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Vol. 17, No. 3

Regulatory and Technical Reports (Abstract Index Journal)

Compilation for
Third Quarter 1992
July – September

U.S. Nuclear Regulatory Commission

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Regulatory Publications Branch
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Office of Administration
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PREFACE

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

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

Secondary Report Number Index
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NRC Contract Sponsor Index (Contractor Reports)
Contractor Index
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Licensed Facility Index

A detailed explanation of the entries precedes each index

The bibliographic elements of the main citations are the following:

Staff Report

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

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

Conference Report

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

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

Contractor Report

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

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

International Agreement Report

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

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

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

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

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NRC Report Codes

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

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

All these report codes are controlled and assigned by the staff of the Publishing and Translations Section of the NRC Division of Publications Services.

Main Citations and Abstracts

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

NUREG-0040 V16 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,April - June 1992.(White Book) * Division of Reactor Inspection & Safeguards (870411-921003). July 1992. 129pp. 9209020331. 62939:252.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from April through June 1992.

NUREG-0090 V15 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.January-March 1992. * Office for Analysis & Evaluation of Operational Data, Director. July 1992. 30pp. 9208060263. 62647:335.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event that the Nuclear Regulatory Commission determines to be significant from the standpoint of public health and safety and requires a quarterly report of such events to be made to Congress. This report covers the period January through March 1992. Three abnormal occurrences involving medical therapy misadministrations at NRC-licensed facilities are discussed in this report. There were no abnormal occurrences at a nuclear power plant, and none were reported by NRC's Agreement States. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0090 V15 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1992. * Office for Analysis & Evaluation of Operational Data, Director. September 1992. 27pp. 9210130169. 63462:036.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event that the Nuclear Regulatory Commission determines to be significant from the standpoint of public health and safety and requires a quarterly report of such events to be made to Congress. This report covers the period April through June 1992. Three abnormal occurrences involving medical therapy misadministrations and one involving a medical diagnostic misadministration at NRC-licensed facilities are discussed in this report. There was one abnormal occurrence at a nuclear power plant, and none were reported by NRC's Agreement States. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0304 V17 N01: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1992,January-March. * Division of Freedom of Information & Publications Services (Post 890205). June 1992. 41pp. 9207270307. 62508:242.

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-0386 D06 R3: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission, Appeal Board And Licensing Board Decisions.July 1972 - September 1991. * Office of the General Counsel (Post 860701). August 1992. 596pp. 9209170066. 63143:001.

This 3rd revision of the sixth edition of the NRC Practice and Procedure Digest contains a digest of a number of Commission, Atomic Safety and Licensing Appeal Board, and Atomic Safety and Licensing Board decisions issued during the period of July 1, 1972 to September 30, 1991, interpreting the NRC's Rules of Practice in 10 CFR Part 2.

NUREG-0525 V01: SAFEGUARDS SUMMARY EVENT LIST (SSEL).Pre-NRC Through December 31, 1989. YARDUMIAN,J.; FADDEN,M. Division of Safeguards & Transportation (Post 870413). July 1992. 479pp. 9209170124. 63145:001.

The Safeguards Summary Event List provides brief summaries of hundreds of safeguards-related events involving nuclear material or facilities regulated by the U.S. Nuclear Regulatory Commission. Events are described under the categories: bomb-related, intrusion, missing/allegedly stolen, transportation-related, tampering/vandalism, arson, firearms-related, radiological sabotage, non-radiological sabotage, and miscellaneous. Because of the public interest, the miscellaneous section also includes events reported involving source material, byproduct material, and natural uranium, which are exempt from safeguards requirements. Information in the event descriptions was obtained from official NRC reports.

NUREG-0525 V02: SAFEGUARDS SUMMARY EVENT LIST (SSEL).January 1, 1990 Through December 31, 1991. YARDUMIAN,J.; FADDEN,M. Division of Safeguards & Transportation (Post 870413). July 1992. 174pp. 9209170120. 63147:001.

See NUREG-0525,V01 abstract.

NUREG-0540 V14 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.May 1-31, 1992. * Division of Freedom of Information & Publications Services (Post 890205). July 1992. 565pp. 9208060272. 62651:001.

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

NUREG-0540 V14 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.June 1-30, 1992. * Division of Freedom of Information & Publications Services (Post 890205). August 1992. 304pp. 9209020320. 62936:001.

See NUREG-0540,V14,N05 abstract.

NUREG-0540 V14 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.July 1-31, 1992. * Division of Freedom of Information & Publications Services (Post 890205). September 1992. 349pp. 9209220359. 63242:001.

See NUREG-0540,V14,N05 abstract.

NUREG-0750 V35 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1992.Pages 189-203. * Division of Freedom of Information & Publications Services (Post 890205). July 1992. 21pp. 9208060257. 62654:221.

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

NUREG-0750 V35 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1992.Page 205-260. * Division of Freedom of Information & Publications Services (Post 890205). August 1992. 63pp. 9209020326. 62939:189.

See NUREG-0750,V35,N05 abstract.

NUREG-0837 V12 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. April-June 1992. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). September 1992. 232pp. 9209240250. 63279:046.

This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facilities throughout the country for the second quarter of 1992.

NUREG-0933 S14: A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R.; RIGGS,R.; MILSTEAD,W.; et al. Division of Safety Issue Resolution (Post 880717). August 1992. 82pp. 9209170127. 63146:120.

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

NUREG-0936 V11 N02: NRC REGULATORY AGENDA.Quarterly Report,April-June 1992. * Division of Freedom of Information & Publications Services (Post 890205). July 1992. 151pp. 9208250253. 62884:001.

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

NUREG-0940 V11 N02: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June 1992. * Ofc of Enforcement (Post 870413). August 1992. 451pp. 9209170097. 63141:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1992) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

NUREG-1145 V08: U.S. NUCLEAR REGULATORY COMMISSION 1991 ANNUAL REPORT. * Office of Administration (Post 890205). July 1992. 270pp. 9208260241. 62882:001.

This report covers the major activities, events, decisions and planning that took place during fiscal year 1991 within the U.S. Nuclear Regulatory Commission (NRC) or involving the NRC.

NUREG-1214 R10: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. ALLENSPACH,F. Division of Licensee Performance & Quality Evaluation (870411-921003). August 1992. 127pp. 9209170112. 63140:076.

The Historical Data Summary of the Systematic Assessment of Licensee Performance (SALP) is produced periodically by the U.S. Nuclear Regulatory Commission. This summary provides the results of the assessment for each facility by NRC region. It is further divided into the following sections: Section 1 presents the most recent SALP report ratings for facilities in operation and under construction; Section 2 presents a chronological listing of all SALP report ratings for each operating facility; Section 3 presents a chronological listing of all SALP report ratings for each facility under construction. For historical purposes, past construction ratings for facilities that recently have been licensed also are listed in Section 3.

NUREG-1242 V01: NRC REVIEW OF ELECTRIC POWER RESEARCH INSTITUTE'S ADVANCED LIGHT WATER REACTOR UTILITY REQUIREMENTS DOCUMENT.Program Summary. * Associate Director for Advanced Reactors & License Renewal (Post 910918). August 1992. 138pp. 9209240187. 63278:268.

The staff of the U.S. Nuclear Regulatory Commission has prepared Volume 1 of a safety evaluation report (SER), "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document - Program Summary," to document the results of its review of the Electric Power Research Institute's "Advanced Light Water Reactor Utility Requirements Document." This SER provides a discussion of the overall purpose and scope of the Requirements Document, the background of the staff's review, the review approach used by the staff, and a summary of the policy and technical issues raised by the staff during its review.

NUREG-1242 V02 P01: NRC REVIEW OF ELECTRIC POWER RESEARCH INSTITUTE'S ADVANCED LIGHT WATER REACTOR UTILITY REQUIREMENTS DOCUMENT. Evolutionary Plant Designs.Chapter 1. * Associate Director for Advanced Reactors & License Renewal (Post 910918). August 1992. 513pp. 9209240193. 63277:084.

The staff of the U.S. Nuclear Regulatory Commission has prepared Volume 2 (Parts 1 and 2) of a safety evaluation report (SER), "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document - Evolutionary Plant Designs," to document the results of its review of the Electric Power Research Institute's "Advanced Light Water Reactor Utility Requirements Document." This SER gives the results of the staff's review of Volume II of the Requirements Document for evolutionary plant designs, which consists of 13 chapters and contains utility design requirements for an evolutionary nuclear power plant (approximately 1300 megawatts - electric).

NUREG-1242 V02 P02: NRC REVIEW OF ELECTRIC POWER RESEARCH INSTITUTE'S ADVANCED LIGHT WATER REACTOR UTILITY REQUIREMENTS DOCUMENT. Evolutionary Plant Designs.Chapters 2-13. * Associate Director for Advanced Reactors & License Renewal (Post 910918). August 1992. 610pp. 9209240198. 63274:001.

See NUREG-1242,V02,P01 abstract.

NUREG-1272 V06 N01: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA.1991 Annual Report - Power Reactors. * Office for Analysis & Evaluation of Operational Data, Director. July 1992. 244pp. 9209020312. 62938:305.

The annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) is devoted to the activities performed during 1991. The report is published in two separate parts. NUREG-1272, Vol. 6, No. 1, covers power reactors and presents an overview of the operating experience of the nuclear power industry from the NRC perspective, including comments about the trends of some key performance measures. The report also includes the principal findings and issues identified in AEOD studies over the past year and summarizes information from such sources as licensee event reports, diagnostic evaluations, and reports to the NRC's Operations Center. The reports contain a discussion of the Incident Investigation Team program and summarize the Incident Investigation Team and Augmented Inspection Team reports for that group of licensees. NUREG-1272, Vol. 6, No. 2, covers nonreactors and presents a review of the events and concerns during 1991 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations. Each volume contains a list of the AEOD reports issued for 1984-1990.

NUREG-1272 V06 N02: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA.1991 Annual Report - Nonreactors. * Office for Analysis & Evaluation of Operational Data, Director. August 1992. 143pp. 9209170074. 63142:092.

The annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) is devoted to the activities performed during 1991. The report is published in two separate parts. NUREG-1272, Vol. 6, No. 1, covers power reactors and presents an overview of the operating experience of the nuclear power industry from the NRC perspective, including comments about the trends of some key performance measures. The report also includes the principal findings and issues identified in AEOD studies over the past year and summarizes information from such sources as licensee event reports, diagnostic evaluations, and reports to the NRC's Operations Center. The reports contain a discussion of the Incident Investigation Team program and summarize the Incident Investigation Team and Augmented Inspection Team reports for that group of licensees. NUREG-1272, Vol. 6, No. 2, covers nonreactors and presents a review of the events and concerns during 1991 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations. Each volume contains a list of the AEOD reports issued for 1981-1991.

NUREG-1377 R03: NRC RESEARCH PROGRAM ON PLANT AGING: LISTING AND SUMMARIES OF REPORTS ISSUED THROUGH JULY 1992. KONDIC,N.N. Division of Engineering (Post 870413). September 1992. 108pp. 9210130165. 63462:065.

The U.S. Nuclear Regulatory Commission is conducting the Nuclear Plant Aging Research (NPAR) Program. This is a comprehensive hardware- oriented engineering research program focused on understanding the aging mechanisms of components and systems in nuclear power plants. The NPAR program also focuses on methods for simulating and monitoring the aging- related degradation of these components and systems. In addition, it provides recommendations for effective maintenance to manage aging and for the implementation of the research results in the regulatory process. This document contains a listing and index of reports generated in the NPAR program that were issued through July 1992 and summaries of those reports. Each summary describes the elements of the research covered in the report and outlines the significant results. For the convenience of the user, the reports are indexed by personal author, corporate author, and subject.

NUREG-1423 V03: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON NUCLEAR WASTE.July 1991 - June 1992. * Advisory Committee on Nuclear Waste. August 1992. 81pp. 9209170105. 63140:002.

This compilation contains 19 reports issued by the Advisory Committee on Nuclear Waste (ACNW) during the fourth year of its operation. The reports were submitted to the Chairman and Commissioners of the U.S. Nuclear Regulatory Commission, the Executive Director for Operations, the Director, Office of Nuclear Regulatory Research, or to the Director, Office of Nuclear Material Safety and Safeguards. All reports prepared by the Committee have been made available to the public through the NRC Public Document Room and the U.S. Library of Congress.

NUREG-1442 R01: EMERGENCY RESPONSE RESOURCES GUIDE.For Nuclear Power Plant Emergencies. WEINSTEIN,E.; BATES,G. NRC - No Detailed Affiliation Given. * Federal Emergency Management Agency. July 1992. 43pp. 9208240333. FEMA-REP-17. 62666:077.

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

NUREG-1451: STAFF TECHNICAL POSITION ON INVESTIGATIONS TO IDENTIFY FAULT DISPLACEMENT HAZARDS AND SEISMIC HAZARDS AT A GEOLOGIC REPOSITORY. MCCONNELL,K.I.; BLACKFORD,M.E.; IBRAHIM,A-B. Division of High-Level Waste Management (Post 870413). July 1992. 68pp. 9209170131. 63140:203.

10 CFR Part 60 does not specify the manner in which potential fault displacement hazards and seismic hazards at a candidate site for a geologic repository are to be identified. The purpose of this staff technical position (STP), therefore, is to provide guidance to the U.S. Department of Energy (DOE) on acceptable geologic repository investigations that can be used to identify fault displacement hazards and seismic hazards. The staff considers that the approach this STP takes to investigations of fault displacement and seismic phenomena is appropriate for the collection of sufficient data for input to analyses of fault displacement hazards and seismic hazards, both for the preclosure and postclosure performance periods. However, detailed analyses of fault displacement and seismic data, such as those required for detailed assessments of repository performance, may identify the need for additional investigations. Section 2.0 of this STP describes the 10 CFR Part 60 requirements that form the basis for investigations to describe the fault displacement hazards and seismic hazards at a geologic repository. Technical position statements and corresponding discussions are presented in Sections 3.0 and 4.0, respectively. Technical position topics in this STP are categorized thusly: (1) investigation considerations; (2) investigations for fault-displacement hazards, and (3) investigation for seismic hazards.

NUREG-1457: RESOURCES AVAILABLE FOR NUCLEAR POWER PLANT EMERGENCIES UNDER THE PRICE-ANDERSON ACT AND THE ROBERT T. STAFFORD DISASTER RELIEF AND EMERGENCY ASSISTANCE ACT. WEINSTEIN,E. Division of Operational Assessment (Post 870413). July 1992. 23pp. 9208170156. 62791:295.

Through a series of TABLETOP exercises and other events that involved participation by State and Federal organizations, the need was identified for further explanation of financial and other related resources available to individuals and State and local governments in a major emergency at a nuclear power plant. A group with representatives from the Nuclear Regulatory Commission, the Federal Emergency Management Agency, and the American Nuclear Insurers/Mutual Atomic Energy Liability Underwriters was established to work toward this end. This report is the result of that effort.

NUREG-1465 DRFT FC: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS. Draft Report For Comment. SOFFER,L.; BURSON,S.B.; FERRELL,C.M.; et al. Division of Safety Issue Resolution (Post 880717). June 1992. 46pp. 9208100123. 62680:298.

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

NUREG-1470 V01: CHIEF FINANCIAL OFFICER'S ANNUAL REPORT -- 1992. * Office of the Controller (Post 890205). September 1992. 28pp. 9210060022. 63424:307.

The Chief Financial Officers Act of 1990 requires the NRC Chief Financial Officer to prepare and submit an annual report to the agency head and the Director of the Office of Management and Budget. This 1992 report is the first annual report for the NRC and includes a description and analysis of the status of financial management and a summary of the reports on internal accounting and administrative control systems.

NUREG/CP-0120: PROCEEDINGS OF THE FIFTH WORKSHOP ON CONTAINMENT INTEGRITY. Held In Washington,DC,May 12-14,1992. PARKS,M.B.; HUGHEY,C.E. Sandia National Laboratories. July 1992. 654pp. 9208170148. SAND92-0173. 62790:001.

The Fifth Workshop on Containment Integrity was held in Washington, DC, on May 12-14, 1992. The purpose of these workshops is to provide an international forum for the exchange of information on performance of containments in nuclear power plants under severe accident loadings. Severe accident investigations of existing containment designs as well as future advanced containments were presented during the workshop. There were 145 participants at the workshop from 15 countries. Ivan Selin, Chairman of the NRC, provided the opening address for the meeting. A total of 39 papers were presented on the following topics: Containment Design Considerations for Severe Accident Conditions, Advanced Containment Designs and Related Research, Containment Behavior Under Accident Conditions, Testing/Analysis of Containment Systems, and Containment Operational Experience (Leakage, Aging, and Operation). A copy of the final program, including last minute changes, is provided in these proceedings. Papers that were presented at the workshop make up the body of this report. The workshop was hosted by Sandia National Laboratories under the sponsorship of the U.S. Nuclear Regulatory Commission. Principal organizers for the workshop were James F. Costello of the U.S. Nuclear Regulatory Commission and Walter A. von Riesemann and M. Brad Parks of Sandia National Laboratories.

NUREG/CP-0122 V01: PROCEEDINGS OF THE AGING RESEARCH INFORMATION CONFERENCE. BERANEK,A.F. Division of Engineering (Post 870413). September 1992. 560pp. 9209290377. 63337:136.

This report presents the proceedings of the Aging Research Information Conference held at the Holiday Inn Crowne Plaza in Rockville, Maryland, on March 24-27, 1992. This conference was held to disseminate research results in the area of nuclear power plant aging from programs sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission. The conference also provided an opportunity for engineers and scientists from around the world to exchange technical information and discuss future international cooperation. The papers and talks appear in the order in which they were presented at the conference, and they are grouped by technical session.

NUREG/CP-0122 V02: PROCEEDINGS OF THE AGING RESEARCH INFORMATION CONFERENCE. BERANEK,A.F. Division of Engineering (Post 870413). September 1992. 465pp. 9209290380. 63336:031.

See NUREG/CP-0122,V01 abstract.

NUREG/CP-0123: PROCEEDINGS OF THE SECOND NRC/ASME SYMPOSIUM ON PUMP AND VALVE TESTING.Held At The Hyatt Regency Hotel,Washington,DC, July 21-23, 1992. * EG&G Idaho, Inc. July 1992. 562pp. 9207270274. EGG-2676. 62510:001.

The 1992 Symposium on Pump and Valve Testing, jointly sponsored by the Board on Nuclear Codes and Standards of the American Society of Mechanical Engineers and by the Nuclear Regulatory Commission, provides a forum for the discussion of current programs and methods for inservice testing and motor-operated valve testing at nuclear power plants. The symposium also provides an opportunity to discuss the need to improve that testing in order to help ensure the reliable performance of pumps and valves. The participation of industry representatives, regulators, and consultants results in the discussion of a broad spectrum of ideas and perspectives regarding the improvement of inservice testing of pumps and valves at nuclear power plants.

NUREG/CP-0124: WORKSHOP ON THE USE OF PRA METHODOLOGY FOR THE ANALYSIS OF REACTOR EVENTS AND OPERATIONAL DATA. RASMUSON,D.M. Division of Safety Programs (Post 870413). DINGMAN,S. Sandia National Laboratories. June 1992. 133pp. 9208070150. 62676:055.

A workshop entitled, "The Use of PRA Methodology for the Analysis of Reactor Events and Operational Data" was held on January 29-30, 1992 in Annapolis, Maryland. Over 50 participants from the NRC, its contractors, and others participated in the meetings. During the first day, presentations were made by invited speakers to discuss issues in relevant topics. On the second day, discussion groups were held to focus on three areas: (1) risk significance of operational events; (2) industry risk profile and generic concerns; and (3) risk monitoring and risk-based performance indicators. Summaries of the discussion sessions are contained in the report as well as important insights gained from the discussions.

NUREG/CR-2907 V10: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1989. TICHLER,J.; NORDEN,K.; CONGEMI,J. Brookhaven National Laboratory. September 1992. 300pp. 9210050068. BNL-NUREG-51581. 63384:238.

Releases of radioactive materials in airborne and liquid effluents from commercial light water reactors during 1989 have been compiled and reported. Data on solid waste shipments as well as selected operating information have been included. This report supplements earlier annual reports issued by the former Atomic Energy Commission and the Nuclear Regulatory Commission. The 1989 release data are summarized in tabular form. Data covering specific radionuclides are summarized.

NUREG/CR-3320 V02: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.PSF Startup Experiments. MCELROY,W.N.; GOLD,R.; MCGARRY,E.D. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1992. 76pp. 9208260297. WHC-EP-0204. 62890:136.

The metallurgical irradiation experiment at the Oak Ridge Research Reactor Poolside Facility (ORR-PSF) is one of the series of benchmark experiments in the framework of the Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP). The goal of this program is to test, against well-established benchmarks, the methodologies and data bases that are used to predict the irradiation embrittlement and fracture toughness of pressure vessel and support structure steels. The prediction methodology includes procedures for neutron physics calculations, dosimetry and spectrum adjustment methods, metallurgical tests, and damage correlations. The benchmark experiments serve to validate, improve, and standardize these procedures. The results of this program are implemented in a set of ASTM Standards on pressure vessel surveillance procedures. These, in turn, may be used as guides for the nuclear industry and for the Nuclear Regulatory Commission (NRC). To serve as a benchmark, a very careful characterization of the ORR-PSF experiment is necessary, both in terms of neutron flux-fluence spectra and of metallurgical test results. Statistically determined uncertainties must be given in terms of variances and covariances to make comparisons between predictions and experimental results meaningful. Detailed descriptions of the PSF physics-dosimetry startup experiments and their results are reported.

NUREG/CR-3950 V07: FUEL PERFORMANCE ANNUAL REPORT FOR 1989. BAILEY,W.J.; BERTING,F.M. Battelle Memorial Institute, Pacific Northwest Laboratory. WU,S.L. Division of Systems Technology (890827-921003). June 1992. 228pp. 9207270321. PNL-5210. 62508:310.

This annual report provides a brief description of fuel performance during 1989 in commercial nuclear power plants. Brief summaries of fuel design changes, fuel surveillance programs, fuel operating experience and trends, fuel problems, high-burnup fuel experience, and items of general significance are provided. References to additional, more detailed information and related NRC evaluations are included.

NUREG/CR-4356 V01: TRAC-BF1/MOD1: AN ADVANCED BEST-ESTIMATE COMPUTER PROGRAM FOR BWR ACCIDENT ANALYSIS.Model Description. BORKOWSKI,J.A.; WADE,N.L.; GILES,M.M.; et al. EG&G Idaho, Inc. August 1992. 306pp. 9209220469. EGG-2626. 63224:001.

The TRAC-BWR code development program at the Idaho National Engineering Laboratory has developed versions of the Transient Reactor Analysis Code (TRAC) for the U.S. Nuclear Regulatory Commission and the public. The TRAC-BF1/MOD1 version of the computer code provides a best- estimate analysis capability for analyzing the full range of postulated accidents in boiling water reactor (BWR) systems and related facilities. This version provides a consistent and unified analysis capability for analyzing all areas of a large- or small-break loss-of-coolant accident (LOCA), beginning with the blowdown phase and continuing through heatup, reflood with quenching, and, finally, the refill phase of the accident. Also provided is a basic capability for the analysis of operational transients up to and including anticipated transients without scram (ATWS). The TRAC-BF1/MOD1 version produces results consistent with previous versions. Assessment calculations using the two TRAC-BF1 versions show overall improvements in agreement with data and computation times as compared to earlier versions of the TRAC-BWR series of computer codes.

NUREG/CR-4356 V02: TRAC-BF1/MOD1:AN ADVANCED BEST-ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR ACCIDENT ANALYSIS.User's Guide. RETTIG,W.H.; WADE,N.L.; GILES,M.M.; et al. EG&G Idaho, Inc. June 1992. 297pp. 9208100136. EGG-2626. 62680:001.

See NUREG/CR-4356.V01 abstract.

NUREG/CR-4391: TRAC/BF1-MOD1 MODELS AND CORRELATIONS. BORKOWSKI,J.A.; WADE,N.L.; ROUHANI,S.Z.; et al. EG&G Idaho, Inc. August 1992. 458pp. 9209220474. EGG-2680. 63239:001.

The TRAC-BWR code development program at the Idaho National Engineering Laboratory has developed versions of the Transient Reactor Analysis Code (TRAC) for the U.S. Nuclear Regulatory Commission and the public. The TRAC-BF1/MOD1 version of the computer code provides a best- estimate analysis capability for analyzing the full range of postulated accidents in boiling water reactor (BWR) systems and related facilities. This version provides a consistent and unified analysis capability for analyzing all areas of a large- or small-break loss-of-coolant accident (LOCA), beginning with the blowdown phase and continuing through heatup, reflood with quenching, and, finally, the refill phase of the accident. Also provided is a basic capability for the analysis of operational transients up to and including anticipated transients without scram (ATWS). The TRAC-BF1/MOD1 version produces results consistent with previous versions. Assessment calculations using the two TRAC-BF1 versions show overall improvements in agreement with data and computation times as compared to earlier versions of the TRAC-BWR series of computer codes.

NUREG/CR-4409 V04: DATA BASE ON DOSE REDUCTION RESEARCH PROJECTS FOR NUCLEAR POWER PLANTS. KHAN,T.A.; VULIN,D.S.; LIANG,H.; et al. Brookhaven National Laboratory. August 1992. 225pp. 9209220454. BNL-NUREG-51934. 63238:001.

This is the fourth volume in a series of reports that provide information on dose reduction research and health physics technology for nuclear power plants. The information is taken from a data base maintained by Brookhaven National Laboratory's ALARA Center for the Nuclear Regulatory Commission. This report presents information on 118 new or updated projects, covering a wide range of activities. Projects including steam generator degradation, decontamination, robotics improvements in reactor materials, and inspection techniques, among others, are described in the research section of the report. The section on health physics technology includes some simple and very cost- effective projects to reduce radiation exposures. Included in this volume is a detailed description of how to access the BNL data bases which store this information. All project abstracts from this report, as well as many other useful documents, can be accessed, with permission, through our online system, ACE. A computer equipped with a modem, or a fax machine, is all that is required to connect to ACE. Many features of ACE, including software, hardware, and communications specifics, are explained in this report.

NUREG/CR-4469 V13: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report. October 1990-March 1991. DOCTOR,S.R.; GOOD,M.S.; HEASLER,P.G.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1992. 88pp. 9208250264. PNL-5711. 62870:001.

The Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors (NDE Reliability) Program at the Pacific Northwest Laboratory was established by the Nuclear Regulatory Commission to determine the reliability of current inservice inspection (ISI) techniques and to develop recommendations that will ensure a suitably high inspection reliability. The objectives of this program include determining the reliability of ISI performed on the primary systems of commercial light-water reactors (LWRs); using probabilistic fracture mechanics analysis to determine the impact of NDE unreliability on system safety; and evaluating reliability improvements that can be achieved with improved and advanced technology. A final objective is to formulate recommended revisions to ASME Code and Regulatory requirements, based on material properties, service conditions, and NDE uncertainties. The program scope is limited to ISI of the primary systems including the piping, vessel, and other components inspected in accordance with Section XI of the ASME Code. This is a progress report covering the programmatic work from October 1990 through March 1991.

NUREG/CR-4469 V14: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report.April 1991-September 1991. DOCTOR,S.R.; DIAZ,A.A.; FRILEY,J.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1992. 82pp. 9208250273. PNL-5711. 62871:228.

The Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors (NDE Reliability) Program at the Pacific Northwest Laboratory was established by the Nuclear Regulatory Commission to determine the reliability of current inservice inspection (ISI) techniques and to develop recommendations that will ensure a suitably high inspection reliability. The objectives of this program include determining the reliability of ISI performed on the primary systems of commercial light-water reactors (LWRs); using probabilistic fracture mechanics analysis to determine the impact of NDE unreliability on system safety; and evaluating reliability improvements that can be achieved with improved and advanced technology. A final objective is to formulate recommended revisions to ASME Code and Regulatory requirements, based on material properties, service conditions, and NDE uncertainties. The program scope is limited to ISI of the primary systems including the piping, vessel, and other components inspected in accordance with Section XI of the ASME Code. This is a progress report covering the programmatic work from April 1991 through September 1991.

NUREG/CR-4599 V02 N1: SHORT CRACKS IN PIPING AND PIPING WELDS.Semiannual Report, April-September 1991. WILKOWSKI,G.M.; BRUST,F.; FRANCINI,R.; et al. Battelle Memorial Institute, Columbus Laboratories. September 1992. 215pp. 9209280056. BMI-2173. 63335:001.

This is the third semiannual report of the U.S. Nuclear Regulatory Commission's Short Cracks in Piping and Piping Welds research program. This 4-year program began in March 1990. The overall objective of this program is to verify and improve fracture analyses for circumferentially cracked large-diameter nuclear piping with crack sizes typically used in leak-before-break analyses or inservice flaw evaluations.

NUREG/CR-4667 V14: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1991 - March 1992. CHUNG,H.M.; KASSNER,T.F.; MAJUMDAR,S.; et al. Argonne National Laboratory. August 1992. 65pp. 9209220422. ANL-92/30. 63240:263.

This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking in light water reactors during the six months from October 1991 through March 1992. Topics that have been investigated during this period include: (1) fatigue and stress corrosion cracking of low-alloy steel used in piping and in steam generator and reactor pressure vessels; (2) radiation-induced segregation and irradiation-assisted SCC of Type 304 SS after accumulation of relatively high fluence; and (3) update of a crack growth data base for austenitic and ferritic steels in high- temperature water. Existing data on fatigue of low-alloy steel in LWR environments have been reviewed. Based on fracture-mechanics models and engineering judgement, interim fatigue design curves are being developed that are consistent with available fatigue-life data. Microchemical and microstructural changes in high- and commercial-purity Type 304 SS specimens from control-blade absorber tubes and a control-blade sheath from operating BWRs were studied by Auger electron spectroscopy and scanning electron microscopy. Slow-strain-rate-tensile tests were conducted on irradiated specimens in air and in simulated BWR water at 289 degrees C. Crack growth data on fracture-mechanics specimens of austenitic and ferritic steels in simulated BWR water, developed in this program over the past 8 years, are compiled into a data base along with references that contain details of test methods, material compositions, metallographic information, and comparisons of data with predictions of Section XI of the ASME Code.

NUREG/CR-4674 V15: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1991 A STATUS REPORT.Main Report And Appendix A. MINARICK,J.W. Science Applications International Corp. (formerly Science Applications, Inc.). CLETCHER,J.W.; COPINGER,D.A.; et al. Oak Ridge National Laboratory. September 1992. 170pp. 9210070028. ORNL/NOAC-232. 63427:005.

Twenty-eight operational events with conditional probabilities of core damage of 1.0 X 10(-6) or higher occurring at commercial light- water reactors during 1991 are considered to be precursors to potential severe core damage. These are described along with associated significance estimates, categorization, and subsequent analyses. This study is a continuation of earlier work, which evaluated the 1969 to 1981 and 1984 to 1990 events. The report discusses (1) the general rationale for this study, (2) the selection and documentation of events as precursors, (3) the estimation and use of conditional probabilities of subsequent severe core damage to rank precursor events, and (4) the plant models used in the analysis process.

NUREG/CR-4674 V16: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1991 A STATUS REPORT.Appendices B,C, And D. MINARICK,J.W. Science Applications International Corp. (formerly Science Applications, Inc.). CLETCHER,J.W.; COPINGER,D.A.; et al. Oak Ridge National Laboratory. September 1992. 600pp. 9210070034. ORNL/NOAC-232. 63425:003.

See NUREG/CR-4674,V15 abstract.

NUREG/CR-4744 V06 N1: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,October 1990 - March 1991. CHOPRA,O.K. Argonne National Laboratory. August 1992. 199pp. 9209220414. ANL-91/22. 63241:001.

This progress report summarizes work performed by Argonne National Laboratory on long-term thermal embrittlement of cast duplex stainless steels in LWR systems during the six months from October 1990 to March 1991. Charpy-impact, tensile, and fracture toughness data are presented for several heats of cast stainless steel that were aged up to 58,000 h at temperatures of 290-400° C. The results indicate that thermal aging increases the tensile stress and decreases the fracture toughness of the materials. In general, CF-3 steels are the least sensitive to thermal aging embrittlement and CF-8M steels are the most sensitive. The increase in flow stress of fully aged cast stainless steels is ≈10% for CF-3 steels and ≈20% for CF-8 and CF-8M steels. The fracture toughness J_{IC} and average tearing modulus for heats that are sensitive to thermal aging (e.g., CF-8M steels) are as low as ≈90 kJ/m^2 and ≈60, respectively.

NUREG/CR-4819 V02: AGING AND SERVICE WEAR OF SOLENOID-OPERATED VALVES USED IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS.Evaluation Of Monitoring Methods. KRYTER,R.C. Oak Ridge National Laboratory. July 1992. 91pp. 9208260239. ORNL/TM-12038. 62878:006.

Solenoid-operated valves (SOVs) were studied at Oak Ridge National Laboratory as part of the USNRC Nuclear Plant Aging Research (NPAR) Program. The primary objective of the study was to identify, evaluate, and recommend methods for inspection, surveillance, monitoring, and maintenance of SOVs that can help ensure their operational readiness- that is, their ability to perform required safety functions under all anticipated operating conditions, since failure of one of these small and relatively inexpensive devices could have serious consequences under certain circumstances. An earlier (Phase I) NPAR program study described SOV failure modes and causes and identified measurable parameters thought to be linked to the progression of ever-present degradation mechanisms that may ultimately result in functional failure of the valve. Using this earlier work as a guide, the present (Phase II) study focused on devising and then demonstrating the effectiveness of techniques and equipment with which to measure performance parameters that show promise for detecting the presence and trending the progress of such degradations before they reach a critical stage. Intrusive techniques requiring the addition of magnetic or acoustic sensors or the application of special test signals were investigated briefly, but major emphasis was placed on the examination of condition-indicating techniques that can be applied with minimal cost and impact on plant operation. These include monitoring coil mean temperature remotely by means of coil dc resistance or ac impedance, determining valve plunger position by means of coil ac impedance, verifying unrestricted SOV plunger movement by measuring current and voltage at their critical bistable (pull-in and drop-out) values, and detecting the presence of shorted turns or insulation breakdown within the solenoid coil using interrupted-current test methods. The first of these techniques, though perhaps the simplest conceptually, will likely benefit the nuclear industry most because SOVs have a history of failure in service as a result of unwitting operation at excessive temperatures.

NUREG/CR-4832 V01: ANALYSIS OF THE LASALLE UNIT 2 NUCLEAR POWER PLANT: RISK METHODS INTEGRATION AND EVALUATION PROGRAM (RMIEP).Summary. PAYNE,A.C. Sandia National Laboratories. July 1992. 130pp. 9208060095. SAND92-0537. 62652:207.

This volume presents an overview of the methodology and results of the integrated accident sequence analysis (Level I) of the LaSalle Unit 2 nuclear power plant performed as part of the Level III PRA performed by Sandia National Laboratories for the Nuclear Regulatory Commission. The Level II/III results are presented in associated reports described in the Foreword. This volume contains a summary description of the LaSalle plant, describes the contents of the other nine volumes of this report and their relationships to each other, the relationship of the LaSalle program to other programs, a step-by-step summary description of the methodology and new techniques used to perform the analysis, and presents the integrated results obtained by merging all of the accident sequence cut sets from the LOCA, transient, transient-induced LOCAs, and anticipated accidents without scram accident sequences resulting from internal initiators with the cut-sets from the fire, flood, and seismic analyses accident sequences.

NUREG/CR-4832 V02: ANALYSIS OF THE LASALLE UNIT 2 NUCLEAR POWER PLANT: RISK METHODS INTEGRATION AND EVALUATION PROGRAM (RMIEP). Integrated Quantification And Uncertainty Analysis. PAYNE,A.C.; SYPE,T.T.; WHITEHEAD,D.W.; et al. Sandia National Laboratories. July 1992. 669pp. 9208060243. SAND92-0537. 62648:001.

This volume presents the methodology and results of the integrated accident sequence analysis of the LaSalle Unit II nuclear power plant. Integrated results are obtained by merging all of the accident sequences' cut sets from the internal and external events analyses. The final dominant accident sequences are determined and the integrated risk reduction, risk increase, and uncertainty importance measures are obtained. Also, an overall ranking of the dominant cut sets was obtained. The total core damage frequency at LaSalle from all events has a mean value of 1.01E-04/R-yr. with a 5th percentile of 5.34E-6/R- yr., a median value of 2.92E-05/R-yr., and a 95th percentile of 2.93E-04/R-yr. The dominant accident, 35.4% of the core damage frequency, involves a loss of all injection as a result of failures occurring after a loss of offsite power. The dominant cut sets of this sequence represent a short-term station blackout type accident. The second most likely sequence, 17.2% of the core damage frequency, is the result of a control room fire which is not suppressed and becomes large enough to require evacuation of the control room. Auto actuation of the systems fails as a result of the fire and the operators do not operate the remote shutdown panel correctly due to the high stress. Loss of all injection occurs and short-term core damage results.

NUREG/CR-4832 V03 P1: ANALYSIS OF THE LASALLE UNIT 2 NUCLEAR POWER PLANT: RISK METHODS INTEGRATION AND EVALUATION PROGRAM (RMIEP).Internal Events Accident Sequence Quantification.Main Report. PAYNE,A.C.; DANIEL,S.L.; WHITEHEAD,D.W.; et al. Sandia National Laboratories. August 1992. 161pp. 9209220470. SAND92-0537. 63244:026.

This volume presents the methodology and results of the internal event accident sequence analysis of the LaSalle Unit 2 nuclear power plant performed as part of the Level III Probabilistic Risk Assessment being performed by Sandia National Laboratories for the Nuclear Regulatory Commission. The total internal core damage frequency has a mean valve of 4.41E-05/R-yr with a 5th percentile of 2.05E-6/R-yr, a median value of 1.64E-05/R-yr, and a 95th percentile of 1.39E-04/R-yr. The dominant sequences involve a loss of off-site power (LOSP), immediate or delayed failure of on-site AC power resulting in station-blackout, and failure of the reactor core isolation cooling system (RCIC). The events most important to risk reduction are: frequency of LOSP, non-recovery of offsite power within one hour, diesel generator (DG) cooling water pump common mode failure, and non-recoverable isolation of RCIC during station blackouts. The events most important to risk increase are: failure of various AC power circuit breakers resulting in partial loss of onsite AC power, failure to scram, and DG cooling water common mode failure. The dominant contributors to uncertainty are: control circuit failure rates, relay coil failure to energize, energized relay coils failing deenergized, frequency of LOSP, and DG failure to start.

NUREG/CR-4832 V03 P2: ANALYSIS OF THE LASALLE UNIT 2 NUCLEAR POWER PLANT: RISK METHODS INTEGRATION AND EVALUATION PROGRAM (RMIEP).Internal Events Accident Sequence Quantification.Appendices. PAYNE,A.C.; DANIEL,S.L.; WHITEHEAD,D.W.; et al. Sandia National Laboratories. August 1992. 750pp. 9209220476. SAND92-0537. 63245:001.

See NUREG/CR-4832,V03,P01 abstract.

NUREG/CR-4832 V07: ANALYSIS OF THE LASALLE UNIT 2 NUCLEAR POWER PLANT: RISK METHODS INTEGRATION AND EVALUATION PROGRAM (RMIEP).External Event Scoping Quantification. RAVINDRA,M.K.; BANON,H. NTS/SMA, Inc. * Sandia National Laboratories. July 1992. 161pp. 9208060219. SAND92-0537. 62654:060.

This report is a description of the scoping quantification study which selected the external events to be included in the Level III PRA of the LaSalle County Nuclear Generating Station Unit 2. The study was performed by NTS/Structural Mechanics Associates (SMA) for Sandia National Laboratories as part of the Level I analysis being performed by the Risk Methods Integration and Evaluation Program (RMIEP). The methodology used is described in detail in a companion report, NUREG/CR-4839. In this report, we describe the process for selecting the external events, the screening analysis, and the detailed bounding calculations for those events not eliminated in the screening analysis. As a result of this analysis, it was concluded that only internal flooding, internal fire, and seismic events were potentially significant at LaSalle. Detailed analyses were performed for each of these and are reported in NUREG/CR-4832, Volumes 10, 9, and 8, respectively.

NUREG/CR-4839: METHODS FOR EXTERNAL EVENT SCREENING QUANTIFICATION: RISK METHODS INTEGRATION AND EVALUATION PROGRAM (RMIEP) METHODS DEVELOPMENT. RAVINDRA,M.K.; BANON,H. NTS/SMA, Inc. * Sandia National Laboratories. July 1992. 126pp. 9208060210. SAND87-7156. 62647:209.

In this report, the scoping quantification procedures for external events in probabilistic risk assessments of nuclear power plants are described. External event analysis in a PRA has three important goals: (1) The analysis should be complete in that all events are considered; (2) By following some selected screening criteria, the more significant events are identified for detailed analysis; (3) The selected events are analyzed in depth by taking into account the unique features of the events: hazard, fragility of structures and equipment, external-event initiated accident sequences, etc. Based on the above goals, external event analysis may be considered as a three-stage process: Stage I: Identification and Initial Screening of External Events; Stage II: Bounding Analysis; Stage III: Detailed Risk Analysis. In the present report, first, a review of published PRAs is given to focus on the significance and treatment of external events in full-scope PRAs. Except for seismic, flooding, fire, and extreme wind events, the contributions of other external events to plant risk have been found to be negligible. Second, scoping methods for external events not covered in detail in the NRC's PRA Procedures Guide are provided. For this purpose, bounding analyses for transportation accidents, extreme winds and tornadoes, aircraft impacts, turbine missiles, and chemical release are described.

NUREG/CR-5305 V01: INTEGRATED RISK ASSESSMENT FOR LASALLE UNIT 2 NUCLEAR POWER PLANT.Phenomenology And Risk Uncertainty Evaluation Program (PRUEP). BROWN,T.D.; PAYNE,A.C.; MILLER,L.A.; et al. Sandia National Laboratories. August 1992. 385pp. 9209220410. SAND90-2765. 63243:001.

A Level III probabilistic risk assessment (PRA) was performed for the LaSalle Unit 2 nuclear power plant. The objective of this study was to provide an estimate of the risk to the offsite population during full power operation of the plant and to include a characterization of the uncertainties in the calculated risk values. Uncertainties were included in the accident frequency analysis, accident progression analysis, and the source term analysis. Only weather uncertainties were included in the consequence analysis. The risk estimates presented in this report include contributions from both internal and external initiators. The offsite risk to the public due to the operation of LaSalle County Station is relatively low, especially with respect to the NRC safety goals. The mean individual early fatality risk within 1 mile is 1.1E-10/R-yr which is more than three orders of magnitude below the safety goal. Similarly, the mean individual latent cancer fatality risk is 8.5E-09/R-yr which is slightly more than two orders of magnitude below the safety goal. In fact, the entire uncertainty distributions for these two risk measures lie below the safety goals. The mean values for early fatality risk and for latent cancer fatality risk are 1.2E-08/R-yr and 0.25/R-yr, respectively.

NUREG/CR-5378: AGING DATA ANALYSIS AND RISK ASSESSMENT--DEVELOPMENT AND DEMONSTRATION STUDY. WOLFORD,A.J. DNV Technica. ATWOOD,C.L.; ROESENER,W.S.; et al. EG&G Idaho, Inc. August 1992. 253pp. 9209240315. EGG-2567. 63273:001.

This work develops and demonstrates a probabilistic risk assessment (PRA) approach to assess the effect of aging and degradation of active components on plant risk. The work: (a) develops a way to identify and quantify age-dependent failure rates of active components, and to incorporate them into PRA; (b) demonstrates the approach by applying it to a fluid-mechanical system, using the key elements of a NUREG-1150 PRA; and (c) presents it as a step-by-step approach, to be used for evaluating the risk significance of aging phenomena in systems of interest. The approach uses statistical tests to detect increasing failure rates and for testing data-pooling assumptions and model adequacy. The component failure rates are assumed to change over time, with several forms used to model the age dependence - exponential, Weibull, and linear. Confidence intervals for the age-dependent failure rates are found and used to develop inputs to a PRA model in order to determine the plant core damage frequency. The approach was used with plant-specific data, obtained from maintenance work requests for the auxiliary feedwater system of an older pressurized water reactor. The approach can be used for extrapolating present trends into the near future and for supporting risk-based aging management decisions.

NUREG/CR-5416: TECHNICAL EVALUATION OF GENERIC ISSUE 113: DYNAMIC QUALIFICATION AND TESTING OF LARGE BORE HYDRAULIC SNUBBERS. NITZEL,M.E.; WARE,A.G. EG&G Idaho, Inc. PAGE,J.D. NRC - No Detailed Affiliation Given. September 1992. 400pp. 9210130176. EGG-2571. 63461:003.

This report summarizes the work performed by the Idaho National Engineering Laboratory (INEL) for the Nuclear Regulatory Commission to resolve Generic Issue 113, "Dynamic Qualification and Testing of Large Bore Hydraulic Snubbers (LBHSs)." The report evaluates LBHS reliability and the need to improve that reliability. The INEL gathered and reviewed information regarding snubber (including LBHS) numbers and uses, design, environmental qualification, operating experience, and the effects of various snubber reduction programs. Limited qualitative and quantitative analyses were performed regarding potential LBHS single failures. A list of potential improvements to LBHS reliability was generated and each item on the list was evaluated by probabilistic risk and cost/benefit analyses. Eleven recommendations were made; five applicable to existing and future plants, five applicable only to future plants, and one for further "single-failure" research.

NUREG/CR-5443: CORE-CONCRETE INTERACTIONS USING MOLTEN URANIA WITH ZIRCONIUM ON A LIMESTONE CONCRETE BASEMAT.The SURC-1 Experiment. COPUS,E.R.; BROCKMANN,J.E.; et al. Sandia National Laboratories. BLOSE,R.E. Ktech Corp. September 1992. 275pp. 9210150151. SAND90-0087. 63525:039.

An inductively heated experiment, SURC-1, was executed as part of the Integral Core-Concrete Interactions Experiments Program. The purpose of this experiment was to measure and assess the variety of source terms produced during core debris/concrete interactions. These source terms include thermal energy released to both the reactor basemat and the containment environment, as well as flammable gas, condensable vapor and toxic or radioactive aerosols generated during the course of a severe reactor accident. The SURC-1 experiment used 204 kg of prototypic UO(2)-ZrO(2) materials to study the interactions between core debris and a limestone basement. The experiment eroded 27 cm of concrete during 130 minutes of sustained interaction at temperatures which ranged from 2,650 to 2,200 K. Comprehensive gas flow rates, gas compositions, and aerosol release rates from the interaction were also measured.

NUREG/CR-5564: CORE-CONCRETE INTERACTIONS USING MOLTEN UO(2) WITH ZIRCONIUM ON A BASALTIC BASEMAT.The SURC-2 Experiment. COPUS,E.R.; BROCKMANN,J.E.; et al. Sandia National Laboratories. BLOSE,R.E. Ktech Corp. August 1992. 297pp. 9209220462. SAND90-1022. 63237:001.

An inductively heated experiment, SURC-2, was executed as part of the Integral Core-Concrete Interactions Experiments Program. The purpose of this experiment was to measure and assess the variety of source terms produced during core debris/concrete interactions. These source terms include thermal energy released to both the reactor basemat and the containment environment, as well as flammable gas, condensable vapor and toxic or radioactive aerosols generated during the course of a severe reactor accident. The SURC-2 experiment used 200 kg of prototypic UO(2)-ZrO(2) materials to study the interactions between core debris and a basaltic basement. The experiment eroded 35 cm of concrete during 160 minutes of sustained interaction at temperatures which ranged from 2700 to 2200 K. Comprehensive gas flow rates, gas compositions, and aerosol release rates from the interaction were also measured.

NUREG/CR-5587: APPROACHES FOR AGE-DEPENDENT PROBABILISTIC SAFETY ASSESSMENTS WITH EMPHASIS ON PRIORITIZATION AND SENSITIVITY STUDIES. VESELY,W.E. Science Applications International Corp. (formerly Science Applications, Inc.). August 1992. 164pp. 9209220333. SAIC-92/1137. 63240:099.

Approaches are described for incorporating component aging reliability models into a probabilistic safety assessment (PSA), or probabilistic risk assessment (PRA), of a nuclear power plant. These approaches and procedures are described from a technical standpoint and are not to be interpreted as having any regulatory implications. Component aging failure rate models and test and maintenance aging control models are presented for utilization. Different approaches for carrying out the aging evaluations are given. Demonstrations are given involving prioritizing aging contributors, evaluating maintenance effectiveness, carrying out time dependent evaluations, and carrying out uncertainty and sensitivity analyses of aging effects.

NUREG/CR-5646: PIPING SYSTEM RESPONSE DURING HIGH-LEVEL SIMULATED SEISMIC TESTS AT THE HEISSDAMPFREAKTOR FACILITY (SHAM TEST FACILITY). STEELE,R.; NITZEL,M.E. EG&G Idaho, Inc. July 1992. 231pp. 9208260229. EGG-2655. 62876:155.

The SHAM seismic research program studied the effects of increasing levels of seismic excitation on a full-scale, in situ nuclear piping system containing a naturally aged United States (U.S.) 8-in. motor- operated gate valve. The program was conducted by Kernforschungszentrum Karlsruhe at the Heissdampfreaktor near Frankfurt, Germany. Participants included the United States, Germany, and England. Fifty- one experiments were conducted, with the piping supported by six different piping support systems, including a typical stiff U.S. piping support system of snubbers and rigid struts. This report specifically addresses the tests conducted with the U.S. system. The piping system withstood large displacements caused by overload snubber failures and local piping strains. Although some limit switch chatter was observed, the motor operator and valve functioned smoothly throughout the tests. The results indicate that sufficient safety margins exist when commonly accepted design methods are applied and that piping systems will likely maintain their pressure boundary in the presence of severe loading and the loss of multiple supports.

NUREG/CR-5673 V02: TRAC-PF1/MOD2 CODE MANUAL.User's Guide. SCHNURR,N.M.; STEINKE,R.G.; MARTINEZ,V.; et al. Los Alamos National Laboratory. July 1992. 883pp. 9208240311. LA-12031-M. 62867:001.

The Transient Reactor Analysis Code (TRAC) was developed to provide advanced best-estimate predictions of postulated accidents in pressurized light-water reactors. The code features either a one- or a three-dimensional treatment of the pressure vessel and its associated internals, a two-fluid nonequilibrium hydrodynamics model with a noncondensable gas field and solute tracking, flow-regime-dependent constitutive equation treatment, optional reflood tracking capability for bottom-flood and falling-film quench fronts, and consistent treatment of entire accident sequences including the generation of consistent initial conditions. In addition to the components contained in previous TRAC versions, TRAC-PFI/MOD2 includes a heat-structure component that allows the user to accurately model complicated geometries. An improved reflood model based on mechanistic and defensible models has been added. The new code also contains improved constitutive models and additions and refinements for several components. This guide describes the components and control systems used in TRAC and gives detailed information the user needs to prepare an input deck and carry out simulations using TRAC-PFI/MOD2.

NUREG/CR-5673 V03: TRAC-PF1/MOD2 CODE MANUAL.Programmer's Guide. GUFFEE,L.A. Science Applications International Corp. (formerly Science Applications, Inc.). WOODRUFF,S.B.; STEINKE,R.G.; et al. Los Alamos National Laboratory. July 1992. 341pp. 9208060249. LA-12031-M. 62653:001.

The Transient Reactor Analysis Code (TRAC) was developed to provide best-estimate predictions of postulated accidents of light-water reactors. The TRAC-PF1/MOD2 program provides this capability for pressurized water reactors and for many thermal-hydraulic test facilities. The code features either a one-or a three-dimensional treatment of the pressure vessel and its associated internals, a two- fluid nonequilibrium hydrodynamics model with a noncondensable gas field and solute tracking, flow-regime-dependent constitutive equation treatment, optional reflood tracking capability for bottom-flood and falling-film quench fronts, and consistent treatment of entire accident sequences, including the generation of consistent initial conditions. This manual is the third volume of a four-volume set of documentation on TRAC-PF1/MOD2. This guide was developed to assist the TRAC programmer and contains information on the TRAC code and data structure, the TRAC calculational sequence, memory management, and various machine configurations supported by TRAC.

NUREG/CR-5685: SEALING PERFORMANCE OF BENTONITE AND BENTONITE/CRUSHED ROCK BOREHOLE PLUGS. OUYANG,S.; DAEMEN,J.J.K. Arizona, Univ. of, Tucson, AZ. July 1992. 340pp. 9208260235. 62877:026.

This study includes a systematic investigation of the sealing performance of bentonite and bentonite/crushed rock plugs. American Colloid C/S granular bentonite and crushed Apache Leap tuff have been mixed to prepare samples. Bentonite weight percent and crushed tuff gradation are the major variables studied. High injection pressure flow tests, polyaxial flow tests, high temperature flow tests, and piping tests have been performed. A composition to yield a permeability lower than 5 x 10(-8) cm/s would have at least 25% bentonite by weight mixed with well-graded crushed rock. Hydraulic properties of the mixture plugs may be highly anisotropic if significant particle segregation occurs during sample installation and compaction. Temperature has no significant effect on sealing performance from room temperature to 60 degrees C. Piping damage is small if the hydraulic gradient does not exceed 120 and 280 for samples with a bentonite content of 25 and 35%, respectively. The hydraulic gradients above which flow of bentonite may take place are deemed critical. Bentonite occupancy percentage and water content at saturation are two major parameters for plug design. A model is developed for predicting the permeability in clays. A piping model, based on plastic flow theory, permits estimating the critical hydraulic gradients at which flow of bentonite takes place. The model can also be used to define the maximum allowable pore diameter i a protective filter layer.

NUREG/CR-5687: BOREHOLE STABILITY IN DENSELY WELDED TUFFS. FUENKAJORN,K.; DAEMEN,J.J.K. Arizona, Univ. of, Tucson, AZ. July 1992. 71pp. 9208250287. 62871:158.

Failure of host rock at seal locations may allow bypass flow around seals. This report presents an investigation of compressive failure around boreholes in densely welded Apache Leap tuff. Triaxial and polyaxial tests have been performed on cylinders and blocks containing coaxial cored holes. Test hole diameters are 14 mm for triaxial testing and 25.4 mm for biaxial testing. To induce breakouts requires stresses that exceed elastically calculated boundary stresses equal to the uniaxial compressive strength. Failure patterns are influenced by heterogeneity: soft inclusions fail first. Such failures remain localized. The stronger surrounding matrix maintains hole stability. An elastic analysis of hole stability in welded tuff may provide a significant safety margin. This conclusion needs to be qualified: all experiments have been conducted on small diameter boreholes. It would be desirable to conduct borehole stability experiments on larger holes. Of particular importance may be the influence of flow layers in tuff on borehole stability. Effects of flow layers have been minimized by preparing all samples normal to the flow layers. Also desirable would be an investigation of the influence of environmental conditions, especially temperature and moisture content, and of the strength under sustained long-term loading. A more comprehensive analysis of the results should be performed, including evaluation of recent theoretical models for borehole breakouts.

NUREG/CR-5700: AGING ASSESSMENT OF REACTOR INSTRUMENTATION AND PROTECTION SYSTEM COMPONENTS.Aging-Related Operating Experiences. GEHL,A.C.; HAGEN,E.W. Oak Ridge National Laboratory. July 1992. 154pp. 9208260222. ORNL/TM-11806. 62876:001.

A study of the aging-related operating experiences throughout a five year period (1984-1988) of six generic instrumentation modules (indicators, sensors, controllers, transmitters, annunciators, and recorders) was performed as a part of the USNRC Nuclear Plant Aging Research Program. The effects of aging from operational and environmental stressors were characterized from results depicted in Licensee Event Reports (LERs). The data are graphically displayed as frequency of events per plant year for operating plant ages from 1 to 28 years to determine aging-related failure trend patterns. Of the six modules studied, indicators, sensors, and controllers account for the bulk (83%) of aging-related failures. Infant mortality appears to be the dominant failure mode for most I&C module categories. Of the LERs issued during 1984-1988 which dealt with malfunctions of the six instrumentation and control modules studied, 28% were found to be aging- related (other studies show a range of 25-50%).

NUREG/CR-5758 V02: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports.CY 1991. MURPHY,S.; FLEMING,T.; WESTRA,C.; et al. Battelle Human Affairs Research Centers. August 1992. 90pp. 9209240238. PNL-7736. 63279:278.

This report summarizes the data from the semiannual reports on fitness-for-duty programs submitted to the NRC by 54 utilities for two reporting periods: January 1, 1991 to June 30, 1991, and from July 1, 1991, to December 31, 1991. During CY 1991, licensees reported that they conducted 262,597 tests for the presence of illegal drugs and alcohol. Of these tests, 1,721 (.66%) were positive. Positive test results varied by category of test and category of worker. The majority of positive test results (983) were obtained through pre-access testing. Of tests conducted on workers having access to the protected area, there were 509 positive tests from random testing, and 167 positive tests from for-cause testing. Followup testing of workers who had previously tested positive resulted in 62 positive tests. Positive test results also varied by category of worker. Overall, short-term and long-term contractor personnel had the highest rates of positive tests. Licensee employees had lower rates of positive test results.

NUREG/CR-5772 V01: AGING, CONDITION MONITORING, AND LOSS-OF-COOLANT ACCIDENT (LOCA) TESTS OF CLASS 1E ELECTRICAL CABLES.Crosslinked Polyolefin Cables. JACOBUS,M.J. Sandia National Laboratories. August 1992. 200pp. 9209240244. SAND91-1766/1. 63263:001.

This report describes the results of aging, condition monitoring, and accident testing of crosslinked polyolefin (XLPO) cables. Three sets of cables were aged for up to 9 months under simultaneous thermal (≈100°C) and radiation (≈0.10 kGy/hr) conditions. A sequential accident consisting of high dose rate irradiation (≈6 kGy/hr) and high temperature steam followed the aging. The test results indicate that most properly installed XLPO cables should be able to survive an accident after 60 years for total aging doses up to 400 kGy and for moderate ambient temperatures on the order of 50-55° C (potentially higher or lower, depending on material specific activation energies). Mechanical measurements (primarily elongation, modulus, and density) were more effective than electrical measurements for monitoring age-related degradation.

NUREG/CR-5779 V01: AGING OF NON-POWER-CYCLE HEAT EXCHANGERS USED IN NUCLEAR POWER PLANTS.Operating Experience And Failure Identification. MOYERS,J.C. Oak Ridge National Laboratory. July 1992. 70pp. 9208250292. ORNL-6687. 62871:090.

This report presents the results of an assessment of the time-related degradation of non-power-cycle heat exchangers used in nuclear power plants. The assessment was sponsored by the U.S. Nuclear Regulatory Commission's Nuclear Plant Aging Research Program. Heat exchanger design characteristics and applications in the plants are described and stressors leading to degradation are identified. Operating experience, as identified from nuclear industry data bases, is reviewed and failure types and causes are summarized. Regulatory requirements for inspection and testing, with a brief discussion of industry practices in this area, are presented.

NUREG/CR-5787 V01: TIMING ANALYSIS OF PWR FUEL PIN FAILURES.Final Report.Main Text And Appendices A-J. JONES,K.R.; WADE,N.L.; KATSMA,K.R.; et al. EG&G Idaho, Inc. September 1992. 350pp. 9210050053. EGG-2657. 63382:205.

Research has been conducted to develop and demonstrate a methodology for calculation of the time interval between receipt of containment isolation signals and the first fuel pin failure for loss- of-coolant accidents. Demonstration calculations were performed for a Babcock and Wilcox (B&W) design (Oconee) and a Westinghouse (W) four- loop design (Seabrook). Sensitivity studies assessed the impacts of fuel pin burnup, axial peaking factor, break size, emergency core cooling system availability, and main coolant pump trip on these times. The analysis was performed using FRAPCON-2 and FRAP-T6 for the calculation of steady-state and transient fuel behavior and SCDAP/RELAP5/MOD3 and TRAC-PF1/MOD1 for the calculation of transient thermal-hydraulic conditions in the reactor system. Using SCDAP/RELAP5/MOD3 and TRAC-PF1/MOD1 thermal-hydraulic data, the shortest time intervals calculated between initiation of containment isolation and fuel pin failure are, respectively, 10.4 and 10.3 seconds for Oconee and 19.1 and 29.1 seconds for Seabrook. These intervals are for a double-ended, offset-shear, cold leg break, using the technical specification maximum peaking factor applied to fuel with maximum design burnup. Using peaking factors commensurate with actual burnups would result in longer intervals for both reactor designs.

NUREG/CR-5787 V02: TIMING ANALYSIS OF PWR FUEL PIN FAILURES.Final Report. Appendices K-L. JONES,K.R.; WADE,N.L.; KATSMA,K.R.; et al. EG&G Idaho, Inc. September 1992. 400pp. 9210050062. EGG-2657. 63383:185.

See NUREG/CR-5787,V01 abstract.

NUREG/CR-5790: RISK EVALUATION FOR A B&W PRESSURIZED WATER REACTOR. EFFECTS OF FIRE PROTECTION SYSTEM ACTUATION ON SAFETY-RELATED EQUIPMENT.Evaluation Of Generic Issue 57. LAMBRIGHT,J. Sandia National Laboratories. LYNCH,J.; ROSS,S.; et al. Science & Engineering Associates, Inc. September 1992. 250pp. 9210150130. SAND91-1535. 63525:317.

Nuclear power plants have experienced inadvertent actuations of fire protection systems (FPSs) under conditions for which these systems were not intended to actuate, and also have experienced advertent actuations with the presence of a fire. These actuations have often damaged plant equipment. A review of the impact of past occurrences of both types of such events on nuclear power plant safety has been performed. Thirteen different scenarios leading to actuation of fire protection systems due to a variety of causes were identified. These scenarios ranged from inadvertent actuation caused by human error to hardware failure, and includes seismic root causes and seismic/fire interaction. A quantification of these thirteen scenarios, where applicable, was performed on a Babcock and Wilcox (B&W) Pressurized Water Reactor (lowered loop design). This report estimates the contribution of FPS actuations to core damage frequency and to risk.

NUREG/CR-5793: A COMPARISON OF ANALYSIS METHODOLOGIES FOR PREDICTING CLEAVAGE ARREST OF A DEEP CRACK IN A REACTOR PRESSURE VESSEL SUBJECTED TO PRESSURIZED-THERMAL-SHOCK LOADING CONDITIONS. KEENEY-WALKER; BASS,B.R. Oak Ridge National Laboratory. September 1992. 31pp. 9209290362. ORNL/TM-11969. 63336:001.

Several calculational procedures are compared for predicting cleavage arrest of a deep crack in the wall of a prototypical reactor pressure vessel (RPV) subjected to pressurized-thermal-shock (PTS) types of loading conditions. Three procedures examined in this study used the following models: (1) a static finite-element model (full bending); (2) a radially constrained static model; and (3) a thermoelastic dynamic finite-element model. A PTS transient loading condition was selected that produced a deep arrest of an axially oriented, initially shallow crack according to calculational results obtained from the static (full-bending) model. Results from the two static models were compared with those generated from the detailed thermoelastic dynamic finite-element analysis. The dynamic analyses modeled cleavage-crack propagation using a node-release technique and application- and generation-mode methodologies. Comparisons presented here indicate that the degree to which dynamic solutions can be approximated by static models is highly dependent on several factors, including the material dynamic fracture curves and the propensity for cleavage reinitiation of the arrested crack under PTS loading conditions. Additional work is required to develop and validate a satisfactory dynamic fracture toughness model applicable to postcleavage arrest conditions in an RPV.

NUREG/CR-5804: REPOSITORY OPERATIONAL CRITERIA ANALYSIS. HAGEMAN,J.P.; CHOWDHURY,A.H. Center for Nuclear Waste Regulatory Analyses. August 1992. 391pp. 9209240189. CNWRA 91-014. 63270:001.

The objective of the "Repository Operational Criteria (ROC) Feasibility Studies" (or ROC task) was to conduct comprehensive and integrated analyses of repository design, construction, and operations criteria in 10 CFR Part 60 regulations, considering the interfaces and impacts of any potential changes to those regulations. The study addresses regulatory criteria related to the preclosure aspects of the geologic repository. The study task developed regulatory concepts or potential repository operational criteria (PROC) based on analysis of a repository's safety functions and other regulations for similar facilities. These regulatory concepts or PROC were used as a basis to assess the sufficiency and adequacy of the current criteria in 10 CFR Part 60. Where the regulatory concepts were the same as current operational criteria, these criteria were referenced. The operations criteria referenced or the PROC developed are given in this report. Detailed analyses used to develop the regulatory concepts and any necessary PROC for those regulations that may require a minor change are also presented. The results of the ROC task showed a need for further analysis and possible major rule change related to the design bases of a geologic repository operations area, siting, and radiological emergency planning.

NUREG/CR-5810: EVALUATION OF MHTGR FUEL RELIABILITY. WICHNER,R.P. Oak Ridge National Laboratory. BARTHOLD,W.P. Barthold & Associates, Inc. July 1992. 71pp. 9208240298. ORNL/TM-12014. 62868:120.

Modular High-Temperature Gas-Cooled Reactor (MHTGR) concepts that house the reactor vessel in a tight but unsealed reactor building place heightened importance on the reliability of the fuel particle coatings as fission product barriers. Though accident consequence analyses continue to show favorable results, the increased dependence on one type of barrier, in addition to a number of other factors, has caused the Nuclear Regulatory Commission (NRC) to consider conservative assumptions regarding fuel behavior. For this purpose, the concept termed "weak fuel" has been proposed on an interim basis. "Weak fuel" is a penalty imposed on consequence analyses whereby the fuel is assumed to respond less favorably to environmental conditions than predicted by behavioral models. The rationale for adopting this penalty, as well as conditions that would permit its reduction or elimination, are examined in this report. The evaluation includes an examination of possible fuel- manufacturing defects, quality-control procedures for defect detection, and the mechanisms by which fuel defects may lead to failure.

NUREG/CR-5819: PROBABILITY AND CONSEQUENCES OF RAPID BORON DILUTION IN A PWR.A Scoping Study. DIAMOND,D.J.; KOHUT,P.; NOURBAKHSH,H.; et al. Brookhaven National Laboratory. June 1992. 98pp. 9207270295. BNL-NUREG-52313. 62512:115.

This report documents the results of a scoping study of rapid dilution events in pressurized water reactors. It reviews the subject in broad terms and focuses on one event of most interest. This event could occur during a restart if there is a loss-of-offsite power when the reactor is being deborated. If the volume control tank is filled with water at a low boron concentration then a slug of this water could accumulate in the lower plenum. This would be the result of the trip of the reactor coolant power. The concern is that this diluted slug will rapidly enter the core after a reactor coolant pump is restarted and this could cause a power excursion leading to fuel damage. This problem was studied probabilistically for three plants and the important design features that affect the core damage frequency were identified. This analysis was augmented by an analysis of the mixing of the diluted water with the borated water already present in the vessel. The mixing was found to be significant so that neglect of this mechanism in the probabilistic analysis leads to very conservative results. Neutronic calculations for one plant were carried out to understand the effect of nuclear design on the consequences of the event.

NUREG/CR-5840: FASTGRASS: A MECHANISTIC MODEL FOR THE PREDICTION OF XE, I, CS, TE, BA, AND SR RELEASE FROM NUCLEAR FUEL UNDER NORMAL AND SEVERE-ACCIDENT CONDITIONS.User's Guide For Mainframe, Workstation, And Personal Computer Applications. REST,J.; ZAWADZKI,S.A. Argonne National Laboratory. September 1992. 179pp. 9209240206. ANL-92/3. 63271:032.

The primary physical/chemical models that form the basis of the FASTGRASS mechanistic computer model for calculating fission-product release from nuclear fuel are described. Calculated results are compared with test data, and the major mechanisms affecting the transport of fission products during steady-state and accident conditions are identified.

NUREG/CR-5849 DRF FC: MANUAL FOR CONDUCTING RADIOLOGICAL SURVEYS IN SUPPORT OF LICENSE TERMINATION.Draft Report For Comment. BERGER,J.D. Oak Ridge Associated Universities. June 1992. 207pp. 9208060313. ORAU-92/C57. 62647:001.

This document describes a process for conducting radiological surveys during decommissioning, to demonstrate that residual radioactive material satisfies criteria established by the U.S. Nuclear Regulatory Commission (NRC) for termination of a license. The manual describes procedures for design and conduct of surveys in a manner which will provide a high degree of assurance that NRC guidelines and conditions have been satisfied. The manual also describes methods for documenting the survey findings in a final report to the NRC. This manual updates information contained in NUREG/CR-2082, "Monitoring for Compliance With Decommissioning Termination Survey Criteria, (ORNL 1981)." It incorporates statistical approaches to survey design and data interpretation used by the Environmental Protection Agency for evaluation of hazardous materials sites under Superfund (CERCLA). Quality assurance is emphasized throughout.

NUREG/CR-5854: UNIVERSAL TREATMENT OF PLUMES AND STRESSES FOR PRESSURIZED THERMAL SHOCK EVALUATIONS. THEOFANOUS,T.G.; ANGELINI,S.; YAN,H. California, Univ. of, Santa Barbara, CA. June 1992. 96pp. 9207270318. 62508:146.

The thermal field in a reactor vessel downcomer and resulting thermal/stress response in the adjacent reactor vessel wall during high-pressure safety injection are examined, especially with regard to departures from one-dimensional behavior. Similarity solutions for the stratification (in the cold leg) that creates the downcomer plumes, and scaling considerations for the thermal conduction and stress fields in the vessel wall are developed to provide generalized criteria for the adequacy of the one-dimensional treatment.

NUREG/CR-5859: MODELING THE INFLUENCE OF IRRADIATION TEMPERATURE AND DISPLACEMENT RATE ON RADIATION-INDUCED HARDENING IN FERRITIC STEELS. STOLLER,R.E. Oak Ridge National Laboratory. July 1992. 46pp. 9208250219. ORNL/TM-12073. 62884:152.

The influence of irradiation temperature and displacement rate have been investigated using a model based on the reaction rate theory description of radiation damage. This theory was developed primarily for the investigation of relatively high-temperature, high-dose radiation effects such as void swelling and irradiation creep. Before applying that theory to the much lower temperature and dose regimes characteristic of light water reactor pressure vessels and support structures, it is necessary to examine the assumptions made in formulating the theory. The major simplifying assumption that has commonly been made is that the interstitial and vacancy concentrations reach a quasi-steady state condition rapidly enough that the steady state concentrations can be used in calculating the observable radiation effects. The results presented here indicate that the assumption of steady state point defect concentrations is not valid for temperatures much below the light water reactor pressure vessel operating temperature of about 288 degrees C. At lower temperatures, the time required for the point defect concentrations to reach steady state can exceed an operating reactor's lifetime. Even at 288 degrees C, the point defect transient time can be long enough to influence the interpretation of irradiation experiments done in materials test reactors at accelerated damage rates. Based on the insights obtained with the simple models of point defect evolution, a more detailed model was developed that incorporates an explicit description of point defect clustering. These clusters are potentially responsible for the fraction of the radiation- induced hardening that is attributed to the so-called "matrix defect." The model considers both interstitial and vacancy clustering. The former are treated as Frank loops while the latter are treated as microvoids. The point defect clusters can be formed either directly in the displacement cascade or by diffusive encounters between free point defects. The results of molecular dynamics simulation studies are used to provide guidance for the clustering parameters. The hardening due to point defect clusters was calculated using a simple dislocation barrier model. The results indicate that both interstitial and vacancy clusters can give rise to significant hardening. The relative importance of each cluster type is shown to be a function of irradiation temperature and displacement rate.

NUREG/CR-5864: THEORETICAL AND USER'S MANUAL FOR PC-PRAISE.A Probabilistic Fracture Mechanics Computer Code For Piping Reliability Analysis. HARRIS,D.O.; DEDHIA,D.D. Failure Analysis Associates, Inc. LU,S.C. Lawrence Livermore National Laboratory. July 1992. 352pp. 9208060118. UCRL-ID-109798. 62650:001.

This document consolidates and updates the earlier reports which provide the theoretical background as well as information needed for the execution of the computer code pc-PRAISE. pc-PRAISE is a probabilistic fracture mechanics computer code written for the IBM personal computers or their compatibles to evaluate the reliability of welds in nuclear power plant piping systems. pc-PRAISE was adapted from the PRAISE computer code which was originally developed for CDC 7600 computers in 1981 by Lawrence Livermore National Laboratory under funding from the U.S. Nuclear Regulatory Commission, and has been considerably expanded and updated over the years.

NUREG/CR-5867: GRADIENT STUDY OF A LARGE WELD JOINING TWO FORGED A 508 SHELLS OF THE MIDLAND REACTOR VESSEL. IRWIN,G.R.; ZHANG,X.J. Maryland, Univ. of, College Park, MD. * Oak Ridge National Laboratory. June 1992. 19pp. 9207270284. ORNLSUB79777810. 62508:287.

The low-carbon welds (WF67 and WF70) in the slab examined contained no abnormalities that would indicate fracture behavior different from that observed in bulk-material fracture tests. The A 508 material in the HAZ region, very close to the welds, contains small (3mm) regions adjacent to each layer of weld runs where grain coarsening and hardness elevation suggest reduction of cleavage initiation toughness. The degree of severity is largest where this local region coincides with a local elevation of carbide density in the A 508 material. The A 508 HAZ region adjacent to the topmost weld run may be the region most likely to assist cleavage-fracture initiation because of its location: close to the free surface, small cracks, and the HAZ region beneath the cladding. It was noted that the small cracks under the cladding have the appearance of prior austenite grain boundary separations that connect to austenite grain boundaries in the cladding. The extreme hardness of a narrow layer of cladding at the fusion boundary may be of interest in further studies of cladding toughness.

NUREG/CR-5868: DEVELOPMENT OF POSITION SENSITIVE PROPORTIONAL COUNTERS FOR HOT PARTICLE DETECTION IN LAUNDRY AND PORTAL MONITORS. SHONKA,J.J.; SCHWAHN,S.O.; BENNETT,T.E.; et al. Shonka Research Associates, Inc. September 1992. 38pp. 9210150137. SRA-9201. 63525:001.

This report summarizes research which demonstrates the use of position sensitive proportional counters in contamination monitoring systems. Both laundry monitoring and portal monitoring systems were deployed. The laundry monitor was deployed at a nuclear power plant where it was used to monitor clothing during an outage. Position sensitive proportional counter based contamination monitoring systems were shown to have significant advantages over systems using conventional proportional counters. These advantages include the ability to directly measure the area and quantity of contamination. This capability permits identification of hot particles. These systems are also capable of self calibration via internal check sources. Systems deployed with this technology should benefit from reduced complexity, cost, and maintenance. The inherent reduction of background that occurs when the counter is electronically divided into numerous detectors permits operation in high background radiation fields and improves detection limits over conventional technology.

NUREG/CR-5872: ORNOZL: A FINITE-ELEMENT MESH GENERATOR FOR NOZZLE-CYLINDER INTERSECTIONS CONTAINING INNER-CORNER CRACKS. KEENEY-WALKER; BASS,B.R. Oak Ridge National Laboratory. September 1992. 42pp. 9210050048. ORNL/TM-11049. 63382:164.

This report describes the ORNOZL finite-element mesh generator program for computational fracture mechanics analysis. The program automatically generates a three-dimensional (3-D) finite-element model for four different geometries of a corner crack in a nozzle-cylinder intersection. ORNOZL generates a core of special wedge or collapsed prism elements at the crack front to introduce the appropriate stress singularity at the crack tip. Regular 20-noded isoparametric brick elements are used away from the crack front in the modeling. Also, an option is included that allows for an embedded or penetrating crack in clad materials. As few as five input cards are required to execute the program. ORNOZL is part of a three-program system, ORNOZL-ADINA- ORVIRT, which addresses linear or nonlinear fracture in 2- or 3-D crack geometries. ORNOZL creates files containing nodal point coordinates and element connectivities that have formats compatible with the ADINA structural analysis program. ORVIRT is a post-processor to ADINA and employs a virtual crack extension technique to compute energy release rates at specified positions along the crack front.

NUREG/CR-5880: NONISOTHERMAL HYDROLOGIC TRANSPORT EXPERIMENTAL PLAN. RASMUSSEN,T.C.; EVANS,D.D. Arizona, Univ. of, Tucson, AZ. September 1992. 53pp. 9209240196. 63275:251.

A field heater experimental plan is presented for investigating hydrologic transport processes in unsaturated fractured rock related to the disposal of high-level radioactive waste (HLW) in an underground repository. The experimental plan provides a methodology for obtaining data required for evaluating conceptual and computer models related to HLW isolation in an environment where significant heat energy is produced. Coupled-process models are currently limited by the lack of validation data appropriate for field scales that incorporate relevant transport processes. Presented in this document is a discussion of previous nonisothermal experiments. Processes expected to dominate heat-driven liquid, vapor, gas, and solute flow during the experiment are explained, and the conceptual model for nonisothermal flow and transport in unsaturated, fractured rock is described. Of particular concern is the ability to conform the hypothesized conceptual model, specifically, the establishment of higher water saturation zones within the host rock around the heat source, and the establishment of countercurrent flow conditions within the host rock near the heat source. Field experimental plans are presented using the Apache Leap Tuff Site to illustrate the implementation of the proposed methodology. Both small-scale preliminary experiments and a long-term experiment are described.

NUREG/CR-5886: EXPERIMENTAL AND ANALYTICAL INVESTIGATION OF THE SHALLOW-FLAW EFFECT IN REACTOR PRESSURE VESSELS. THEISS,T.J.; SHUM,D.K. Oak Ridge National Laboratory. ROLFE,S.T. Kansas, Univ. of, Lawrence, KS. July 1992. 63pp. 9208060236. ORNL/TM-12115. 62654:001.

The Heavy-Section Steel Technology (HSST) Program is investigating the increase in effective fracture toughness of A 533 B steel associated with shallow flaws and the implications of the shallow-flaw effect on reactor pressure vessel (RPV) life assessments. Test data from beams indicate a significant increase in the fracture toughness of shallow- crack specimens compared with deep-crack specimens in the transition region of the toughness curve for unirradiated A 533 B steel. If the toughness increase present in the test specimens were also present in a reactor vessel, the impact on pressurized-thermal shock (PTS) analyses could be significant. To facilitate transferability of the specimen data to an RPV, posttest finite-element analyses have been performed on several test specimens and a reactor vessel for a single (PTS) transient. The analyses are sufficiently refined to allow interpretation of the results in terms of the J-integral and the so- called Q-stress parameter under plane-strain analysis assumptions. A negative Q-stress parameter is indicative of a loss of crack-tip constraint, which is associated with an increase in the fracture toughness. Analyses of the test specimens indicate that at the onset of crack initiation the deep-crack specimens exhibit an essentially zero Q- stress parameter but that the shallow-crack specimen exhibits a Q- stress parameter of about -0.7, which indicates a substantial loss of constraint in the shallow-crack beam. Using the test data and posttest analysis, a locus of toughness data in terms of the J-integral and the Q-stress parameter has been constructed for a particular temperature. Analyses were also performed on an RPV with a shallow flaw under PTS loading conditions up to the maximum value of J. At maximum J, the analyses reveal a Q-stress parameter about -0.2 to -0.4, which indicates some constraint loss but less than in the shallow-crack test specimens. Considering the RPV in terms of J-integral and Q-stress suggests there may be a larger margin of safety than would be found using the J- integral alone. Thermal-shock data, which were generated using cylindrical vessels under thermal shock loading, show no significant increase in toughness even for shallow-flaw depths. The thermal shock data seem to indicate two offsetting effects: a shallow-flaw effect, which increases toughness, and an out-of-plane (biaxial) stress effect, which decreases toughness. Additional work is necessary to resolve outstanding issues for applying shallow-crack data to an RPV and validating the J-Q technique for fracture evaluations.

NUREG/CR-5891: ACCELERATED IRRADIATION TEST OF GUNDREMMINGEN REACTOR VESSEL TREPAN MATERIAL. HAWTHORNE,J.R. Materials Engineering Associates, Inc. August 1992. 78pp. 9209240284. MEA-2468. 63276:001.

Initial mechanical properties tests of beltline material trepanned from the decommissioned KRB-A pressure vessel and archive material irradiated in the UBR test reactor revealed a major anomaly in relative radiation embrittlement sensitivity. Poor correspondence of material behavior in test vs. power reactor environments was observed for the weak test orientation (ASTM L-C) whereas correspondence was good for the strong orientation (ASTM C-L). To resolve the anomaly directly, Charpy- V specimens from a low (essentially-nil) fluence region of the vessel were irradiated together with archive material at 279 degrees C in the UBR test reactor. Properties tests before UBR irradiation revealed a significant difference in 41-J transition temperature and upper shelf energy level between the materials. However, the materials exhibited essentially the same radiation embrittlement sensitivity (both orientations), proving that the anomaly is not due to a basic difference in material irradiation resistances. Possible causes of the original anomaly and the significance to NRC Regulatory Guide 1.99 are discussed.

NUREG/CR-5896: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE ST. LUCIE UNIT 1 NUCLEAR POWER GENERATION STATION. PUGH,R.; GORE,B.F.; VO,T.V. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1992. 31pp. 9209240293. PNL-8102. 63278:237.

In a study sponsored by the U.S. Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory has developed and applied a methodology for deriving plant-specific risk-based inspection guidance for the auxiliary feedwater (AFW) system at pressurized water reactors that have not undergone probabilistic risk assessment (PRA). This methodology uses existing PRA results and plant operating experience information. Existing PRA-based inspection guidance information recently developed for the NRC for various plants was used to identify generic component failure modes. This information was then combined with plant-specific and industry-wide component information and failure data to identify failure modes and failure mechanisms for the AFW system at the selected plants. St. Lucie Unit 1 was selected as one of a series of plants for study. The product of this effort is a prioritized listing of AFW failures which have occurred at the plant and at other PWRs. This listing is intended for use by NRC inspectors in the preparation of inspection plans addressing AFW risk-important components at the St. Lucie Unit 1 plant.

NUREG/CR-5905: REVIEW AND DEVELOPMENT OF COMMON NOMENCLATURE FOR NAMING AND LABELING SCHEMES FOR PROBABILISTIC RISK ASSESSMENT. TRUSTY,A.D.; MACKOWIAK,D.P. EG&G Idaho, Inc. August 1992. 65pp. 9209280052. EGG-2615. 63335:216.

This report describes the review and development of common nomenclature for naming and labeling schemes for probabilistic risk assessments (PRAs) conducted by the Idaho National Engineering Laboratory (INEL). Based on the review, the INEL recommends using an existing basic event labeling scheme and existing naming schemes for systems, component types, and component failure modes. The review showed no adequate accident sequence labeling schemes currently exist. Therefore, the INEL developed a scheme that would meet the review requirements of not exceeding 16 characters and being highly descriptive of the accident sequence involved. As parts of the developed accident sequence labeling scheme, the INEL also developed transient and loss- of-coolant accident initiating event codes, a sequence naming scheme, and accident type codes. Applications of the accident sequence labeling scheme are presented along with tables to allow changes from other schemes to the recommended naming schemes. The review and development were conducted to provide the Nuclear Regulatory Commission with the means to coordinate and integrate their internal activities through a common nomenclature for their many data bases.

NUREG/CR-5910: LOSS OF ESSENTIAL SERVICE WATER IN LWRS (GI-153).Scoping Study. CRAMOND,W.R.; MITCHELL,D.B. Sandia National Laboratories. YAKLE,J.L.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). August 1992. 365pp. 9209240367. SAND92-1084. 63276:079.

The contribution of essential service water (ESW) system failure to core damage frequency has long been a concern of the NRC. The objective of this study is to assess the safety significance of the loss of ESW systems in LWRs relative to core damage frequency (CDF) and perform a limited value/impact analysis of potential modifications to solve ESW vulnerabilities using a prototypical (pilot) plant. Previous studies indicate that service water systems contribute from <1% to 65% of the total internal CDF. For the pilot plant analyzed, common ESW vulnerabilities are failure of standby service water pumps to start, backflow through check valves for cross-tied pumps, and failure of normally closed isolation valves in diesel generator cooling loops to open on demand. For the potential modifications evaluated for the pilot plant, the results showed that they could reduce the CDF by as much as 33 percent. However, the dollars per person REM measures resulting from various groups of these modifications significantly exceeded the current criteria of $1,000. The results, since they only apply to the pilot plant, are not typical of all BWRs. Due to the importance of service water to CDF and the plant specific nature of ESW systems there could be plants for which there would be cost effective modifications. Additional analysis would be required to identify them.

NUREG/CR-5930: HIGH INTEGRITY SOFTWARE STANDARDS AND GUIDELINES. WALLACE,D.R.; IPPOLITO,L.M.; KUHN,D.R. National Institute of Standards & Technology (formerly National Bureau of Standa. September 1992. 108pp. 9210050044. NIST SP 500-204. 63382:057.

This report presents results of a study of standards, draft standards, and guidelines (all of which will hereafter be referred to as documents) that provide requirements for the assurance of software in safety systems in nuclear power plants. The study focused on identifying the attributes necessary in a standard for providing reasonable assurance for software in nuclear systems. The study addressed some issues involved in demonstrating conformance to a standard. The documents vary widely in their requirements and the precision with which the requirements are expressed. Recommendations are provided for guidance addressing the assurance of high integrity software. It is recommended that a nuclear industry standard be developed based on the documents reviewed in this study with additional attention to the concerns identified in this report.

NUREG/CR-6001: AGING ASSESSMENT OF BWR STANDBY LIQUID CONTROL SYSTEMS. BUCKLEY,G.D.; ORTON,R.D.; JOHNSON,A.B.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1992. 41pp. 9208250215. PNL-8020. 63037:217.

Pacific Northwest Laboratory conducted a Phase I aging assessment of the standby liquid control (SLC) system used in boiling-water reactors. The study was based on detailed reviews of SLC system component and operating experience information obtained from the Nuclear Plant Reliability Database System, the Nuclear Document System, Licensee Event Reports, and other databases. Sources on sodium pentaborate, borates, and boric acid, as well as the effects of environment and corrosion in the SLC system were also reviewed to characterize chemical properties and corrosion characteristics of borated solutions. Relatively few SLC component failures were attributed to sodium pentaborate buildup or corrosion. The leading aging degradation concern to date appears to be setpoint drift in relief valves, which has been discovered during routine surveillance and is thought to be caused by mechanical wear. A higher setpoint results in loss of system over- pressure protection, and a decrease in setpoint results in a reduction of boron injection rate. Degradation was also observed in pump seals and internal valves, which could prevent the pumps from operating as required by the technical specifications. In general, however, the results of the Phase I study indicate that age-related degradation of SLC systems has not been serious.

NUREG/CR-6003: DENSITY-WAVE INSTABILITIES IN BOILING WATER REACTORS. MARCH-LEUBA,J. Oak Ridge National Laboratory. September 1992. 54pp. 9210050038. ORNL/TM-12130. 63382:001.

This report contains a review of issues related to density-wave instabilities in boiling water reactors (BWRs). The report describes the types of instability modes that can be expected in operating reactors. These modes are: (1) the channel thermo-hydraulic instability mode; (2) the core-wide instability mode; and (3) the out- of-phase instability mode. The physical mechanisms leading to each type of instability are reviewed and documented, along with some common mathematical models used in stability calculations. The main approximations used in these mathematical models are presented, and their impact on the accuracy of the calculations is reviewed. The linear behavior of a BWR is studied through the use of transfer functions, and the nonlinear behavior and limit cycle development are studied. A summary of the sensitivities to physical parameters is also included in this report.

NUREG/CR-6008: CONSTRAINT EFFECTS ON FRACTURE TOUGHNESS FOR CIRCUMFERENTIALLY ORIENTED CRACKS IN REACTOR PRESSURE VESSELS. BASS,B.R.; SHUM,D.K.; KEENEY-WALKER Oak Ridge National Laboratory. August 1992. 151pp. 9209240303. ORNL/TM-12131. 63272:187.

Pressurized-thermal-shock (PTS) loading produces biaxial stress fields in a reactor pressure vessel (RPV) wall with one of the principal stresses aligned parallel to postulated surface cracks in either longitudinal or circumferential welds. The limited quantity of existing biaxial test data suggest a significant decrease of fracture toughness under out-of-plane (i.e., parallel to the crack front) biaxial loading conditions when compared with toughness values obtained under uniaxial conditions. Any increase in crack-tip constraint resulting from these out-of-plane biaxial stresses would act in opposition to the in-plane constraint relaxation that has been previously demonstrated for shallow cracks. Consequently, understanding of both in-plane and out-of-plane crack-tip constraint effects is necessary to a refined analysis of fracture initiation from shallow cracks under PTS transient loading. This report is the second in a series investigating the potential impact of far-field out-of-plane stresses and strains on fracture initiation toughness. Selected fracture prediction models, previously validated for small-scale fracture specimens under nearly plane strain conditions, were applied to additional large-scale data with the objective of validating models in the plane stress-to-plane strain domain before applying them to positive out-of-plane strain conditions. The general finding was that applications of the models resulted in predictions of fracture behavior that conflicted with existing experimental data considered relevant to the problem. Because of the conflicting results, it is apparent that testing of RPV steels is required: (1) to determine the magnitude of out-of-plane biaxial loading effects on fracture toughness; and (2) to provide a basis for development of predictive models. This course of action is necessary to support a refined treatment of in-plane and out-of-plane constraint effects in PTS analysis. Proposed in this report are criteria for a biaxial specimen that would form the basis of a testing program designed to provide data to explain differences between theoretical predictions and measured material behavior. Results of design studies on the biaxial specimen will be presented in a future report from the Heavy-Section Steel Technology Program.

NUREG/CR-6009 V01: DEVELOPING AND ASSESSING ACCIDENT MANAGEMENT PLANS FOR NUCLEAR POWER PLANTS.Development Process And Criteria. HANSON,D.J.; BLACKMAN,H.S.; MEYER,O.R.; et al. EG&G Idaho, Inc. August 1992. 78pp. 9209240241. EGG-2682. 63273:254.

This document is the first volume of a two-volume report. It describes a four-phase approach for developing criteria that can be used for assessing the adequacy of severe accident management plans for nuclear power plants. The general attributes of accident management plans (Phase 1) are identified, and a process for developing and implementing severe accident management plans (Phase 2) is described. This process is based on a prototype process described in NUREG/CR-5543. The prototype process was revised using results from an evaluation of this process (Phase 3), which is documented in Volume 2. General criteria for assessing the adequacy of accident management plans are also presented (Phase 4). These criteria were based on process specific criteria presented in Volume 2 and NUREG/CR-5543.

NUREG/CR-6009 V02: DEVELOPING AND ASSESSING ACCIDENT MANAGEMENT PLANS FOR NUCLEAR POWER PLANTS.Evaluation Of A Prototype Process. HANSON,D.J.; JOHNSON,S.P.; BLACKMAN,H.S.; et al. EG&G Idaho, Inc. July 1992. 182pp. 9207270263. EGG-2682. 62509:178.

This document is the second of a two-volume report that discusses development of accident management plans for nuclear power plants. The first volume: (a) describes a four-phase approach for developing criteria that could be used for assessing the adequacy of accident management plans; (b) identifies the general attributes of accident management plans (Phase 1); (c) presents a prototype process for developing and implementing severe accident management plans (Phase 2); and (d) presents criteria that can be used to assess the adequacy of accident management plans. This volume: (a) describes results from an evaluation of the capabilities of the prototype process to produce an accident management plan (Phase 3), and (b), based on these results and preliminary criteria included in NUREG/CR-5543, presents modifications to the criteria where appropriate.

NUREG/CR-6010: HISTORY AND CURRENT STATUS OF GENERATION 3 THERMAL SLEEVES IN WESTINGHOUSE NUCLEAR POWER PLANTS. MARTIN,G. MARTIN Consulting Services, Inc. * S. Cohen & Associates, Inc. July 1992. 89pp. 9208250214. 62871:001.

From mid-1982 until 1987, loose thermal sleeves or sleeves with cracked attachment welds were found in several operating Westinghouse nuclear power plants. Westinghouse investigations concluded that these occurrences had been confined to those thermal sleeves of "Generation 3" design. The sleeve problem was a generic issue (Generic Issue No. 73) but affected only those plants using Generation 3 sleeves. The NRC's Safety Evaluation Report, "Evaluation of Thermal Sleeve Problems in Westinghouse Plants", dated October 28, 1983, contains a proposed resolution for the thermal sleeve issue. Because a resolution had been proposed, the issue was considered to be nearly resolved. This report presents updated and supplemental information on the thermal sleeve problem, which affected Westinghouse nuclear plants using Generation 3 thermal sleeves. The information presented is intended to be the basis for deciding whether the issue can be considered resolved or if additional information is required to resolve it.

NUREG/IA-0042: DISPERSED FLOW FILM BOILING.An Investigation Of The Possibility To Improve The Models Implemented In The NRC Computer Codes For The Reflooding Phase Of The LOCA. ANDREANI,M. Paul Scherrer Institute. ANDREANI,M.; YADIGAROGLU,G.; et al. Swiss Federal Institute of Technology (ETH). August 1992. 68pp. 9209220485. 63261:206.

Dispersed Flow Film Boiling is the heat transfer regime that occurs at high void fractions in a heated channel. The way this heat transfer mode is modelled in the NRC computer codes (RELAP5 and TRAC) and the validity of the assumptions and empirical correlations used is discussed. An extensive review of the theoretical and experimental work related with heat transfer to highly dispersed mixtures reveals the basic deficiencies of these models: the investigation refers mostly to the typical conditions of low rate bottom reflooding, since the simulation of this physical situation by the computer codes has often showed poor results. The alternative models that are available in the literature are reviewed, and their merits and limits are highlighted. The modifications that could improve the physics of the models implemented in the codes are identified.

NUREG/IA-0054: ASSESSMENT OF RELAP5/MOD2, CYCLE 36.02, USING NEPTUN REFLOODING EXPERIMENTAL DATA. RICHNER,M.; ANALYTIS,G.T.; AKSAN,S.N. Paul Scherrer Institute. August 1992. 105pp. 9209240217. 63271:211.

Seven NEPTUN reflooding experiments with varying parameters flooding rate, single rod power, pressure and initial rod temperatures were simulated with the code RELAP5/MOD2, version 36.02, to assess the code, especially its reflood model. These calculations were performed with the specific objectives of evaluating the general prediction capability as well as specific problem areas of the RELAP5/MOD2 code in modelling boil-off and reflood behavior. The differences between code predictions and experiments are described and analyzed. Implementing new correlations into the code and modifying or correcting existing correlations, for example for wall heat transfer or interphase friction, some of the weak points of the code during reflooding could be identified.

NUREG/IA-0067: RECIRCULATION SUCTION LARGE BREAK LOCA ANALYSIS OF THE SANTA MARIA DE GARONA NUCLEAR POWER PLANT USING TRAC-BF1(G1J1). LOPEZ,J.V. Polytechnic Univ. of Madrid, Madrid, Spain. CRESPO,J.L. Cantabria, Univ. of, Spain. FERNANDEZ,R.A. Nuclenor, S.A. (Spain). August 1992. 63pp. 9208250167. ICSP-GA-LOCA-T. 62872:223.

A best estimate analysis of a recirculation suction pipe large break loss-of-coolant accident analysis for Santa Maria De Garona nuclear power plant using TRAC-BF1 code is presented.

NUREG/IA-0088: POST-TEST-ANALYSIS AND NODALIZATION STUDIES OF OECD LOFT EXPERIMENT LP-02-6 WITH RELAP5/MOD2 CY36-02. LUBBESMEYER,D. Paul Scherrer Institute. August 1992. 186pp. 9209240233. PSI-BERICHTNR92. 63272:001.

This report presents the results and analysis of nine post-test calculations of the Experiment LP-02-6 by using RELAP5/MOD2 CY36-02 computer code with different nodalizations. Starting with a "standard nodalization" we have reduced the number of volumes and junctions as well as the number of radial zones in the fuel rods, for different nodalization studies. Except for the cladding temperatures, only small discrepancies have been observed for the other main parameters of the results of runs using different nodalizations but reduced number of volumes and junctions usually have lead to a decreased running time for the problem. The time behaviors of the cladding temperatures have been significantly affected by the chosen nodalizations. The most comparable results with the experimental data have been achieved by using medium number of nodes. With respect to high mass-flux, early bottom-up rewetting, one of the key-events of Experiment LP-02-6 as well as of most of the other LOFT large break experiments, RELAP5/MOD2 was not able to predict this phenomenon except with a certain manipulation by initiating the reflood option.

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This index was developed from keywords and word strings in titles and abstracts. During this development period, there will be some redundancy, which will be removed later when a reasonable thesaurus has been developed through experience. Suggestions for improvements are welcome.

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EDO - OFFICE OF THE CONTROLLER (PRE 820418 & POST 890205)
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DIVISION OF SAFETY PROGRAMS (POST 870413)
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EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
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EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA

DIVISION OF SAFETY PROGRAMS (POST 870413)

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

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EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS

DIVISION OF HIGH-LEVEL WASTE MANAGEMENT (POST 870413)

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NUREG/CR-5700: AGING ASSESSMENT OF REACTOR INSTRUMENTATION AND PROTECTION SYSTEM COMPONENTS.Aging-Related Operating Experiences.

NUREG/CR-5772 V01: AGING, CONDITION MONITORING, AND LOSS-OF-COOLANT ACCIDENT (LOCA) TESTS OF CLASS 1E ELECTRICAL CABLES.Crosslinked Polyolefin Cables.

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NUREG/CR-5793: A COMPARISON OF ANALYSIS METHODOLOGIES FOR PREDICTING CLEAVAGE ARREST OF A DEEP CRACK IN A REACTOR PRESSURE VESSEL SUBJECTED TO PRESSURIZED-THERMAL-SHOCK LOADING CONDITIONS.

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