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Vol. 20, No. 1

Regulatory and Technical Reports (Abstract Index Journal)

Compilation for
First Quarter 1995
January – March

U.S. Nuclear Regulatory Commission

Office of Administration

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PREFACE

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

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

Secondary Report Number Index
Personal Author Index
Subject Index
NRC Originating Organization Index (Staff Reports)
NRC Originating Organization Index (International Agreements)
NRC Contract Sponsor Index (Contractor Reports)
Contractor Index
International Organization Index
Licensed Facility Index

A detailed explanation of the entries precedes each index.

The bibliographic elements of the main citations are the following:

Staff Report

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

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

Conference Report

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

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

Main Citations and Abstracts

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

NUREG-0040 V18 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report. October - December 1994. (White Book) • Office of Nuclear Reactor Regulation (Post 941001). February 1995. 200pp. 9503140385. 83069:031.

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

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

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence (AO) as an unscheduled incident or event that the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report provides a description of those events that have been determined to be abnormal occurrences during the period of July 1 through September 30, 1994. This report addresses five abnormal occurrences (AOs) at NRC-licensed facilities. One involved a medical brachytherapy misadministration, two involved medical teletherapy misadministrations, one involved a medical sodium iodide misadministration, and one involved a medical sodium iodide event. One AO report submitted by an Agreement State is included. It involved the loss of management and procedural control of a radioactive source. (Due to publication schedule constraints, NRC was unable to include all of the AO information received from the Agreement States. Any Agreement State information that was not included in this report will be published in the next quarterly report. The report also contains updates of six AOs previously reported by NRC licensees and three AOs previously reported by Agreement State licensees. Two "Other Events of Interest" concerning nuclear power reactors are also reported. One involved the fracture of a frozen pipe at Dresden Unit 1 with a consequent release of water, and the other involved the possible deliberate exposure of a contract laborer to radiation at Quad Cities Nuclear Power Station.

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

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

NUREG-0304 V19 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1994. • Division of Freedom of Information & Publications Services (Post 940714). March 1995. 133pp. 9504180345. 83566:063.

See NUREG-0304,V19,N03 abstract.

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

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

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

NRC is committed to the periodic publication of licensed fuel facility inventory difference data, following agency review of the information and completion of any related NRC investigations. Information in this report includes inventory difference data for active fuel fabrication facilities possessing more than one effective kilogram of special nuclear material.

NUREG-0540 V16 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30, 1994. • Division of Freedom of Information & Publications Services (Post 940714). January 1995. 307pp. 9502080222. 82662:001.

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

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

See NUREG-0540,V16,N11 abstract.

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

See NUREG-0540,V16,N11 abstract.

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

2 Main Citations and Abstracts

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

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

This report summarizes the occupational radiation exposure information that has been reported to the NRC's Radiation Exposure Information Reporting System (REIRS) by nuclear power facilities and certain other categories of NRC licensees during the years 1969 through 1993. The bulk of the data presented in the report was obtained from annual radiation exposure reports submitted in accordance with the requirements of 10 CFR 20.407 and the technical specifications of nuclear power plants. Data on workers terminating their employment at certain NRC licensed facilities were obtained from reports submitted pursuant to 10 CFR 20.408. The 1993 annual reports submitted by about 360 licensees indicated that approximately 189,711 individuals were monitored, 169,872 of whom were monitored by nuclear power facilities. They incurred an average individual dose of 0.16 rem (cSv) and an average measurable dose of about 0.31 (cSv). Termination radiation exposure reports were analyzed to reveal that about 99,749 individuals completed their employment with one or more of the 360 covered licensees during 1993. Some 91,000 of these individuals terminated from power reactor facilities, and about 12,685 of them were considered to be transient workers who received an average dose of 0.49 rem (cSv).

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

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

NUREG-0750 V40 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1994. Pages 169-318. * Division of Freedom of Information & Publications Services (Post 940714). February 1995. 149pp. 9503150152. 83107:001.

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

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

See NUREG-0750, V40, N05 abstract.

NUREG-0750 V41 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1995. Pages 1-69. * Division of Freedom of Information & Publications Services (Post 940714). March 1995. 75pp. 9503280211. 83302:178.
See NUREG-0750, V40, N05 abstract.

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

This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facilities throughout the country for the fourth quarter of 1994.

NUREG-0936 V13 N03: NRC REGULATORY AGENDA. Semiannual Report, July-December 1994. * Division of Freedom of Information & Publications Services (Post 940714). February 1995. 56pp. 9503140450. 83071:106.

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

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

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (October - December 1994) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to reactor licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

NUREG-0940 V13 N4 P2: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MEDICAL LICENSEES. Quarterly Progress Report, October-December 1994. * Ofc of Enforcement (Post 870413). February 1995. 300pp. 9503140434. 83077:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (October - December 1994) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to medical licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

NUREG-0940 V13 N4 P3: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MATERIAL LICENSEES (NON-MEDICAL). Quarterly Progress Report, October-December 1994. * Ofc of Enforcement (Post 870413). February 1995. 360pp. 9503140438. 83075:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (October - December 1994) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to Material Licensees (non-Medical) with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to

improve safety by avoiding future violations similar to those described in this publication.

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

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

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

As part of ongoing U.S. Nuclear Regulatory Commission (NRC) efforts to ensure the quality and accountability of safety issue information, a program was established whereby an annual NUREG report would be published on the status of licensee implementation and NRC verification of safety issues in major NRC requirements areas. This information was compiled and reported in three NUREG volumes. Volume 1, published in March 1991, addressed the status of Three Mile Island (TMI) Action Plan Requirements. Volume 2, published in May 1991, addressed the status of unresolved safety issues (USIs). Volume 3, published in June 1991, addressed the implementation and verification status of generic safety issues (GSIs). Supplement 1, published in December, 1991, combined these volumes into a single report and provided updated information as of September 30, 1991. Supplement 2, published in December, 1992, provided updated information on TMI, USI, and GSI issues and included status of all Other Multiplant Actions (MPAs). Supplement 3, published in December, 1993, provided updated information as of September 30, 1993. This annual NUREG report provides updated information on TMI, USI, GSI and other MPAs as of September 30, 1994. The data contained in these NUREG reports are a product of the NRC's Safety Issues Management system (SIMs) data base, which is maintained by the Program Management Staff in the Office of Nuclear Reactor Regulation and by NRC regional personnel. This report is to provide a comprehensive description of the implementation and verification status of TMI Action Plan Requirements, USIs, GSIs and Other MPAs that have been resolved and involve implementation of an action or actions by licensees. This report makes the information available to other interested parties, including the public. An additional purpose of this NUREG report is to serve as a follow-on to NUREG-0933, "A Prioritization of Generic Safety Issues," which tracks safety issues up until requirements are approved for imposition at licensed plants or until the NRC issues a request for action by licensees.

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

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

rent LWR licensees may voluntarily propose applications based upon it.

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

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

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

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

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

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

4 Main Citations and Abstracts

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

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

NUREG/CP-0141: PROCEEDINGS OF THE 23RD DOE/NRC NUCLEAR AIR CLEANING CONFERENCE. Held In Buffalo, New York, July 25-28, 1994. FIRST, M.W. Harvard School of Public Health, Boston, MA. February 1995. 800pp. 9503140367. CONF-940736. 83072:114.

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

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

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

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

The Workshop on Developing Safe Software was held July 22-23, 1992, at the Hotel del Coronado, San Diego, California.

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

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

This report summarizes work performed by Argonne National Laboratory (ANL) on fatigue and environmentally assisted cracking (EAC) in light water reactors during the six months from October 1993 to March 1994. Topics that have been investigated include fatigue of low-alloy steel used in piping, steam generators, and reactor pressure vessels; EAC of wrought and cast austenitic stainless steels (SSs); and radiation-induced segregation and irradiation-assisted stress corrosion cracking (IASCC) of Type 304 SS after accumulation of high fluence. Fatigue tests have been conducted on A302-Gr B low-alloy steel to verify whether the current predictions of modest decreases of fatigue life in simulated pressurized water reactor water are valid for high-sulfur heats that show environmentally enhanced fatigue crack growth rates. Additional crack growth data were obtained on fracture-mechanics specimens of austenitic SSs to investigate threshold stress intensity factors for EAC in high-purity oxygenated water at 289 degrees C. Microchemical changes in high- and commercial-purity Type 304 SS specimens from control-blade absorber tubes and a control-blade sheath from operating boiling water reactors were studied by Auger electron spectroscopy and scanning electron microscopy to determine whether trace impurity elements may contribute to IASCC of solution-annealed materials.

NUREG/CR-5462: AGING STUDY OF BOILING WATER REACTOR HIGH PRESSURE INJECTION SYSTEMS. CONLEY, D.A.; EDSON, J.L.; FINEMAN, C.F. Idaho National Engineering Laboratory. March 1995. 114pp. 9504120089. INEL-94/0090. 83474:051.

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

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

The primary goal of the Heavy-Section Steel Irradiation Program is to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior, and in particular the fracture toughness properties, of typical pressure vessel steels as they relate to light-water reactor pressure-vessel integrity. Effects of specimen size, material chemistry, product form and microstructure, irradiation fluence, flux, temperature and spectrum, and post-irradiation annealing are being examined on a wide range of fracture properties. The HSSI Program is arranged into 10 tasks: (1) program management, (2) K(lc) curve shift in high-copper welds, (3) K(la) curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K(lc) and K(la) curve shifts in low upper-shelf welds, (6) irradiation effects in a commercial low upper-shelf weld, (7) microstructural analysis of irradiation effects, (8) in-service aged material evaluations, (9) correlation monitor materials, and (10) special technical assistance. This report provides an overview of the activities within each of these tasks from October 1991 to September 1992.

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

The goal of the Heavy-Section Steel Irradiation Program is to provide a thorough, quantitative assessment of effects of neutron irradiation on material behavior, and in particular the fracture toughness properties, of typical pressure vessel steels as they relate to light-water reactor pressure-vessel integrity. Effects of specimen size, material chemistry, product form and microstructure, irradiation fluence, flux, temperature and spectrum, and post-irradiation annealing are being examined on a wide range of fracture properties. The HSSI Program is arranged into 14 tasks: (1) program management, (2) fracture toughness (K(lc)) curve shift in high-copper welds, (3) crack-arrest toughness (K(la)) curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K(lc) and K(la) curve shifts in low upper-shelf welds, (6) annealing effects in low upper-shelf welds, (7) irradiation effects in a commercial low upper-shelf weld, (8) microstructural analysis of irradiation effects, (9) in-service aged material evaluations, (10) correlation monitor materials, (11) special technical assistance, (12) JPDR steel examination, (13) technical assistance for JCCNRS Working Groups 3 and 12, and (14) additional requirements for materials. This report provides an overview of the activities within each of these tasks from April to September 1993.

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

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

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

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

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

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

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

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

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lation dose within 1000 mi of the reactor, individual early fatality probability within 1 mi of the reactor, and maximum early fatality distance.

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

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

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

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

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

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

were also estimated. The final product of the analysis was the integration of the accident frequencies with the consequences of the accidents to form an expression for aggregate risk.

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

The document contains the deterministic code calculations performed with the MELCOR Code that were used to support the development and quantification of the PRA models used in the analysis of internally initiated events for Grand Gulf, Unit 1, as it operates in the Low Power and Shutdown Plant Operational State 5 during a refueling outage. The background for the work documented in this report is summarized including how deterministic codes are used in PRAs, why the MELCOR code is used, what the capabilities and features of MELCOR are and how the code has been used by others in the past. Brief descriptions of the Grand Gulf plant and its configuration during LP&S operation and of the MELCOR input model developed for the Grand Gulf plant in its LP&S configuration are given. The results of MELCOR analyses of various accident sequences for the plant operating state (POS) 5 configuration during refueling (approximately Cold Shutdown as defined by Grand Gulf Technical Specifications) are presented for accidents initiated at several different times after scram and shutdown including shortened thermal hydraulic and core damage calculations done in support of the Level 1 analysis and full plant analyses including containment response and source terms supporting the Level 2 analysis.

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

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

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

A new multigroup cross-section library based on ENDF/B-VI data has been produced and tested for light water reactor shielding and reactor pressure vessel dosimetry applications. The broad-group library, which is designated BUGLE-93, is intended to replace the aging BUGLE-80 and SAILOR libraries. The processing methodology is consistent with ANSI/ANS 6.1.2, since the ENDF data were first processed into a fine-group, pseudo-problem-independent format and then collapsed into the final broad-group format. The fine-group library, which is designated VITAMIN-B6, contains 120 nuclides. The BUGLE-93 47-neutron-group/20-gamma-ray-group library contains the same 120 nuclides processed as infinitely dilute and collapsed

using a weighting spectrum typical of a concrete shield. Additionally, BUGLE-93 contains 105 nuclides processed with resonance self-shielding and weighted using spectra specific to BWR and PWR material compositions and reactor models. Several dosimetry response functions and kerma factors for all 120 nuclides are also included with the library. An extensive integral data testing effort was performed to qualify the new library. In general, results using the new data show significant improvements relative to earlier ENDF data.

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

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

NUREG/CR-6240: APPLICATION OF BOUNDING SPECTRA TO SEISMIC DESIGN OF PIPING BASED ON THE PERFORMANCE OF ABOVE GROUND PIPING IN POWER PLANTS SUBJECTED TO STRONG MOTION EARTHQUAKES. STEVENSON, J.D. Stevenson & Associates. * Oak Ridge National Laboratory. February 1995. 104pp. 9503140368. ORNL SUB94SD4271. 83069:312.

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

NUREG/CR-6242: CASKS (COMPUTER ANALYSIS OF STORAGE CASKS): A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR STORAGE CASK DESIGN REVIEW. User's Manual To Version 1b (Including Program Reference). CHEN, T.F.; GERHARD, M.A.; TRUMMER, D.J.; et al. Lawrence Livermore National Laboratory. February 1995. 200pp. 9503140405. UCRL-ID-117418. 83070:058.

CASKS (Computer Analysis of Storage CaskS) is a microcomputer-based system of computer programs and databases developed at the Lawrence Livermore National Laboratory (LLNL) for evaluating safety analysis reports on spent-fuel storage casks. The bulk of the complete program and this user's manual are based upon the SCANS (Shipping Cask Analysis System) program previously developed at LLNL. A number of enhancements and improvements were added to the original SCANS program to meet requirements unique to storage casks. CASKS is an easy-to-use system that calculates global response of storage casks to impact loads, pressure loads and thermal conditions. This provides reviewers with a tool for an in-

dependent check on analyses submitted by licensees. CASKS is based on microcomputers compatible with the IBM-PC family of computers. The system is composed of a series of menus, input programs, cask analysis programs, and output display programs. All data is entered through fill-in-the-blank input screens that contain descriptive data requests.

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

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

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

See NUREG/CR-6244, V01 abstract.

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

See NUREG/CR-6244, V01 abstract.

NUREG/CR-6257: CANDU 3 TRANSIENT ANALYSIS USING ATOMIC ENERGY OF CANADA LTD CODES. JEDD, J.L.; SHUMWAY, R.W. Idaho National Engineering Laboratory. EBERT, D.D. Reactor & Plant Systems Branch (Post 941217). January 1995. 88pp. 9503270305. INEL-95/0070. 83268:267.

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

8 Main Citations and Abstracts

NUREG/CR-6259: CONSTRAINT EFFECTS ON FRACTURE INITIATION LOADS IN HSST WIDE-PLATE TESTS. DODDS,R.H. Illinois, Univ. of, Urbana, IL. DODDS,R.H. Oak Ridge National Laboratory. December 1994. 42pp. 9502080251. ULLUENG942009. 82660:304.

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

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

Recent test data indicate that the effects of the light water reactor (LWR) environment could significantly reduce the fatigue resistance of materials used in the reactor coolant pressure boundary components of operating nuclear power plants. Argonne National Laboratory has developed interim fatigue curves based on test data simulating LWR conditions, and published them in NUREG/CR-5999. In order to assess the significance of these interim fatigue curves, fatigue evaluations of a sample of the components in the reactor coolant pressure boundary of LWRs were performed. The sample consists of components from facilities designed by each of the four U.S. nuclear steam supply system vendors. For each facility, six locations were studied, including two locations on the reactor pressure vessel. In addition, there are older vintage plants where components of the reactor coolant pressure boundary were designed to codes that did not require an explicit fatigue analysis of the components. In order to assess the fatigue resistance of the older vintage plants, an evaluation was also conducted on selected components of three of these plants. This report discusses the insights gained from the application of the interim fatigue curves to components of seven operating nuclear power plants.

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

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

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

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

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

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

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

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

The preliminary phases of a program to develop and evaluate fracture methodologies for the assessment of crack-tip constraint effects on fracture toughness of reactor pressure vessel

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

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

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

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

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

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

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

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

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

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

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

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

10 Main Citations and Abstracts

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

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

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

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

See NUREG/CR-6293, V01 abstract.

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

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

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

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

suggestions for treating structural uncertainty for submodels are presented.

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

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

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

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

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

This report is the third volume in a series of reports describing the results of the Expert System Verification and Validation (V&V) project that is jointly funded by the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute to develop guidelines for the V&V of expert and other systems. The purpose of this activity was to survey and document techniques presently in use for expert systems V&V. Via extensive

telephone contacts, site visits, and through bibliographic searches a wide sampling of expert system V&V was accomplished. The major finding was that V&V of expert systems is not nearly as established or prevalent as V&V of conventional software systems. There were few examples of V&V in the early stage of development. However, there is a very active research area concerning the development of methods to assess the knowledge bases of expert and knowledge-based systems.

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

This report is the fourth volume in a series of reports describing the results of the Expert System Verification and Validation (V&V) project that is jointly funded by the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute to develop guidelines for the V&V of expert and other systems. Here are presented the results of the Knowledge Base Certification activity that was concerned with developing and testing various static analysis methods for assuring the quality of knowledge bases. The testing procedure used was that of a behavioral experiment involving evaluation of four different V&V methods. The study used two real nuclear expert systems: a boiling water reactor emergency operating procedures tracking system, and a pressurized water reactor safety assessment system. The twenty participants were from three nuclear utilities, the USNRC Technical Training Center, the University of Maryland, EG&G Idaho, and SAIC. The major conclusion was that the use of tools in static knowledge base certification results in significant improvement in detecting all types of defects, avoiding false alarms, and completing the effort in less time. The simulated knowledge-checking tool, based on supplemental information, was the most effective of the tools.

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

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

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

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

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

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

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

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

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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ADLER,M.V.

NUREG-1514: GUIDANCE FOR A LARGE TABLETOP EXERCISE FOR A NUCLEAR POWER PLANT.

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

AMARASOORIYA,H.

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

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

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

BROWN,T.D.

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

BROWN,W.S.

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

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

CALLEY,M.B.

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

CAMPBELL,V.

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

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

CAPPS,E.L.

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

CHEN,P.

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

CHEN,T.F.

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

CHOPRA,O.K.

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

CHUNG,H.M.

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

CONLEY,D.A.

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

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NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment.Main Report.
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 NUREG/CR-6291 V04: NUCLEAR PLANT ANALYZER. Programmer's Manual.

Radiation Dose

NUREG/CP-0143: PROCEEDINGS OF THE THIRD INTERNATIONAL WORKSHOP ON THE IMPLEMENTATION OF ALARA AT NUCLEAR POWER PLANTS. Held At Hauppauge, Long Island, New York.

Radiation Exposure

NUREG-0713 V15: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES, 1993. Twenty-Sixth Annual Report.
 NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

Radioactive Material Safety Program

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

Reactor

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

Reactor Accident

NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
 NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

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Reactor Pressure Vessel

NUREG/CR-6214: PRODUCTION AND TESTING OF THE VITAMIN-B6 FINE GROUP AND THE BUGLE-93 BROAD-GROUP NEUTRON/PHOTON CROSS-SECTION LIBRARIES DERIVED FROM ENDF/B-VI NUCLEAR DATA.
NUREG/CR-6273: BIAXIAL LOADING EFFECTS ON FRACTURE TOUGHNESS OF REACTOR PRESSURE VESSEL STEEL.

Reactor Safety

NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.
NUREG/CR-6311: EVALUATING PREDICTION UNCERTAINTY.

Reactor accident

NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.

Refueling Outage

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

Regulatory Agenda

NUREG-0936 V13 N03: NRC REGULATORY AGENDA. Semiannual Report, July-December 1994.

Regulatory And Technical Report

NUREG-0304 V19 N03: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Third Quarter 1994, July-September.
NUREG-0304 V19 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1994.

Relay

NUREG/CR-6220: AN ASSESSMENT OF FIRE VULNERABILITY FOR AGED ELECTRICAL RELAYS.

Relief Valve

NUREG/CR-6192: AGING AND SERVICE WEAR OF SPRING-LOADED PRESSURE RELIEF VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.

Remote Afterloading

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

Risk Analysis

NUREG/CR-6311: EVALUATING PREDICTION UNCERTAINTY.

Rules

NUREG-0936 V13 N03: NRC REGULATORY AGENDA. Semiannual Report, July-December 1994.

SAPHIRE

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

Safety Criteria

NUREG/CR-6284: CRITICALITY SAFETY CRITERIA FOR LICENSE REVIEW OF LOW-LEVEL WASTE FACILITIES.

Safety Issue

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

Safety Issues

NUREG-1435 S04: STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS. TMI Action Plan Requirements, Unresolved Safety Issues, Generic Safety Issues, Other Multiplant Action Issues.

Safety-Related System

NUREG/CR-6192: AGING AND SERVICE WEAR OF SPRING-LOADED PRESSURE RELIEF VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.

Seismic Design

NUREG/CR-6240: APPLICATION OF BOUNDING SPECTRA TO SEISMIC DESIGN OF PIPING BASED ON THE PERFORMANCE OF ABOVE GROUND PIPING IN POWER PLANTS SUBJECTED TO STRONG MOTION EARTHQUAKES.

Sensitivity Analysis

NUREG/CR-6134: UNCERTAINTY AND SENSITIVITY ANALYSIS OF CHRONIC EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
NUREG/CR-6135: UNCERTAINTY AND SENSITIVITY ANALYSIS OF EARLY EXPOSURE RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
NUREG/CR-6136: UNCERTAINTY AND SENSITIVITY ANALYSIS OF FOOD PATHWAY RESULTS WITH THE MACCS REACTOR ACCIDENT CONSEQUENCE MODEL.
NUREG/CR-6311: EVALUATING PREDICTION UNCERTAINTY.

Severe Accident

NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS.
NUREG/CR-6143 V06 P1: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Main Report And Appendices.
NUREG/CR-6143 V06 P2: EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT GRAND GULF, UNIT 1. Evaluation Of Severe Accident Risks For Plant Operational State 5 During A Refueling Outage. Supporting MELCOR Calculations.
NUREG/CR-6285: SEVERE ACCIDENT NATURAL CIRCULATION STUDIES AT THE INEL.

Shutdown Operation

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

Software

NUREG/CP-0145: WORKSHOP ON DEVELOPING SAFE SOFTWARE. Held At Hotel Del Coronado, San Diego, CA, July 22-23, 1992.
NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Main Report.
NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Appendices A-D.
NUREG/CR-6316 V01: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.
NUREG/CR-6316 V02: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Assessment Of Conventional Software Verification And Validation Methods.
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NUREG/CR-6316 V04: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Evaluation Of Knowledge Base Certification Methods.
NUREG/CR-6316 V05: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Rationale And Description Of V&V Guideline Packages And Procedures.
NUREG/CR-6316 V06: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Validation Scenarios.
NUREG/CR-6316 V07: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. User's Manual.
NUREG/CR-6316 V08: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Bibliography.

Source Term

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

South Texas Project

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

Storage Cask

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

Stress Corrosion Cracking

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

Stress Triaxiality

NUREG/CR-6259: CONSTRAINT EFFECTS ON FRACTURE INITIATION LOADS IN HSST WIDE-PLATE TESTS.

Surveillance Requirement

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

System Safety

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

TLD

NUREG-0837 V14 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. October-December 1994.

TMI Action Plan

NUREG-1435 S04: STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS. TMI Action Plan Requirements, Unresolved Safety Issues, Generic Safety Issues, Other Multiplant Action Issues.

Tabletop

NUREG-1514: GUIDANCE FOR A LARGE TABLETOP EXERCISE FOR A NUCLEAR POWER PLANT.

Technical Specifications

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

Thermohydraulic

NUREG/CR-6257: CANDU 3 TRANSIENT ANALYSIS USING ATOMIC ENERGY OF CANADA LTD CODES.

Thermoluminescent Dosimeter

NUREG-0837 V14 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report. October-December 1994.

Title List

NUREG-0540 V16 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30, 1994.
NUREG-0540 V16 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. December 1-31, 1994.
NUREG-0540 V17 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31, 1995.

Topical Report

NUREG-0390 V09 N01: TOPICAL REPORT REVIEW STATUS.

Transient Analysis

NUREG/CR-6257: CANDU 3 TRANSIENT ANALYSIS USING ATOMIC ENERGY OF CANADA LTD CODES.

Uncertainty Analysis

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

VITAMIN-B6

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

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

Validation

NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Main Report.
NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Appendices A-D.
NUREG/CR-6316 V01: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.
NUREG/CR-6316 V02: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Assessment Of Conventional Software Verification And Validation Methods.
NUREG/CR-6316 V03: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Documentation Of Expert System Verification And Validation Methodologies.
NUREG/CR-6316 V04: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Evaluation Of Knowledge Base Certification Methods.
NUREG/CR-6316 V05: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Rationale And Description Of V&V Guideline Packages And Procedures.
NUREG/CR-6316 V06: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Validation Scenarios.
NUREG/CR-6316 V07: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. User's Manual.
NUREG/CR-6316 V08: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Bibliography.

Validity Limit

NUREG/CR-6264 V01: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. An Assessment Of The J(M) Parameter.
NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION. A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions.

Vendor Inspection

NUREG-0040 V18 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report. October - December 1994. (White Book)

Verification

NUREG/CR-6293 V01: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Main Report.
NUREG/CR-6293 V02: VERIFICATION AND VALIDATION GUIDELINES FOR HIGH INTEGRITY SYSTEMS. Appendices A-D.
NUREG/CR-6316 V01: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE.
NUREG/CR-6316 V02: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Assessment Of Conventional Software Verification And Validation Methods.
NUREG/CR-6316 V03: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Survey And Documentation Of Expert System Verification And Validation Methodologies.
NUREG/CR-6316 V04: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Evaluation Of Knowledge Base Certification Methods.
NUREG/CR-6316 V05: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Rationale And Description Of V&V Guideline Packages And Procedures.
NUREG/CR-6316 V06: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Validation Scenarios.
NUREG/CR-6316 V07: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. User's Manual.
NUREG/CR-6316 V08: GUIDELINES FOR THE VERIFICATION AND VALIDATION OF EXPERT SYSTEM SOFTWARE AND CONVENTIONAL SOFTWARE. Bibliography.

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

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

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

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

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

Wide-Plate

NUREG/CR-6259: CONSTRAINT EFFECTS ON FRACTURE INITIATION LOADS IN HSST WIDE-PLATE TESTS.

NRC Originating Organization Index (Staff Reports)

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

OFFICE OF EXECUTIVE DIRECTOR FOR OPERATIONS (EDO) REGION 1 (POST 820201)

NUREG-0837 V14 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report, October-December 1994.
OFC OF ENFORCEMENT (POST 870413)
NUREG-0940 V13 N4 P1: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED REACTOR LICENSEES. Quarterly Progress Report, October-December 1994.
NUREG-0940 V13 N4 P2: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MEDICAL LICENSEES. Quarterly Progress Report, October-December 1994.
NUREG-0940 V13 N4 P3: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MATERIAL LICENSEES (NON-MEDICAL). Quarterly Progress Report, October-December 1994.
OFC OF INVESTIGATIONS (POST 880201)
NUREG-1517: REPORT OF THE SOUTH TEXAS PROJECT ALLEGATIONS REVIEW TEAM.

EDO - OFFICE OF ADMINISTRATION (PRE 870413 & POST 890205)

DIVISION OF FREEDOM OF INFORMATION & PUBLICATIONS SERVICES (POST 940714)
NUREG-0304 V19 N03: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Third Quarter 1994, July-September.
NUREG-0304 V19 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1994.
NUREG-0540 V16 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30, 1994.
NUREG-0540 V16 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. December 1-31, 1994.
NUREG-0540 V17 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31, 1995.
NUREG-0750 V40 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-December 1994.
NUREG-0750 V40 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1994. Pages 169-318.
NUREG-0750 V40 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1994. Pages 319-387.
NUREG-0750 V41 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1995. Pages 1-69.
NUREG-0936 V13 N03: NRC REGULATORY AGENDA. Semiannual Report, July-December 1994.

EDO - OFFICE OF THE CONTROLLER (PRE 820418 & POST 890205) DIVISION OF BUDGET & ANALYSIS (POST 890205)

NUREG-1100 V11: BUDGET ESTIMATES. Fiscal Years 1996-1997.

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA

OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA, DIRECTOR
NUREG-0090 V17 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. July-September 1994.

NUREG-1514: GUIDANCE FOR A LARGE TABLETOP EXERCISE FOR A NUCLEAR POWER PLANT.

EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS

OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
NUREG-0430 V14: LICENSED FUEL FACILITY STATUS REPORT. Inventory Difference Data. July 1, 1993 - June 30, 1994.
DIVISION OF INDUSTRIAL & MEDICAL NUCLEAR SAFETY (POST 870729)
NUREG-1516 DRFT FC: MANAGEMENT OF RADIOACTIVE MATERIAL SAFETY PROGRAMS AT MEDICAL FACILITIES. Draft Report For Comment.

U.S. NUCLEAR REGULATORY COMMISSION

NRC - NO DETAILED AFFILIATION GIVEN
NUREG/CR-6244 V03: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion and Deposition Uncertainty Assessment. Appendices C, D, E, F, G, H.

EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)

DIVISION OF REGULATORY APPLICATIONS (POST 941217)
NUREG-0713 V15: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES. 1993. Twenty-Sixth Annual Report.
NUREG-1493 DFC: PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM. Draft Report For Comment.
DIVISION OF SYSTEMS TECHNOLOGY (POST 941217)
NUREG-0700 R01 DFC: HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE. Draft Report For Comment.
NUREG-1465: ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS.
PROBABILISTIC RISK ANALYSIS BRANCH (POST 941217)
NUREG/CR-6244 V01: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Main Report.
NUREG/CR-6244 V02: PROBABILISTIC ACCIDENT CONSEQUENCE UNCERTAINTY ANALYSIS. Dispersion And Deposition Uncertainty Assessment. Appendices A And B.
REACTOR & PLANT SYSTEMS BRANCH (POST 941217)
NUREG/CR-6257: CANDU 3 TRANSIENT ANALYSIS USING ATOMIC ENERGY OF CANADA LTD CODES.

EDO - OFFICE OF NUCLEAR REACTOR REGULATION (POST 800428)

OFFICE OF NUCLEAR REACTOR REGULATION (POST 941001)
NUREG-0040 V18 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report. October - December 1994. (White Book)
NUREG-0390 V09 N01: TOPICAL REPORT REVIEW STATUS.
NUREG-1435 S04: STATUS OF SAFETY ISSUES AT LICENSED POWER PLANTS. TMI Action Plan Requirements, Unresolved Safety Issues, Generic Safety Issues, Other Multiplant Action Issues.
NUREG-1517: REPORT OF THE SOUTH TEXAS PROJECT ALLEGATIONS REVIEW TEAM.

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.

NRC Contract Sponsor Index (Contractor Reports)

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

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA

DIVISION OF SAFETY PROGRAMS (POST 870413)
NUREG/CR-6266: ANALYSIS OF BORON DILUTION IN A FOUR-LOOP PWR.

EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS DIVISION OF INDUSTRIAL & MEDICAL NUCLEAR SAFETY (POST 870729)

NUREG/CR-6276: A COMPILATION OF CURRENT REGULATIONS, STANDARDS, AND GUIDELINES IN REMOTE AFTERLOADING BRACHYTHERAPY.
DIVISION OF WASTE MANAGEMENT (NMSS 940403)
NUREG/CR-6284: CRITICALITY SAFETY CRITERIA FOR LICENSE REVIEW OF LOW-LEVEL WASTE FACILITIES.

EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)

DIVISION OF SYSTEMS RESEARCH (880717-941217)
NUREG/CR-6141: HANDBOOK OF METHODS FOR RISK-BASED ANALYSES OF TECHNICAL SPECIFICATIONS.

OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 941217)

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

NUREG/CR-6259: CONSTRAINT EFFECTS ON FRACTURE INITIATION LOADS IN HSST WIDE-PLATE TESTS.

DIVISION OF ENGINEERING TECHNOLOGY (POST 941217)

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

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

NUREG/CR-5591 V03: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Progress Report For October 1991 - September 1992.

NUREG/CR-5591 V04 N2: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For April-September 1993.

NUREG/CR-6192: AGING AND SERVICE WEAR OF SPRING-LOADED PRESSURE RELIEF VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.

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

NUREG/CR-6220: AN ASSESSMENT OF FIRE VULNERABILITY FOR AGED ELECTRICAL RELAYS.

NUREG/CR-6240: APPLICATION OF BOUNDING SPECTRA TO SEISMIC DESIGN OF PIPING BASED ON THE PERFORMANCE OF ABOVE GROUND PIPING IN POWER PLANTS SUBJECTED TO STRONG MOTION EARTHQUAKES.

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

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

NUREG/CR-6264 V02: VALIDITY LIMITS IN J-RESISTANCE CURVE DETERMINATION.A Computational Approach To Ductile Crack Growth Under Large-Scale Yielding Conditions.

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

DIVISION OF REGULATORY APPLICATIONS (POST 941217)

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

DIVISION OF SYSTEMS TECHNOLOGY (POST 941217)

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OFFICE OF NUCLEAR REACTOR REGULATION (POST 941001)
NUREG/CR-6260: APPLICATION OF NUREG/CR-5999 INTERIM FATIGUE CURVES TO SELECTED NUCLEAR POWER PLANT COMPONENTS.

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.

ARGONNE NATIONAL LABORATORY

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

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

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STEVENSON & ASSOCIATES

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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; as well as international agreement reports. The entries in this compilation are indexed for access by title and abstract, secondary report number, personal author, subject, NRC organization for staff and international agreements, contractor, international organization, and licensed facility.

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

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