPTS ASSOCIATES

NUREG/CR-6544: METHODOLOGY FOR ANALYZING PRECURSORS TO EARTHQUAKE-INITIATED AND FIRE-INITIATED ACCIDENT SEQUENCES.

ARGONNE NATIONAL LABORATORY

NUREG/CR-4667 V24: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT-WATER REACTORS. Semiannual Report,January-June 1997.

NUREG/CR-6511 V02: STEAM GENERATOR TUBE INTEGRITY PROGRAM.Annual Report,August 1995 - September 1996.

NUREG/CR-6575: FAILURE BEHAVIOR OF INTERNALLY PRESSURIZED FLAWED AND UNFLAWED STEAM GENERATOR TUBING AT HIGH TEMPERATURE -EXPERIMENTS AND COMPARISON WITH MODEL PREDICTIONS.

NUREG/CR-6583: EFFECTS OF LWR COOLANT ENVIRONMENTS ON FATIGUE DESIGN CURVES OF CARBON AND LOW-ALLOY STEELS.

BATTELLE MEMORIAL INSTITUTE, COLUMBUS LABORATORIES

NUREG/CR-6540: STATE-OF-THE-ART REPORT ON PIPING FRACTURE MECHANICS.

BATTELLE MEMORIAL INSTITUTE, PACIFIC NORTHWEST NATIONAL LABORATORY

NUREG/CR-6377: EFFECTS ON RADIONUCLIDE CONCENTRATIONS BY CEMENT/GROUND-WATER INTERACTIONS IN SUPPORT OF PERFORMANCE ASSESSMENT OF LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES.

NUREG/CR-6534 V02: FRAPCON-3: A COMPUTER CODE FOR THE CALCULATION OF STEADY-STATE, THERMAL-MECHANICAL BEHAVIOR OF OXIDE FUEL RODS FOR HIGH BURNUP.

NUREG/CR-6534 V03: FRAPCON-3: INTEGRAL ASSESSMENT.

NUREG/CR-6589: THE EFFECTS OF SURFACE CONDITION ON AN ULTRASONIC INSPECTION: ENGINEERING STUDIES USING VALIDATED COMPUTER MODEL.

NUREG/CR-6605: AN EVALUATION OF HUMAN FACTORS RESEARCH FOR ULTRASONIC INSERVICE INSPECTION.

BOSTON COLLEGE, WESTON, MA

NUREG/CR-6573: "INVESTIGATING SEISMOTECTONICS IN THE EASTERN UNITED STATES USING A GEOGRAPHIC INFORMATION SYSTEM."

BROOKHAVEN NATIONAL LABORATORY

NUREG-1507: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS.

NUREG/CP-0162 V01: PROCEEDINGS OF THE TWENTY-FIFTH WATER REACTOR SAFETY INFORMATION MEETING.Plenary Sessions,Pressure Vessel Research,BWR Strainer Blockage And Other Generic Safety Issues,Environmentally Assisted Degradation Of LWR....

NUREG/CP-0162 V02: PROCEEDINGS OF THE TWENTY-FIFTH WATER REACTOR SAFETY INFORMATION MEETING.Human Reliability Analysis And Human Performance Evaluation, Technical Issues Related To Rulemakings, Risk-Informed, Performance-Based Initiatives...

NUREG/CP-0162 V03: PROCEEDINGS OF THE TWENTY-FIFTH WATER REACTOR SAFETY INFORMATION MEETING.Thermal-Hydraulic Research And Codes, Digital Instrumentation And Control, Structural Performance.

NUREG/CR-6359 V01: RAMONA-4B: A COMPUTER CODE WITH THREE-DIMENSIONAL NEUTRON KINETICS FOR BWR AND SBWR SYSTEM TRANSIENTS.Models And Correlations.

NUREG/CR-6359 V02: RAMONA-4B: A COMPUTER CODE WITH THREE-DIMENSIONAL NEUTRON KINETICS FOR BWR AND SBWR SYSTEM TRANSIENTS.User's Manual.

NUREG/CR-6364: HUMAN PERFORMANCE IN RADIOLOGICAL SURVEY SCANNING.

NUREG/CR-6472: PRELIMINARY PHENOMENA IDENTIFICATION AND RANKING TABLES FOR SIMPLIFIED BOILING WATER REACTOR LOSS-OF-COOLANT ACCIDENT SCENARIOS.

NUREG/CR-6479: TECHNICAL BASIS FOR ENVIRONMENTAL QUALIFICATION OF MICROPROCESSOR-BASED SAFETY-RELATED EQUIPMENT IN NUCLEAR POWER PLANTS.

NUREG/CR-6509: THE EFFECT OF INITIAL TEMPERATURE ON FLAME ACCELERATION AND DEFLAGRATION-TO-DETONATION TRANSITION PHENOMENON.

NUREG/CR-6554: FINITE ELEMENT ANALYSES FOR SEISMIC SHEAR WALL INTERNATIONAL STANDARD PROBLEM.

NUREG/CR-6579: DIGITAL I&C SYSTEMS IN NUCLEAR POWER PLANTS.Risk-Screening Of Environmental Stressors And A Comparison Of Hardware Unavailability With An Existing Analog System.

CALIFORNIA, UNIV. OF, SANTA BARBARA, CA

NUREG-1629: THE CHARACTERIZATION OF VICKER'S MICROHARDNESS INDENTATIONS AND PILE-UP PROFILES AS A STRAIN-HARDENING MICROPROBE.

NUREG/CR-6564: ANALYSES OF SOURCE SPECTRA, ATTENUATION, AND SITE EFFECTS FROM CENTRAL AND EASTERN UNITED STATES EARTHQUAKES.

DEFENSE, DEPT. OF

NUREG-1575: MULTI-AGENCY RADIATION SURVEY AND SITE INVESTIGATION MANUAL (MARSSIM).Final Report.

DELFT UNIVERSITY OF TECHNOLOGY

NUREG/CR-6545 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Early Health Effects Uncertainty Assessment.Main Report.

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NUREG/CR-6571 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Uncertainty Assessment For Internal Dosimetry.Appendices.

ENERGY, DEPT. OF

NUREG-1575: MULTI-AGENCY RADIATION SURVEY AND SITE INVESTIGATION MANUAL (MARSSIM).Final Report.

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NUREG-1575: MULTI-AGENCY RADIATION SURVEY AND SITE INVESTIGATION MANUAL (MARSSIM).Final Report.

EQE INTERNATIONAL

NUREG/CR-6544: METHODOLOGY FOR ANALYZING PRECURSORS TO EARTHQUAKE-INITIATED AND FIRE-INITIATED ACCIDENT SEQUENCES.

EUROPEAN COMMUNITIES, COMMISSION OF

NUREG/CR-6571 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Uncertainty Assessment For Internal Dosimetry.Main Report.

NUREG/CR-6571 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Uncertainty Assessment For Internal Dosimetry.Appendices.

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NUREG/CR-6537: INFLUENCE OF LONG-TERM THERMAL AGING ON THE MICROSTRUCTURAL EVOLUTION OF NUCLEAR REACTOR PRESSURE VESSEL MATERIALS.An Atom Probe Study.

FUTURE RESOURCES ASSOCIATES, INC.
NUREG/CR-6544: METHODOLOGY FOR ANALYZING PRECURSORS TO EARTHQUAKE-INITIATED AND FIRE-INITIATED ACCIDENT SEQUENCES.

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NUREG/CR-6555 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS.Late Health Effects Uncertainty Assessment.Main Report.
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NUREG/CR-6611: RESULTS OF PRESSURE LOCKING AND THERMAL BINDING TESTS OF GATE VALVES.

INSTITUTE FOR MATERIALS RESEARCH
NUREG-1629: THE CHARACTERIZATION OF VICKER'S MICROHARDNESS INDENTATIONS AND PILE-UP PROFILES AS A STRAIN-HARDENING MICROPROBE.

JOHNS HOPKINS UNIV., BALTIMORE, MD
NUREG/CR-6546: A DAMAGE MECHANICS BASED APPROACH TO STRUCTURAL DETERIORATION AND RELIABILITY.

LAWRENCE LIVERMORE NATIONAL LABORATORY
NUREG/CR-4554 V01 R2: SCANS (SHIPPING CASK ANALYSIS SYSTEM) A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR SHIPPING CASK DESIGN REVIEW.User's Manual to Version 3a.
NUREG/CR-5502: ENGINEERING DRAWINGS FOR 10 CFR PART 71 PACKAGE APPROVALS.
NUREG/CR-6606: INVESTIGATION OF TECHNIQUES FOR THE DEVELOPMENT OF SEISMIC DESIGN BASIS USING THE PROBABILISTIC SEISMIC HAZARD ANALYSIS.
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MASSACHUSETTS INSTITUTE OF TECHNOLOGY, CAMBRIDGE, MA
NUREG/CR-6544: METHODOLOGY FOR ANALYZING PRECURSORS TO EARTHQUAKE-INITIATED AND FIRE-INITIATED ACCIDENT SEQUENCES.

MICHIGAN, UNIV. OF, ANN ARBOR, MI
NUREG/GR-0016: THE ROLE OF TIME-DEPENDENT DEFORMATION IN INTERGRANULAR CRACK INITIATION OF ALLOY 600 STEAM GENERATOR TUBING MATERIAL.

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NUREG/CR-6571 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Uncertainty Assessment For Internal Dosimetry.Main Report.
NUREG/CR-6571 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Uncertainty Assessment For Internal Dosimetry.Appendices.

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NUCLEAR POWER ENGINEERING CORP.
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NUREG-1507: MINIMUM DETECTABLE CONCENTRATIONS WITH TYPICAL RADIATION SURVEY INSTRUMENTS FOR VARIOUS CONTAMINANTS AND FIELD CONDITIONS.
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NUREG/CR-4674 V25: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1996. A Status Report.
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NUREG/CR-6453: H. B. ROBINSON-2 PRESSURE VESSEL BENCHMARK.
NUREG/CR-6479: TECHNICAL BASIS FOR ENVIRONMENTAL QUALIFICATION OF MICROPROCESSOR-BASED SAFETY-RELATED EQUIPMENT IN NUCLEAR POWER PLANTS.
NUREG/CR-6536: VERIFICATION OF THE LWRARC CODE FOR LIGHT-WATER-REACTOR AFTERHEAT RATE CALCULATIONS.
NUREG/CR-6537: INFLUENCE OF LONG-TERM THERMAL AGING ON THE MICROSTRUCTURAL EVOLUTION OF NUCLEAR REACTOR PRESSURE VESSEL MATERIALS.An Atom Probe Study.
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NUREG/CR-6615: A SURVEY OF REPAIR PRACTICES FOR NUCLEAR POWER PLANT CONTAINMENT METALLIC PRESSURE BOUNDARIES.

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NUREG/CR-4667 V24: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT-WATER REACTORS. Semiannual Report,January-June 1997.

SANDIA NATIONAL LABORATORIES
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NUREG/CR-6119 V02 R1: MELCOR COMPUTER CODE MANUALS.Reference Manuals,Version 1.8.4,July 1997.
NUREG/CR-6412: AGING AND LOSS-OF-COOLANT ACCIDENT (LOCA) TESTING OF ELECTRICAL CONNECTIONS.
NUREG/CR-6479: TECHNICAL BASIS FOR ENVIRONMENTAL QUALIFICATION OF MICROPROCESSOR-BASED SAFETY-RELATED EQUIPMENT IN NUCLEAR POWER PLANTS.
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NUREG/CR-6604: RADTRAD: A SIMPLIFIED MODEL FOR RADIONUCLIDE TRANSPORT AND REMOVAL AND DOSE ESTIMATION.
NUREG/CR-6613 V01: CODE MANUAL FOR MACCS2.User's Guide.
NUREG/CR-6613 V02: CODE MANUAL FOR MACCS2.Preprocessor Codes COMIDA2, FGRDCF, IDCF2.

SCIENCE APPLICATIONS INTERNATIONAL CORP. (FORMERLY SCIENCE APPLICATIONS,
NUREG-0713 V18: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES,1996.Twenty-Ninth Annual Report.
NUREG/CR-4674 V25: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1996. A Status Report.
NUREG/CR-6410: NUCLEAR FUEL CYCLE FACILITY ACCIDENT ANALYSIS HANDBOOK.
NUREG/CR-6579: DIGITAL I&C SYSTEMS IN NUCLEAR POWER PLANTS.Risk-Screening Of Environmental Stressors And A Comparison Of Hardware Unavailability With An Existing Analog System.

TECHNADYNE ENGINEERING CONSULTANTS, INC.
NUREG/CR-6613 V01: CODE MANUAL FOR MACCS2.User's Guide.
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NUREG/CR-6555 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS.Late Health Effects Uncertainty Assessment.Appendices.

VANDERBILT UNIV., NASHVILLE, TN
NUREG/CR-5562: DATING AND EARTHQUAKES: REVIEW OF QUATERNARY GEOCHRONOLOGY AND ITS APPLICATION TO PALEOSEISMOLOGY.

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.

RUSSIA
NUCLEAR SAFETY INSTITUTE
NUREG/IA-0024: APPLICATION OF RELAP5/MOD3.1 CODE TO THE LOFT TEST L3-6.
NUREG/IA-0025: RELAP5/MOD3 SUBCOOLED BOILING MODEL ASSESSMENT.

Licensed Facility Index

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

50-261 H.B. Robinson Plant, Unit 2, Carolina Power & Light Co. NUREG/CR-6453

72-0020 INEL TMI-2 ISFSI, NUREG-1626

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10. SUPPLEMENTARY NOTES

L. L. Stevenson, Project Manager

11. ABSTRACT *(200 words or less)*

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

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