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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:

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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).

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

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence (AO) as an unscheduled incident or event that the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report provides a description of those events that have been determined to be abnormal occurrences during the period of January 1 through March 31, 1994. This report addresses seven AOs at NRC-licensed facilities. One involved inoperable main steam isolation valves at a boiling water reactor, four involved medical brachytherapy misadministrations, one involved a medical teletherapy misadministration, and one involved four lost reference sources. One AO that was reported by an Agreement State is also discussed; the information is current as of April 25, 1994. This event involved a therapeutic radiopharmaceutical misadministration. The report also contains updates on seven abnormal occurrences previously reported by NRC licensees and one abnormal occurrence previously reported by an Agreement State licensee. For the period January 1 to March 31, 1994, no new "Other Events of Interest" were reported but an update to a therapeutic misadministration previously reported as an "Other Event of Interest" is included.

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

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

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

See NUREG-0304,V19,N01 abstract.

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

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

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

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

The staff of the Nuclear Regulatory Commission is performing nuclear power plant design certification reviews based on a design process plan that describes the human factors engineering (HFE) program elements that are necessary and sufficient to develop an acceptable detailed design specification and an acceptable implemented design. There are two principal reasons for this approach. First, the initial design certification applications submitted for staff review did not include detailed design information. Second, since human performance literature and industry experiences have shown that many significant human factors issues arise early in the design process, review of the design process activities and results is important to the evaluation of an overall design. However, current regulations and guidance documents do not address the criteria for design process review. Therefore, the HFE Program Review Model (HFE PRM) was developed as a basis for performing design certification reviews that include design process evaluations as well as review of the final design. A central tenet of the HFE PRM is that the HFE aspects of the plant should be developed, designed, and evaluated on the basis of a structured top-down system analysis using accepted HFE principles. The HFE PRM consists of ten component elements. Each element is divided into four sections: Background, Objective, Applicant Submittals, and Review Criteria. This report describes the development of the HFE PRM and gives a detailed description of each HFE review element.

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

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

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

NUREG-0750 V39 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January -March 1994. * Division of Freedom of Information & Publications Services (Post 940714). July 1994. 41pp. 9408150162. 80537:291.
Digests and indexes for issuances of the Commission, the Atomic Safety and Licensing Board Panel, the Administrative Law Judges, the Directors' Decisions, and the Denials of Petitions for Rulemaking are presented.

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

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

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

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

NUREG-0837 V14 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. January-March 1994. STRUCKMEYER,R. Region 1 (Post 820201). June 1994. 231pp. 9407250293. 80329:001.
This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facilities throughout the country for the first quarter of 1994.

NUREG-0837 V14 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. April-June 1994. STRUCKMEYER,R. Region 1 (Post 820201). August 1994. 231pp. 9409260062. 81042:112.
This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facilities throughout the country for the second quarter of 1994.

NUREG-0936 V13 N01: NRC REGULATORY AGENDA.Quarterly Report,January-March 1994. * Division of Freedom of Information & Publications Services (890206-940714). June 1994. 58pp. 9407250298. 80328:235.
The NRC Regulatory Agenda is a compilation of all rules on which the NRC has recently completed action, or has proposed action, or is considering action, and all petitions for rulemaking which have been received by the Commission and are pending disposition by the Commission. The Regulatory Agenda is updated and issued each quarter.

NUREG-0936 V13 N02: NRC REGULATORY AGENDA.Quarterly Report,April-June 1994. * Division of Freedom of Information & Publications Services (Post 940714). August 1994. 68pp. 9410060320. 81223:167.
See NUREG-0936,V13,N01 abstract.

NUREG-0940 V13N01P01: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED REACTOR LICENSEES.Quarterly Progress Report,January-March 1994. * Ofc of Enforcement (Post 870413). June 1994. 210pp. 9407260127. 80345:099.
This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January - March 1994) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to reactor licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

NUREG-0940 V13N01P02: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MEDICAL LICENSEES.Quarterly Progress Report,January-March 1994. * Ofc of Enforcement (Post 870413). June 1994. 145pp. 9407260176. 80346:001.
See NUREG-0940,V13,N01,P01 abstract.

NUREG-0940 V13N01P03: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED INDUSTRIAL LICENSEES.Quarterly Progress Report,January-March 1994. * Ofc of Enforcement (Post 870413). June 1994. 118pp. 9407260183. 80346:146.
See NUREG-0940,V13,N01,P01 abstract.

NUREG-0940 V13N02P01: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED REACTOR LICENSEES.Quarterly Progress Report,April-June 1994. * Ofc of Enforcement (Post 870413). August 1994. 200pp. 9409270322. 81046:108.
This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1994) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to reactor licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

NUREG-0940 V13N02P02: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MEDICAL LICENSEES.Quarterly Progress Report,April-June 1994. * Ofc of Enforcement (Post 870413). August 1994. 354pp. 9410130039. 81263:001.
See NUREG-0940,V13,N02,P01 abstract.

NUREG-0940 V13N02P03: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED INDUSTRIAL LICENSEES.Quarterly Progress Report,April-June 1994. * Ofc of Enforcement (Post 870413). August 1994. 177pp. 9410070244. 81232:001.
See NUREG-0940,V13,N02,P01 abstract.

NUREG-1145 V10: U.S. NUCLEAR REGULATORY COMMISSION 1993 ANNUAL REPORT. * Office of Administration, Director (Post 940714). September 1994. 306pp. 9410130036. 81262:001.
This report covers the major activities, events, decisions, and planning that took place during Fiscal Year 1993 within the U.S. Nuclear Regulatory Commission (NRC) or involving the NRC.

NUREG-1242 V03 PT01: NRC REVIEW OF ELECTRIC POWER RESEARCH INSTITUTE'S ADVANCED LIGHT WATER REACTOR UTILITY REQUIREMENTS DOCUMENTS.Passive Plant Designs.Chapter 1.Project Number 669. * Associate Director for Advanced Reactors & License Renewal (Post 910918). August 1994. 450pp. 9410060313. 81222:001.
The staff of the U.S. Nuclear Regulatory Commission has prepared Volume 3 (Parts 1 and 2) of a safety evaluation report (SER), "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements - Passive Plant

Designs," to document the results of its review of the Electric Power Research Institute's "Advanced Light Water Reactor Utility Requirements Document." This SER gives the results of the staff's review of Volume III of the Requirements Document for passive plant designs, which consists of 13 chapters and contains utility design requirements for nuclear power plants for which passive features will be used in their design (approximately 600 megawatts-electric).

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

See NUREG-1242,V03,P01 abstract.

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

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

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

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

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

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

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

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

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

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

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

See NUREG-1462,V01 abstract.

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

The Chief Financial Officers Act of 1990 requires the NRC Chief Financial Officer to prepare and submit an annual financial statement to the Director of the Office of Management and Budget (OMB). The OMB has replaced the requirement for the CFO's Annual Report with the annual financial statement. The annual financial statement was previously included in the Chief Financial Officer's Annual Report. This report is the third annual report for the NRC and includes an overview of the NRC, the audited principal financial statements and audit reports for fiscal year 1993, and supplemental financial and management information.

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

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

NUREG-1484 V01: FINAL ENVIRONMENTAL IMPACT STATEMENT FOR THE CONSTRUCTION AND OPERATION OF CLAIBORNE ENRICHMENT CENTER, HOMER, LOUISIANA.Docket No. 70-3070.Louisiana Energy Services, L.P. Environmental Impact Statement. ZEITOUN,A. Science Applications International Corp. (formerly Science Applications, Inc.). August 1994. 390pp. 9409070096. 80790:001.

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

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

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

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

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

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

See NUREG-1496,V01,DRF FC abstract.

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

The Nuclear Regulatory Commission's (NRC) regulations in 10 CFR Part 20 are being amended to include radiological criteria for decommissioning of lands and structures at nuclear facilities. 10 CFR Part 20, Subpart E establishes criteria for the remediation of contaminated sites or facilities that will allow their release for future use with or with-out restrictions. The criteria include a Total Effective Dose Equivalent (TEDE) limit of 15 mrem/year (0.15 mSv/y) that should not be exceeded by an average individual among those who could potentially receive the greatest exposure from any residual activity within a facility or on a site. The criteria also require a licensee to reduce any residual radioactivity to as-low- as-reasonably-achievable (ALARA) levels. This staff draft guide describes acceptable procedures for determining the predicted dose level (PDL) from any residual radioactivity at the site. It describes the basic features of the calculational models and the associated default assumptions and parameter values the NRC staff would find acceptable in calculating PDLs. Appendices A, B, and C provide numerical values that can be used to estimate the dose from residual radioactivity remaining at a site. Since 10 CFR Part 20, Subpart E introduces several new concepts, definitions and discussions are included in a regulatory position concepts section of the guide to assist licensees in understanding some of the philosophy underlying the rule.

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

This report was originally published as an appendix to the draft U.S. Nuclear Regulatory Commission (NRC) document entitled, "Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for Decommissioning of NRC-Licensed Nuclear Facilities." Because of the great interest in this report by members of the public, citizen and environmental organizations, academicians, licensees, and regulators, the NRC staff is publishing this report separately, so that it can be readily available to a diverse audience. This report was created to assist both the NRC staff and interested members of the public in evaluating background radiation (background) as a decommissioning criterion, by serving as a primer on background and providing information on the existing applications of background in regulatory criteria and standards. This report also discusses some of the methods available to measure and distinguish between the very low radiation levels associated with background and man-made sources of radiation. Two approaches are considered for applying background as a decommissioning criterion; these are the use of background dose rates and background radionuclide concentrations. This report concludes that the temporal and spatial variability of background produces a wide range of doses to United States residents, which prevents the application of background dose rates as a decommissioning criterion. Instead, this report recommends that local background radionuclide concentrations serve as a benchmark for decommissioning criteria, while taking into account the concept of reducing residual radioactivity to a level as low as is reasonably achievable.

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

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

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

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

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

See NUREG-1503,V01 abstract.

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

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

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

The 1994 Symposium on Valve and Pump Testing, jointly sponsored by the Board of Nuclear Codes and Standards of the American Society of Mechanical Engineers and by the Nuclear Regulatory Commission, provides a forum for the discussion of current programs and methods for inservice testing and motor-operated valve testing at nuclear power plants. The symposium also provides an opportunity to discuss the need to improve that testing in order to help ensure the reliable performance of pumps and valves. The participation of industry representatives, regulators, and consultants results in the discussion of a broad spectrum of ideas and perspectives regarding the improvement of inservice testing of pumps and valves at nuclear power plants.

NUREG/CP-0137 V02: PROCEEDINGS OF THE THIRD NRC/ ASME SYMPOSIUM ON VALVE AND PUMP TESTING.Held At The Hyatt Regency Hotel, Washington,DC. July 18--21, 1994.Session 3A -Session 4B. * EG&G Idaho, Inc. July 1994. 387pp. 9408030114. EGG-2742. 80424:197.

See NUREG/CP-0137,V01 abstract.

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

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

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

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

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

This report presents a revision of the procedure and correlations presented earlier in NUREG/CR-4513, ANL-90/42 (June 1991) for predicting the change in mechanical properties of cast stainless steel components due to thermal aging during service in light water reactors at 280-330 degrees C (535-625 degrees F). The correlations presented in this report are based on an expanded database and have been optimized with mechanical-property data on cast stainless steels aged up to ≈58,000 h at 290-350 degrees C (554-633 degrees F). The correlations for estimating the change in tensile stress, including the Ramberg/Osgood parameters for strain hardening, are also described. The fracture toughness J-R curve, tensile stress, and Charpy-impact energy of aged cast stainless steels are estimated from known material information. Mechanical properties of a specific cast stainless steel are estimated from the extent and kinetics of thermal embrittlement. Embrittlement of cast stainless steels is characterized in terms of room-temperature Charpy-impact energy. The extent or degree of thermal embrittlement at "saturation," i.e., the minimum impact energy that can be achieved for a material after long- term aging, is determined from the chemical composition of the steel. Charpy-impact energy as a function of time and temperature of reactor service is estimated from the kinetics of thermal embrittlement, which are also determined from the chemical composition. The initial impact energy of the unaged steel is required for these estimations.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

NUREG/CR-5861: CRACK-SPEED RELATIONS INFERRED FROM LARGE SINGLE-EDGE-NOTCHED SPECIMENS OF A 533 B STEEL. SCHWARTZ,C.W. Oak Ridge National Laboratory. July 1994. 32pp. 9408180237. ORNLSUB79-77789. 80595:312.

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

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

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

NUREG/CR-5908 V02: ADVANCED HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE. Evaluation Procedures And Guidelines For Human Factors Engineering Reviews. O'HARA,J.M.; BROWN,W.S. Brookhaven National Laboratory. BAKER,C.C.; et al. Affiliation Not Assigned. July 1994. 277pp. 9408250004. BNL-NUREG-52333. 80656:001.

See NUREG/CR-5908,V01 abstract.

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

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

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

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

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

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

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

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

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

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several microcomputer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. The Integrated Reliability and Risk Analysis System (IRRAS) is a state-of-the-art, microcomputer-based probabilistic risk assessment (PRA) model development and analysis tool to address key nuclear plant safety issues. IRRAS is an integrated software tool that gives the user the ability to create and analyze fault trees and accident sequences using a microcomputer. This program provides functions that range from graphical fault tree construction to cut set generation and quantification to report generation. Version 5.0 of IRRAS provides the same capabilities as earlier versions and adds the ability to perform location transformations, seismic analysis, and provides enhancements to the user interface as well as improved algorithm performance. Additionally, version 5.0 contains new alphanumeric fault tree and event tree editors, and a powerful set of macro-based rule editors. These editors are used for event tree rules, recovery rules, and end state partitioning.

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

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several microcomputer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. This volume is the tutorial manual for the Integrated Reliability and Risk Analysis System (IRRAS) Version 5.0, a state-of-the-art, microcomputer-based probabilistic risk assessment (PRA) model development and analysis tool to address key nuclear plant safety issues. IRRAS is an integrated software tool that gives the user the ability to create and analyze fault trees and accident sequences using a microcomputer. A series of lessons is provided that guides the user through basic steps common to most analyses performed with IRRAS. The tutorial is divided into two major sections: basic and additional features. The basic section contains lessons that lead the student through development of a very simple problem in IRRAS, highlighting the program's most basic features. The additional features section contains lessons that expand on basic analysis features of IRRAS 5.0.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

To satisfy the need for verification of the computer programs and modeling techniques that will be used to perform the final piping analysis for the ABB/ Combustion Engineering System 80+ Standardized Plant, three benchmark problems were developed. The problems are representative piping systems subjected to representative dynamic loads with solutions developed using the methods being proposed for analysis for the System 80+ standard design. It will be required that the combined license licensees demonstrate that their solutions to these problems are in agreement with the benchmark problem set.

NUREG/CR-6143 V02P1A: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUT-

DOWN OPERATIONS AT GRAND GULF,UNIT 1.Analysis Of Core Damage Frequency From Internal Events For Plant Operational State 5 During A Refueling Outage.Sections 1-9. WHITEHEAD,D. Sandia National Laboratories. DARBY,J. Science & Engineering Associates, Inc. YAKLE,J.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). June 1994. 284pp. 9408030161. SAND93-2440. 80442:001.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects being performed by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL). Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied. The objectives of the program are to assess the risks of severe accidents initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The objective of this report is to document the approach utilized in the Grand Gulf plant and discuss the results obtained. A parallel report for the Surry plant is prepared by SNL.

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

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

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

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

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

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

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

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

NUREG/CR-6144 V02P3B: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY,UNIT 1.Analysis Of Core Damage Frequency From Internal Events During Mid-Loop Operations.Appendices E (Sections E.9-E.16). CHU,T.L.; MUSICKI,Z.; KOHUT,P.; et al. Brookhaven National Laboratory. June 1994. 502pp. 9408150234. BNL-NUREG-52399. 80556:202.

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

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

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

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

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

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

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

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

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

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

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

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

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects being performed by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL). Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied. The objectives of the program are to assess the risks of severe accidents initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The objective of this report is to document the approach utilized in the Surry plant and discuss the results obtained. A parallel report for the Grand Gulf plant is prepared by SNL.

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

Goals of the Nondestructive Evaluation Reliability Program sponsored by the NRC at PNL are to 1) assess current inspection techniques and requirements of all pressure boundary systems and components, 2) determine if improvements are needed, and 3) if necessary, develop recommendations for revising the applicable ASME Codes and regulatory requirements. In evaluating approaches that could be used to provide a technical basis for improved inservice inspection (ISI) plans, PNL has developed and applied a method that uses results of probabilistic risk assessment (PRA) to rank pressure boundary systems for ISI. In the PNL program, the feasibility of developing risk-based generic ISI requirements is being conducted in two phases. Phase I involves identifying and prioritizing the systems most relevant to plant safety. The results of these evaluations will be consolidated into requirements for comprehensive piping system inspections that will be developed in Phase II. This report presents the Phase I results of evaluations for eight selected plant PRAs. Based on the results of this study, it appears that there are generic insights that can be extrapolated from the selected plants to specific classes of light water reactors.

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

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

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

Previous work by the authors described a micromechanis fracture model to correct measured J(c)-values for the mechanistic effects of large-scale yielding. This new work extends the model to also include the influence of ductile crack extension prior to cleavage. Ductile crack extensions of 10-15 X the initial crack tip opening displacement at initiation are considered in plane-strain, finite element computations. The finite element results demonstrate a significant elevation in crack-tip constraint due to macroscopic "sharpening" of the extending tip relative to the blunt tip at the initiation of growth. However this effect is offset partially by the additional plastic deformation associated with the increased applied J required to grow the crack. The initial a/W ratio, tearing modulus, strain hardening exponent and specimen size interact in a complex manner to define the evolving near-tip conditions for cleavage fracture. The paper explores development of the new model, provides necessary graphs and procedures for its application and demonstrates the effects of the model on fracture data sets for two pressure vessel steels (A533B and A515).

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

The results of Direct Containment Heating (DCH) integral experiments are presented. The experiments simulated a high pressure melt ejection in the Zion Nuclear Power Plant. Experiments were conducted in a 1/40 scale model of the Zion containment. The model included the vessel lower head, cavity and instrument tunnel, and the lower containment structures. The melt ejections were driven by steam. There were two main objectives of these experiments. The first was to investigate the effect of scale on DCH phenomena. The IET test series addressed this by conducting counterpart integral tests in a 1/40 scale facility at Argonne National Laboratory and in a 1/10 scale facility at Sandia National Laboratories. Iron/alumina thermite with chromium was used as a core melt simulant in the IET test series. The second objective was to address potential experiment distortions introduced by the use of non-prototypic iron/alumina thermite. The second objective was met in the U series of tests which utilized a prototypic core melt. Corium experiments were conducted that were counterpart to the IET-1RR and IET-6 iron/alumina tests.

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

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

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

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

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

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

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

This manual provides for determining the minimum safe standoff distance between vital safety related equipment and the design basis vehicle bomb threat adopted by the U.S. Nuclear Regulatory Commission. Vital safety related equipment should survive the design basis vehicle bomb attack when the minimum safe standoff distance is provided. Guidance is provided for exposed vital safety related equipment and for equipment housed within vital area barriers. The types of vital area barriers addressed are 12-, 18-, 24-, and 30 inch-thick reinforced concrete slabs with reinforcing ratios of 0.2, 0.4, 0.6, 0.8, 1.0 percent.

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

See NUREG/CR-6190,V01 abstract.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

This report presents the results of the pipe fracture experiments, analyses, and material characterization efforts performed within Subtask 1.1 of the IPIRG Program. The objective of Subtask 1.1 was to experimentally verify the analysis methodologies for circumferentially cracked pipe subjected primarily to inertial stresses. Eight cracked-pipe experiments were conducted on 6-inch nominal diameter TP3O4 and A1O6B pipe. The experimental procedure was developed using nonlinear time- history finite element analyses which included the nonlinear behavior due to the crack. The model did an excellent job of predicting the displacements, forces, and times to maximum moment. The comparison of the experimental loads to the predicted loads by the Net-Section-Collapse (NSC), Dimensionless Plastic-Zone Parameter, J-estimation schemes, R6, and ASME Section XI inservice flaw assessment criteria tended to underpredict the measured bending moments except for the NSC analysis of the A1O6B pipe. The effects of flaw geometry and loading history on toughness were evaluated by calculating the toughness from the pipe tests and comparing these results to C(T) values. These effects were found to be variable. The surface-crack geometry tended to increase the toughness (relative to C(T) results), whereas a negative load-ratio significantly decreased the TP304 stainless steel surface-cracked pipe apparent toughness. The inertial experiments tended to achieve complete failure within a few cycles after reaching maximum load in these relatively small diameter pipe experiments. Hence, a load-controlled fracture mechanics analysis may be more appropriate than a displacement-controlled analysis for these tests.

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

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

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

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

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

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

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

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

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

In response to public concern about the cleanup of the Three Mile Island, Unit 2 (TMI-2) facility after an accident on March 28, 1979 involving a loss of reactor coolant and subsequent damage to the reactor fuel, twelve citizens were asked to serve on an independent Advisory Panel to consult with the Nuclear Regulatory Commission (NRC) on the decontamination and cleanup of the facility. The panel met 78 times over a period of thirteen years, holding public meetings in the vicinity of TMI-2 and meeting regularly with NRC Commissioners in Washington, D.C. This report describes the results of a project designed to identify and describe the lessons learned from the Advisory Panel and place those lessons in the context of what we generally know about citizen advisory groups. A summary of the empirical literature on citizen advisory panels is followed by a brief history of the TMI-2 Advisory Panel. The body of the report contains the analysis of the lessons learned, preliminary conclusions about the effectiveness of the Panel, and implications for the NRC in the use of advisory panels. Data for the report include meeting transcripts and interviews with past and present Panel participants.

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

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

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

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

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

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

Secondary Report Number Index

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

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Personal Author Index

This index lists the personal authors of NRC staff, contractor, and international agreement reports in alphabetical order. Each name is followed by the NUREG number and the title of the report(s) prepared by the author. If further information is needed, refer to the main citation by the NUREG number.

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Subject Index

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

NRC Originating Organization Index (Staff Reports)

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

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NRC Originating Organization Index (International Agreements)

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

There were no NUREG/IA reports published during this quarter.

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NRC Contract Sponsor Index (Contractor Reports)

This index lists the NRC organizations that sponsored the contractor reports listed in this compilation. It is arranged alphabetically by major NRC organization (e.g., program office) and then by subsections of these (e.g., divisions) where appropriate. The sponsor organization is followed by the NUREG/CR number and title of the report(s) prepared by that organization. If further information is needed, refer to the main citation by the NUREG/CR number.

Contractor Index

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

ternal Events For Plant Operational State 5 During A Refueling Outage.Internal....

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NUREG/CR-6143 V02P1C: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF,UNIT 1.Analysis Of Core Damage Frequency From Internal Events For Plant Operational State 5 During A Refueling Outage.Main Report.

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NUREG/CR-6143 V04: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF,UNIT 1.Analysis Of Core Damage Frequency From Internally Induced Flooding Events For Plant Operational State 5 During a Refueling.....

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NUREG/CR-6144 V02P1A: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY,UNIT 1.Analysis Of Core Damage Frequency From Internal Events During Mid-Loop Operations.Main Report (Chapters 1-6).

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NUREG/CR-6208: AN EMPIRICAL INVESTIGATION OF OPERATOR PERFORMANCE IN COGNITIVELY DEMANDING SIMULATED EMERGENCIES.

International Organization Index

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

There were no NUREG/IA reports published during this quarter.

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Licensed Facility Index

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

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Washington, DC 20555–0001

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

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

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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