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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)
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NRC Contract Sponsor Index (Contractor Reports)
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A detailed explanation of the entries precedes each index.

The bibliographic elements of the main citations are the following:

Staff Report

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

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

Conference Report

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

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

Contractor Report

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

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

International Agreement Report

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

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

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

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

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

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NRC Report Codes

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

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

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

Main Citations and Abstracts

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

NUREG-0040 V17 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, April-June 1993. (White Book) * Division of Reactor Inspection & Licensee Performance (Post 921004). August 1993. 109pp. 9309210034. 76483:049.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from April through June 1993.

NUREG-0090 V16 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. January-March 1993. * Office for Analysis & Evaluation of Operational Data, Director. June 1993. 30pp. 9307220177. 75743:332.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event that the Nuclear Regulatory Commission determines to be significant from the standpoint of public health and safety and requires a quarterly report of such events to be made to Congress. This report covers the period January through March 1993. There is one abnormal occurrence at a nuclear power plant discussed in this report that involved a steam generator tube rupture at Palo Verde Unit 2, and none for fuel cycle facilities. Three abnormal occurrences involving medical misadministrations (two therapeutic and one diagnostic) at NRC-licensed facilities are also discussed in this report. No abnormal occurrences were reported by NRC's Agreement States. The report also contains information updating previously reported abnormal occurrences.

NUREG-0304 V18 N02: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Second Quarter 1993, April-June. * Division of Freedom of Information & Publications Services (Post 890205). August 1993. 48pp. 9309210040. 76483:158.

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

NUREG-0386 D06 R07: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST. Commission, Appeal Board And Licensing Board Decisions. July 1972 - September 1992. * Office of the General Counsel (Post 860701). August 1993. 600pp. 9309090041. 76379:138.

This 7th revision of the sixth edition of the NRC Practice and Procedure Digest contains a digest of a number of Commission, Atomic Safety and Licensing Appeal Board, and Atomic Safety and Licensing Board decisions issued during the period of July 1, 1972 to September 30, 1992, interpreting the NRC's Rules of Practice in 10 CFR Part 2.

NUREG-0525 V02 R01: SAFEGUARDS SUMMARY EVENT LIST (SSEL). January 1, 1990 Through December 31, 1992. FADDEIJ, M.; YARDUMIAN, J. Operations Branch. July 1993. 250pp. 9308160105. 76142:001.

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

NUREG-0540 V15 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. May 1-31, 1993. * Division of Freedom of Information & Publications Services (Post 890205). July 1993. 350pp. 9308160099. 76115:010.

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 V15 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. June 1-30, 1993. * Division of Freedom of Information & Publications Services (Post 890205). August 1993. 421pp. 9309030160. 76325:158.

See NUREG-0540, V15, N05 abstract.

NUREG-0540 V15 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. July 1-31, 1993. * Division of Freedom of Information & Publications Services (Post 890205). September 1993. 350pp. 9309210229. 76485:077.

See NUREG-0540, V15, N05 abstract.

NUREG-0713 V13: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES, 1991. Twenty-Fourth Annual Report. RADDATZ, C.T. Division of Regulatory Applications (Post 870413). HAGEMeyer, D. Science Applications International Corp. (formerly Science Applications, Inc.). July 1993. 300pp. 9308160144. 76114:077.

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 1991. The bulk of the data presented in the report was obtained from annual radiation exposure reports submitted in accordance with the requirements of 10CFR20.407 and the technical specifications of nuclear power plants. Data

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on workers terminating their employment at certain NRC licensed facilities were obtained from reports submitted pursuant to 10CFR20.408. The 1991 annual reports submitted by about 436 licensees indicated that approximately 206,732 individuals were monitored, 182,334 of whom were monitored by nuclear power facilities. They incurred an average individual dose of 0.15 rem (cSv) and an average measurable dose of about 0.31 (cSv). Termination radiation exposure reports were analyzed to reveal that about 96,231 individuals completed their employment with one or more of the 436 covered licensees during 1991. Some 68,115 of these individuals terminated from power reactor facilities, and about 7,763 of them were considered to be transient workers who received an average dose of 0.52 rem (cSv).

NUREG-0750 V37 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-March 1993. * Division of Freedom of Information & Publications Services (Post 890205). July 1993. 55pp. 9308160096. 76122:200.

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 V37 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1993. Pages 251-354. * Division of Freedom of Information & Publications Services (Post 890205). July 1993. 110pp. 9308160091. 76119:007.

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

NUREG-0750 V37 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1993. Pages 355-418. * Division of Freedom of Information & Publications Services (Post 890205). August 1993. 69pp. 9308190001. 76152:038.

See NUREG-0750,V37,N04 abstract.

NUREG-0750 V37 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1993. Pages 419-515. * Division of Freedom of Information & Publications Services (Post 890205). August 1993. 103pp. 9309210012. 76481:034.

See NUREG-0750,V37,N04 abstract.

NUREG-0750 V38 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1993. Pages 1-24. * Division of Freedom of Information & Publications Services (Post 890205). September 1993. 32pp. 9310120315. 76741:114.

See NUREG-0750,V37,N04 abstract.

NUREG-0837 V13 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK Progress Report. April-June 1993. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). August 1993. 250pp. 9309210028. 76482:103.

This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facilities throughout the country for the second quarter of 1993.

NUREG-0910 R02 S01: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE. * Division of Information Support Services (Post 890205). September 1993. 89pp. 9310120062. 76738:121.

The approved records disposition schedules specify the appropriate duration of retention and the final disposition for records created or maintained by the NRC. NUREG-0910, Revision 2, Supplement 1 makes editorial and administrative changes to the NRC Schedule and forwards 3 sets of changes to the National Archives and Records Administration's General Record Schedule.

NUREG-0936 V12 N02: NRC REGULATORY AGENDA. Quarterly Report, April-June 1993. * Division of Freedom of Information & Publications Services (Post 890205). July 1993. 138pp. 9308160147. 76121:001.

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

NUREG-0940 V12 N02: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED. Quarterly Progress Report, April-June 1993. * Ofc of Enforcement (Post 870413). September 1993. 395pp. 9310120039. 76737:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1993) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

NUREG-1145 V09: U.S. NUCLEAR REGULATORY COMMISSION 1992 ANNUAL REPORT. * Office of Administration (Post 890205). July 1993. 287pp. 9309090029. 76391:001.

This report covers the major activities, events, decisions and planning that took place during fiscal year 1992 within the U.S. Nuclear Regulatory Commission (NRC) or involving the NRC.

NUREG-1214 R12: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. ALLENSPACH,F. Division of Reactor Inspection & Licensee Performance (Post 921004). August 1993. 134pp. 9309210018. 76500:101.

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

NUREG-1272 V07 N01: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA. 1992 Annual Report - Power Reactors. * Office for Analysis & Evaluation of Operational Data, Director. July 1993. 300pp. 9309210044. 76483:206.

The annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) is devoted to the activities performed during 1992. The report is published in two separate parts. NUREG-1272, Vol. 7, No. 1, covers power reactors and presents an overview of the operating experience of the nuclear power industry from the NRC perspective, including comments about the trends of some key performance measures. The report also includes the principal findings and issues identified in AEOD studies over the past year and summarizes information from such sources as licensee event reports, diagnostic evaluations, and reports to the NRC's Operations Center. The reports contain a discussion of the Incident Investigation Team program and summarize the Incident Investigation Team and Augmented Inspection Team reports for that group of licensees. NUREG-1272, Vol. 7, No. 2, covers nonreactors and presents a review of the events and concerns during 1992 associated with the use of licensed material in non-reactor applications, such as personnel overexposures and medical misadministrations. Each volume contains a list of the AEOD reports issued for 1984-1992.

NUREG-1363 V05: ATOMIC SAFETY AND LICENSING BOARD PANEL ANNUAL REPORT Fiscal Year 1992. COTTER, B.P. Atomic Safety & Licensing Board Panel. September 1993. 43pp. 9310120269. 76761 001

In Fiscal Year 1992, the Atomic Safety and Licensing Board Panel ("the Panel") handled 38 proceedings. The cases addressed issues in the construction, operation, and maintenance of commercial nuclear power reactors and other activities requiring a license from the Nuclear Regulatory Commission. This report sets out the Panel's caseload during the year and summarizes, highlights, and analyzes how the wide-ranging issues raised in those proceedings were addressed by the Panel's judges and licensing boards.

NUREG-1400: AIR SAMPLING IN THE WORKPLACE Final Report. HICKEY, E.E.; STOETZEL, G.A.; STROM, D.J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1993. 104pp. 9310120325. 76742 181.

This report provides technical information on air sampling that will be useful for facilities following the recommendations in the NRC's Regulatory Guide 8.25, Revision 1, "Air Sampling in the Workplace." That guide addresses air sampling to meet the requirements in NRC's regulations on radiation protection, 10CFR20. This report describes how to determine the need for air sampling based on the amount of material in process modified by the type of material, release potential, and confinement of the material. The purposes of air sampling and how the purposes affect the types of air sampling provided are discussed. The report discusses how to locate air samplers to accurately determine the concentrations of airborne radioactive materials that workers will be exposed to. The need for and the methods of performing airflow pattern studies to improve the accuracy of air sampling results are included. The report presents and gives examples of several techniques that can be used to evaluate whether the airborne concentrations of material are representative of the air inhaled by workers. Methods to adjust derived air concentrations for particle size are described. Methods to calibrate for volume of air sampled and estimate the uncertainty in the volume of air sampled are described. Statistical tests for determining minimum detectable concentrations are presented. How to perform an annual evaluation of the adequacy of the air sampling is also discussed.

NUREG-1423 V04: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON NUCLEAR WASTE July 1992 - June 1993. * Advisory Committee on Nuclear Waste. August 1993. 81pp. 9309210031. 76482 337

This compilation contains 17 reports issued by the Advisory Committee on Nuclear Waste (ACNW) during the fifth year of its operation. The reports were submitted to the Chairman and Commissioners of the U.S. Nuclear Regulatory Commission, the Executive Director for Operations, the Director, Office of Nuclear Material Safety and Safeguards, or to the Director, Division of High Level Waste Management, Office of Nuclear Material Safety and Safeguards. All reports prepared by the Committee have been made available to the public through the NRC Public Document Room and the U.S. Library of Congress.

NUREG-1449: SHUTDOWN AND LOW-POWER OPERATION AT NUCLEAR POWER PLANTS IN THE UNITED STATES Final Report. * Division of Systems Safety & Analysis (Post 921004). September 1993. 200pp. 9310130052. 76743 001

The report contains the results of the NRC staff's evaluation of shutdown and low-power operations at U.S. commercial nuclear power plants. The report describes studies conducted by the staff in the following areas: operating experience related to shutdown and low-power operations; probabilistic risk assessment of shutdown and low-power conditions and utility programs for planning and conducting activities during periods the plant is shut down. The report also documents evaluations of a number of technical issues regarding shutdown and low-power operations performed by the staff, including the principal findings and conclusions. Potential new regulatory requirements are

discussed, as well as potential changes in NRC programs. A draft report was issued for comment in February 1992. This report is the final version and includes the responses to the comments along with the staff regulatory analysis of potential new requirements.

NUREG-1453: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC ISSUE 142: LEAKAGE THROUGH ELECTRICAL ISOLATORS IN INSTRUMENTATION CIRCUITS. HOURK, C.J. Division of Safety Issue Resolution (Post 880717). September 1993. 21pp. 9310130042. 76744 266.

Generic Issue (GI) 142 deals with staff concerns about the design of isolation devices used to ensure separation between Class 1E and non-Class 1E electrical control and instrumentation circuits. This issue was initiated in June 1987. Staff reviews of the implementation of the Safety Parameter Display System (SPDS) requirement indicated that some isolation devices used to provide an interface between the non-Class 1E SPDS and the Class 1E safety systems would allow signal leakage if electrically challenged. It was unknown if the amount of leakage posed a hazard to safe operation of the Class 1E system. A review of failure records does not reveal any incidents of system damage caused by isolation device challenge. Furthermore, a review of existing PRA data indicates that the safety significance of ID challenge is low, at the expected challenge event frequency. However, based upon the potential design variations in future control systems resulting from application of computer technology, additional design and qualification test requirements for future plants are recommended.

NUREG-1461: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC ISSUE 153: LOSS OF ESSENTIAL SERVICE WATER IN LWRs. SU, T.-M. Division of Safety Issue Resolution (Post 880717). August 1993. 32pp. 9309030241. 76328 181

In this report, the staff of the U.S. Nuclear Regulatory Commission provides a regulatory analysis for the proposed resolution of Generic Issue 153 (GI-153), "Loss of Essential Service Water in LWRs." GI-153 deals with the concerns pertaining to the reliability of essential service water (ESW) system and related problems for all light water reactors except the seven multi-unit sites addressed by GI-130, "Essential Service Water Pump failures at Multi-Unit Sites." On the basis of the technical findings of a scoping study for GI-153, the staff recommends that the insights gained from the study serve as a complement to the on-going ESW performance inspection program. The staff also concludes that ESW system reliability is being addressed by various on-going regulatory programs. Therefore, the staff recommends that GI-153 should be considered "RESOLVED." The need for future action(s) on ESW reliability is expected to be determined from these on-going programs.

NUREG-1463: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC SAFETY ISSUE 105: INTERFACING SYSTEM LOSS-OF-COOLANT ACCIDENT IN LIGHT-WATER REACTORS. * Division of Safety Issue Resolution (Post 880717). July 1993. 77pp. 9308160153. 76114 001

An interfacing systems loss of coolant accident (ISLOCA) involves failure or improper operation of pressure isolation valves (PIVs) that compose the boundary between the reactor coolant system and low-pressure rated systems. Some ISLOCAs can bypass containment and result in direct release of fission products to the environment. A cost/benefit evaluation, using three PWR analyses, calculated the benefit of two potential modifications to the plants. Alternative I is improved plant operations to optimize the operator's performance and reduce human error probabilities. Alternative II adds pressure sensing devices, cabling, and instrumentation between two PIVs to provide operators with continuous monitoring of the first PIV. These two alternatives were evaluated for the base case plants (Case 1) and for each plant, assuming the plants had a particular auxiliary building design in which severe flooding would be a problem if

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an ISLOCA occurred. The auxiliary building design (Case 2) was selected from a survey that revealed a number of designs with features that provided less than optimal resistance to ECCS equipment loss caused by a ISLOCA-induced environment. The results were judged not to provide sufficient basis for generic requirements. It was concluded that the most viable course of action to resolve Generic Issue 105 is licensee participation in individual plant examinations (IPEs).

NUREG-1470 V02: CHIEF FINANCIAL OFFICER'S ANNUAL REPORT -- 1993. * Office of the Controller (Post 890205). September 1993. 116pp. 9311030220.

The Chief Financial Officers Act of 1990 requires the NRC Chief Financial Officer to prepare and submit an annual report to the agency head and the Director of the Office of Management and Budget. This 1993 report is the second annual report for the NRC and includes a description and analysis of the status of financial management for Fiscal Year 1993, an audited financial statement and audit reports for Fiscal Year 1992, and a summary of the reports on internal accounting and administrative control systems for 1992.

NUREG-1476: FINAL ENVIRONMENTAL IMPACT STATEMENT TO CONSTRUCT AND OPERATE A FACILITY TO RECEIVE, STORE, AND DISPOSE OF 11E.(2) BYPRODUCT MATERIAL NEAR CLIVE, UTAH. Docket No. 40-8989, Envirocare Of Utah, Inc. BRUMMETT, E.; ABU-EID, R.; MULLINS, A.; et al. Division of Low-Level Waste Management & Decommissioning (Post 870413). August 1993. 206pp. 9309210022. 76481:269.

A Final Environmental Impact Statement (FEIS) related to the licensing of Envirocare of Utah, Inc.'s proposed disposal facility in Tooele County, Utah (Docket No. 40-8989) for byproduct material as defined in Section 11e.(2) of the Atomic Energy Act, as amended, has been prepared by the Office of Nuclear Material Safety and Safeguards. This statement describes and evaluates: (1) the purpose of and need for the proposed action; (2) the alternatives considered; and (3) the environmental consequences of the proposed action. The NRC has concluded that the proposed action evaluated under the National Environmental Policy Act of 1969 (NEPA) and 10 CFR Part 51, is to permit the applicant to proceed with the project as described in this Statement.

NUREG-1479: RESULTS FROM TWO WORKSHOPS: STATE RADIATION CONTROL PROGRAMS DEVELOPING AND AMENDING REGULATIONS AND FUNDING. PARKER, G. Office of State Programs (Post 911117). September 1993. 34pp. 9310130049. 76743:190.

The first section of this document presents the results of a technical workshop on the process of regulations development and amendment sponsored by the Nuclear Regulatory Commission (NRC). This workshop focused on methods for reducing the time it takes to promulgate regulations to help those States that are having difficulty meeting the three-year deadline for adopting new NRC regulations. Workshop participants responded to six questions, reviewed the procedures used by the various States for revising and adopting changes to their regulations, and reviewed the time-flow charts used by various States. This workshop was designed to provide guidance to States that are promulgating and revising regulations. The second section of this document summarizes the proceedings of a technical workshop, also sponsored by the NRC, on funding radiation control programs that emphasized fee schedules and effective strategies for the 1990s. This workshop focused on determining the true costs of running a program, on setting realistic fees for the various categories of licenses, and on the most efficient methods for sending invoices, recording receipts, depositing money received, and issuing licenses. Workshop participants responded to seven questions; reviewed the methods various States use to determine true costs; reviewed the procedure that the various States use to produce invoices and licenses; reviewed the procedures that the States are required to abide by

when they receive money; and reviewed the method used by the NRC to determine the cost of its various programs.

NUREG/CP-0040: PROCEEDINGS OF WORKSHOP V: FLOW AND TRANSPORT THROUGH UNSATURATED FRACTURED ROCK -- RELATED TO HIGH-LEVEL RADIOACTIVE WASTE DISPOSAL. Held At Radisson Suite Hotel, Tucson, Arizona, January 7-10, 1991. EVANS, D.D.; NICHOLSON, T.J. Arizona, Univ. of, Tucson, AZ. June 1993. 250pp. 9307220313. 75744:119.

The "Workshop on Flow and Transport Through Unsaturated Fractured Rock Related to High-Level Radioactive Waste Disposal" was cosponsored by the NRC, the Center for Nuclear Waste Regulatory Analyses, and the University of Arizona (UAZ) and was held in Tucson, Arizona, on January 7-10, 1991. The focus of this workshop, similar to the earlier four (the first being in 1982), related to hydrogeologic technical issues associated with possible disposal of commercial high-level nuclear waste (HLW) in a geologic repository within an unsaturated fractured rock system which coincides with the UAZ field studies on HLW disposal. The presentations and discussions centered on flow and transport processes and conditions, relevant parameters, as well as state-of-the-art measurement techniques, and modeling capabilities. The workshop consisted of: four half-day technical meetings; a one-day field visit to the Apache Leap test site to review ongoing field studies that are examining site characterization techniques and developing data sets for model validation studies; and a final half-day session devoted to examining research needs related to modeling groundwater flow and radionuclide transport in unsaturated, fractured rock. These proceedings provide extended abstracts of the technical presentations and short summaries of the research group reports.

NUREG/CP-0129: PROCEEDINGS OF THE WORKSHOP ON PROGRAM FOR ELIMINATION OF REQUIREMENTS MARGINAL TO SAFETY. DEY, M. Advanced Reactors Branch (Post 910830). ARSENAULT, F.; PATTERSON, M.; et al. SCIENTECH, Inc. September 1993. 194pp. 9310120259. 76741:245.

These are the proceedings of the Public Workshop on the U.S. Nuclear Regulatory Commission's Program for Elimination of Requirements Marginal to Safety. The workshop was held at the Holiday Inn, Bethesda, on April 27 and 28, 1993. The purpose of the workshop was to provide an opportunity for public and industry input to the program. The workshop addressed the institutionalization of the program to review regulations with the purpose of eliminating those that are marginal. The objective is to avoid the dilution of safety efforts. One session was devoted to discussion of the framework for a performance-based regulatory approach. In addition, panelists and attendees discussed scope, schedules, and status of specific regulatory items: containment leakage testing requirements, fire protection requirements, requirements for environmental qualification of electrical equipment, requests for information under 10CFR50.54(f), requirements for combustible gas control systems, and quality assurance requirements.

NUREG/CP-0130 V01: PROCEEDINGS OF THE 22ND DOE/NRC NUCLEAR AIR CLEANING CONFERENCE. Sessions 1-8. Held In Denver, Colorado, August 24-27, 1992. FIRST, M.W. Harvard School of Public Health, Boston, MA. July 1993. 500pp. 9307270008. CONF-9020823. 75801:001.

This document contains the papers and the associated discussions of the 22nd DOE/NRC Nuclear Air Cleaning Conference. Major topics are: (1) advanced reactors; (2) reprocessing; (3) filter testing; (4) waste management; (5) instruments and sampling; (6) reactor accidents; (7) filters and filter performance; (8) adsorber testing and performance; (9) carbon testing; and (10) ventilation systems.

NUREG/CP-0130 V02: PROCEEDINGS OF THE 22ND DOE/NRC NUCLEAR AIR CLEANING CONFERENCE. Sessions 9-16. Held in Denver, Colorado, August 24-27, 1992. FIRST, M.W. Harvard School of Public Health, Boston, MA. July 1993. 500pp. 9307270050. CONF-9020823. 75803:070.
See NUREG/CP-0130, V01 abstract.

NUREG/CR-3469 V07: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA. KAURIN, D.G.; KHAN, T.A.; SULLIVAN, S.G.; et al. Brookhaven National Laboratory. July 1993. 106pp. 9308160137. BNL-NUREG-51708. 76120:001.

The ALARA Center at Brookhaven National Laboratory publishes a series of bibliographies of selected readings in radiation protection and ALARA in the continuing effort to collect and disseminate information on radiation dose reduction at nuclear power plants. This is volume 7 of the series. The abstracts in this bibliography were selected from proceedings of technical meetings and conferences, journals, research reports, and searches of the Energy Science and Technology database of the U.S. Department of Energy. The subject material of these abstracts relates to radiation protection and dose reduction, and ranges from use of robotics to operational health physics, to water chemistry. Material on the design, planning, and management of nuclear power stations is included, as well as information on decommissioning and safe storage efforts. Volume 7 contains 293 abstracts, an author index, and a subject index. The author index is specific for this volume. The subject index is cumulative and lists all abstract numbers from volumes 1 to 7. The numbers in boldface indicate the abstracts in this volume; the numbers not in boldface represent abstracts in previous volumes.

NUREG/CR-4667 V16: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, October 1992 - March 1993. CHUNG, H.M.; CHOPRA, O.K.; RUTHER, W.E.; et al. Argonne National Laboratory. September 1993. 67pp. 9310120274. ANL-93/27. 76742:069.

This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors (LWRs) during the six months from October 1992 to March 1993. Fatigue and EAC of piping, pressure vessels, and core components in LWRs are important concerns as extended reactor lifetimes are envisaged. Topics that have been investigated include (1) fatigue of low-alloy steel used in piping, steam generators, and reactor pressure vessels, (2) EAC of cast stainless steels (SSs), (3) radiation-induced segregation and irradiation-assisted stress corrosion cracking of Type 304 SS after accumulation of relatively high fluence, and (4) EAC of low-alloy steels. Fatigue tests were conducted on medium-sulfur-content A106-Gr B piping and A533-Gr B pressure vessel steels in simulated PWR water and in air. Additional crack growth data were obtained on fracture-mechanics specimens of cast austenitic SSs in the as-received and thermally aged conditions and chromium-nickel-plated A533-Gr B steel in simulated boiling-water reactor (BWR) water at 289 degrees C. The data were compared with predictions based on crack growth correlations for ferritic steels in oxygenated water and correlations for wrought austenitic SS in oxygenated water developed at ANL and rates in air from Section XI of the ASME Code. Microchemical and microstructural changes in high- and commercial-purity Type 304 SS specimens from control-blade absorber tubes and control-blade sheath from operating BWRs were studied by Auger electron spectroscopy and scanning electron microscopy.

NUREG/CR-4744 V07 N2: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS. Semiannual Report, April-September 1992. CHOPRA, O.K. Argonne National Laboratory. July 1993. 54pp. 9308160133. ANL-93/11. 76120:301.

This progress report summarizes work performed by Argonne National Laboratory on long-term thermal embrittlement of cast duplex stainless steels in LWR systems during the six months from April-September 1992. A procedure and correlations are presented for predicting Charpy-impact energy, tensile flow stress, fracture toughness J-R curves, tearing modulus and J(lc) of aged cast stainless steels from known material information. The "saturation" impact strength and fracture toughness of a specific cast stainless steel, i.e., the minimum value that would be achieved for the material after long-term service, is estimated from the chemical composition of the steel. Mechanical properties as a function of time and temperature of reactor service are estimated from impact energy and flow stress of the unaged material and the kinetics of embrittlement, which are also determined from chemical composition. The J(lc) values are determined from the estimated J-R curve and flow stress. Examples of estimating mechanical properties of cast stainless steel components during reactor service are presented. A common "lower-bound" J-R curve for cast stainless steels of unknown chemical composition is also defined for a given grade of steel, ferrite content, and temperature.

NUREG/CR-5404 V02: AUXILIARY FEEDWATER SYSTEM AGING STUDY. Phase I Follow-On Study. KUECK, J.D. Oak Ridge National Laboratory. July 1993. 39pp. 9308160268. ORNL-6566/V1. 76118:327.

This report documents the results of a Phase I follow-on study of the Auxiliary Feedwater (AFW) System that has been conducted for the U.S. Regulatory Commission's Nuclear Plant Aging Research Program. The Phase I study found a number of significant AFW System functions that are not being adequately tested by conventional test methods and some that are actually being degraded by conventional testing. Thus, it was decided that this follow-on study would focus on these testing omissions and equipment degradation. The deficiencies in current monitoring and operating practice are categorized and evaluated. Areas of component degradation caused by current practice are discussed. Recommendations are made for improved diagnostic methods and test procedures.

NUREG/CR-5754: BOILING-WATER REACTOR INTERNALS AGING DEGRADATION STUDY. Phase 1. LUK, K.H. Oak Ridge National Laboratory. September 1993. 56pp. 9310120363. ORNL/TM-11876. 76740:333.

This report documents the results of an aging assessment study for boiling water reactor (BWR) internals. Major stressors for BWR internals are related to unsteady hydrodynamic forces generated by the primary coolant flow in the reactor vessel. Welding and cold-working, dissolved oxygen and impurities in the coolant, applied loads and exposures to fast neutron fluxes are other important stressors. Based on results of a component failure information survey, stress corrosion cracking (SCC) and fatigue are identified as the two major aging-related degradation mechanisms for BWR internals. Significant reported failures include SCC in jet-pump hold-down beams, in-core neutron flux monitor dry tubes and core spray spargers. Fatigue failures were detected in feedwater spargers. The implementation of a plant Hydrogen Water Chemistry (HWC) program is considered as a promising method for controlling SCC problems in BWR. More operating data are needed to evaluate its effectiveness for internal components. Long-term fast neutron irradiation effects and high-cycle fatigue in a corrosive environment are uncertainty factors in the aging assessment process. BWR internals are examined by visual inspections and the method is access limited. The presence of a large water gap and an absence of ex-core neutron flux monitors may handicap the use of advanced inspection methods, such as neutron noise vibration measurements, for BWR.

6 Main Citations and Abstracts

NUREG/CR-5758 V03: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY. Annual Summary Of Program Performance Reports, CY 1992. FLEMING, T.; WESTRA, C.; FIELD, J., et al. Battelle Human Affairs Research Centers. July 1993. 140pp. 9308180128. PNL-8688. 76142:200.

This report summarizes the data from the semiannual reports on fitness-for-duty programs submitted to the NRC by 54 utilities for two reporting periods. January 1, 1992 to June 30, 1992, and from July 1, 1992 to December 31, 1992. During CY 1992, licensees reported that they conducted 266,551 tests for the presence of illegal drugs and alcohol. Of these tests, 1,818 (.68%) were confirmed positive. Positive test results varied by category of test and category of worker. The majority of positive test results (1,110) were obtained through pre-access testing. Of tests conducted on workers having access to the protected area, there were 461 positive tests from random testing, and 178 positive tests from for-cause testing. Followup testing of workers who had previously tested positive resulted in 69 positive tests. Positive test results also varied by category of worker. Overall, short-term and long-term contractor personnel had the highest rates of positive tests. Licensee employees had lower rates of positive test results.

NUREG/CR-5778 V03: NEW YORK/NEW JERSEY REGIONAL SEISMIC NETWORK. Final Report For April 1985 - September 1992. SEEGER, L.; JOHNSON, D.; ARMBRUSTER, J. Lamont-Doherty Geological Observatory. July 1993. 74pp. 9307270035. 75802:317.

For almost 20 years, Lamont-Doherty Earth Observatory has operated the primary network for monitoring earthquake activity in the New York State, northern New Jersey, and northwestern Vermont area, with support by both NRC and the USGS. The primary purpose of this research is directed toward the determination of local seismicity and the possible identification of associated geologic and tectonic features. From April 1985 to September 1992, the network recorded and located 346 regional earthquakes. Scientific activity, primarily in the form of after-shock monitoring, was concentrated upon a number of significant earthquakes: The Ardsley, NY; the Chardon, OH; the Ash-tabula, OH; and the Saguenay, Canada earthquakes; in addition to the Summit, NY event. These studies involved the deployment of portable seismographs in the epicentral areas. Many of these sequences were in northeastern North America, but outside the L-DEO seismic network and were not covered by other permanent networks. Spatial correlations between structures and earthquakes were found at a wide range of scales, and systematic searches of archival material were used to improve constraints on historic sources.

NUREG/CR-5782: PRESSURIZED THERMAL SHOCK PROBABILISTIC FRACTURE MECHANICS SENSITIVITY ANALYSIS FOR YANKEE ROWE REACTOR PRESSURE VESSEL. DICKSON, T.L.; CHEVERTON, R.D.; BRYSON, J.W., et al. Oak Ridge National Laboratory. August 1993. 114pp. 9309210205. ORNL/TM-11945. 76484:336.

The Nuclear Regulatory Commission (NRC) requested Oak Ridge National Laboratory (ORNL) to perform a pressurized-thermal-shock (PTS) probabilistic fracture mechanics (PFM) sensitivity analysis for the Yankee Rowe reactor pressure vessel, for the fluences corresponding to the end of operating cycle 22, using a specific small-break-loss-of-coolant transient as the loading condition. Regions of the vessel with distinguishing features were to be treated individually: upper axial weld, lower axial weld, circumferential weld, upper plate spot welds, upper plate regions between the spot welds, lower plate spot welds, and the lower plate regions between the spot welds. The fracture analysis methods used in the analysis of through-clad surface flaws were those contained in the established OCA-P computer code, which was developed during the Integrated Pressurized Thermal Shock (IPTS) Program. The NRC request specified that the OCA-P code be enhanced for this study to also calculate the conditional probabilities of failure for subclad flaws and embedded flaws. The results of this sensitivity analy-

sis provide the NRC with (1) data that could be used to assess the relative influence of a number of key input parameters in the Yankee Rowe PTS analysis and (2) data that can be used for readily determining the probability of vessel failure once a more accurate indication of vessel embrittlement becomes available. This report is designated as HSST report No. 117.

NUREG/CR-5817 V03 N2: NRC HIGH-LEVEL RADIOACTIVE WASTE RESEARCH AT CNWRA. July-December 1992. ABAROU, R.; AHOLA, M.P.; BACA, R.G., et al. Center for Nuclear Waste Regulatory Analyses. July 1993. 200pp. 9309090005. CNWRA 92-02S. 76393:001.

Progress from July 1 to December 31, 1992 on the nine NRC-sponsored research projects conducted at the Center for Nuclear Waste Regulatory Analyses is described. Ion-exchange experiments between clinoptilolite and aqueous solutions of Na(+) and Sr(2+) and three applications of reaction-path modeling are described in the Unsaturated Mass Transport (Geochemistry) project. Numerical simulation of a laboratory-scale non-isothermal two-phase flow is discussed in the Thermohydrology chapter. Methods for estimating rock joint roughness coefficient are the focus of the Seismic Rock Mechanics project for which the Tilt Test, Tse and Cruden's equations, and fractal-based equations were tested and found to be unsatisfactory. In the Integrated Waste Package Experiments chapter, investigations of pit initiation and repassivation potential for alloys 825 and C-22 and stainless steel 304L and 316L are described. Testing of the BIGFLOW computer code and visualization of fracture topology is the theme of the Stochastic Hydrology project. Preliminary analysis of field data from the Akrotiri site in Greece is developed in the Geochemical Analogs project. Mechanistic modeling of sorption using the MINTEQA2 code is investigated as part of the Sorption project. Adaptive gridding and "modified equations" methods for solving the flow and transport equations are described in the Performance Assessment chapter. Finally, the Volcanism chapter focuses on using nonhomogeneous Poisson processes for estimating probability of volcanic events at the potential repository site.

NUREG/CR-5829: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDANCE FOR THE DAVIS-BESSE NUCLEAR POWER PLANT. NICKOLAUS, J.R.; MOFFITT, N.E.; GORE, B.F., et al. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1993. 38pp. 9310120332. PNL-7905. 76742:139.

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

NUREG/CR-5833: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE H.B. ROBINSON NUCLEAR POWER PLANT. MOFFITT, N.E.; LLOYD, R.C.; GORE, B.F., et al. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1993. 37pp. 9308160303. PNL-7907. 76119:293.

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

NUREG/CR-5927 V01: EVALUATION OF A PERFORMANCE ASSESSMENT METHODOLOGY FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES. Evaluation Of Modeling Approaches. KOZAK, M.W.; OLAGUE, N.E.; RAO, R.R.; et al. Sandia National Laboratories. August 1993. 87pp. 9309210242. SAND91-2802. 76486. 118.

This report represents an update to our earlier reports on low-level waste performance assessment. This update addresses needed improvements and recommended approaches to the existing state of the art in modeling, treatment of uncertainty, and use of data. Greater attention is paid to developing an integrated approach to performance assessment than was done in earlier developments of the methodology. Furthermore, insights are being developed by participating in validation exercises, and by evaluating which validation data are needed to improve confidence in the methodology. It is emphasized that the performance assessment methodology update is a work in progress; the recommendations given here will form the general directions toward which the methodology is heading, but some of the specific approaches may continue to evolve as the research progresses.

NUREG/CR-5928: ISLOCA RESEARCH PROGRAM. Final Report. GALYEAN, W.J.; KELLY, D.L.; SCHROEDER, J.A.; et al. EG&G Idaho, Inc. July 1993. 107pp. 9308160131. EGG-2685. 76120. 108.

This report contains a compilation of information generated during the ISLOCA research program. Presented is a screening analysis and a procedures guide for performing an ISLOCA evaluation. This methodology has been distilled from past analyses performed for the U.S. Nuclear Regulatory Commission and documented in a series of NUREG/CR reports. The methodology comprises five distinct steps: (a) containment penetration screening; (b) interfaces for ISLOCA analysis; (c) mechanisms for failing the pressure boundary; (d) construction of event trees and estimation of rupture probabilities; and (e) quantification of the event tree. Included in the methodology are steps required for a detailed human reliability analysis. In addition, this report presents a BWR ISLOCA evaluation, a survey of PWR auxiliary building designs and identification of one design deemed most disadvantageous with respect to ISLOCA risk, and a PWR ISLOCA cost/benefit analysis.

NUREG/CR-5942: SEVERE ACCIDENT SOURCE TERM CHARACTERISTICS FOR SELECTED PEACH BOTTOM SEQUENCES PREDICTED BY THE MELCOR CODE. CARBAJO, J.J. Oak Ridge National Laboratory. September 1993. 345pp. 9310120226. ORNL/TM-12229. 76739. 001.

The purpose of this report is to compare in-containment source terms developed for NUREG-1159, which used the Source Term Code Package (STCP), with those generated by MELCOR to identify significant differences. For this comparison, two short-term depressurized station blackout sequences (with

a dry cavity and with a flooded cavity) and a Loss Of Coolant Accident (LOCA) concurrent with complete loss of the Emergency Core Cooling System (ECCS) were analyzed for the Peach Bottom Atomic Power Station (a BWR-4 with a Mark I containment). The results indicate that for the sequences analyzed, the two codes predict similar total in-containment release fractions for each of the element groups. However, the MELCOR/COBRH Package predicts significantly longer time for vessel failure and reduced energy of the released material for the station blackout sequences (when compared to the STCP results). MELCOR also calculated smaller releases into the environment than STCP for the station blackout sequences.

NUREG/CR-5943: SENSITIVITY ANALYSIS AND BENCHMARKING OF THE BLT LOW-LEVEL WASTE SOURCE TERM CODE. SUEN, C.J.; SULLIVAN, T.M. Brookhaven National Laboratory. July 1993. 81pp. 9307270012. BNL NUREG-52346. 75802. 236.

To evaluate the source term for low-level waste disposal, a comprehensive model had been developed and incorporated into a computer code, called BLT (Breach-Leach-Transport). Since the release of the original version, many new features and improvements had also been added to the Leach model of the code. This report consists of two different studies based on the new version of the BLT code: 1) a series of verification/sensitivity tests, and 2) benchmarking of the BLT code using field data. Based on the results of the verification/sensitivity tests, we concluded that the new version represents a significant improvement and it is capable of providing more realistic simulations of the leaching process. Benchmarking work was carried out to provide a reasonable level of confidence in the model predictions. In this study, the experimentally measured release curves for nitrate, technetium-99 and tritium from the saltstone lysimeters operated by Savannah River Laboratory were used. The model results are observed to be in general agreement with the experimental data, within the acceptable limits of uncertainty.

NUREG/CR-5944: A CHARACTERIZATION OF CHECK VALVE DEGRADATION AND FAILURE EXPERIENCE IN THE NUCLEAR POWER INDUSTRY. CASADA, D.A.; TODD, M.D. Oak Ridge National Laboratory. September 1993. 187pp. 9310120234. ORNL-6734. 76739. 342.

Check valve operating problems in recent years have resulted in significant operating transients, increased cost and decreased system availability. As a result, additional attention has been given to check valves by utilities (resulting in the formation of the Nuclear Industry Check Valve Group), as well as the U.S. Nuclear Regulatory Commission and the American Society of Mechanical Engineers Operation and Maintenance Committee. All these organizations have the fundamental goal of ensuring reliable operation of check valves. A key ingredient to an engineering-oriented reliability improvement effort is a thorough understanding of relevant historical experience. A detailed review of historical failure data, available through the Institute of Nuclear Power Operation's Nuclear Plant Reliability Data System, has been conducted. The focus of the review is on check valve failures that have involved significant degradation of the valve internal parts. A variety of parameters are considered, including size, age, system of service, method of failure discovery, the affected valve parts, attributed causes, and corrective actions.

NUREG/CR-5969: J AND CTOD ESTIMATION EQUATIONS FOR SHALLOW CRACKS IN SINGLE EDGE NOTCH BEND SPECIMENS. KIRK, M.T.; DODDS, R.H. Illinois, Univ. of, Urbana, IL. Navy, Dept. of July 1993. 28pp. 9308160121. UILU-ENG91-2013. 76121. 333.

Fracture toughness values determined using shallow cracked single edge notch bend, SE(B), specimens of structural thickness are useful for structural integrity assessments. Results from two dimensional plane strain finite-element analyses are used to develop J and CTOD estimation strategies appropriate for application to both shallow and deep crack SE(B) speci-

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mens. Crack depth to specimen width (a/W) ratios between 0.05 and 0.70 are modelled using Ramberg-Osgood strain hardening exponents (n) between 4 and 50. The estimation formulas divide J and CTOD into small scale yielding (SSY) and large scale yielding (LSY) components. For each case, the SSY component is determined by the linear elastic stress intensity factor, $K(I)$. The formulas differ in evaluation of the LSY component. The techniques considered include: estimating J or CTOD from plastic work based on load line displacement ($A(p1)/LLD$), from plastic work based on crack mouth opening displacement ($A(p1)/CMOD$), and from the plastic component of crack mouth opening displacement ($CMOD(p1)$). $A(p1)/CMOD$ provides the most accurate J estimation possible.

NUREG/CR-5970: APPROXIMATE TECHNIQUES FOR PREDICTING SIZE EFFECTS ON CLEAVAGE FRACTURE TOUGHNESS (JC). KIRK, M.T.; DODDS, R.H. Illinois, Univ. of, Urbana, IL. * Navy, Dept. of. July 1993. 38pp. 9308160116. ULLU-ENG92-2016. 76122:001.

This investigation examines the ability of an elastic T-stress analysis coupled with a modified boundary layer (MBL) solution to predict stresses ahead of a crack tip in a variety of planar geometries. The approximate stresses are used as input to estimate the effective driving force for cleavage fracture ($J(o)$) using the micromechanically based approach introduced by Dodds and Anderson. Finite element analyses for a wide variety of planar cracked geometries are conducted which have elastic biaxiality parameters (β) ranging from -0.99 (very low constraint) to +2.96 (very high constraint). The magnitude and sign of β indicate the rate at which crack-tip constraint changes with increasing applied load. All results pertain to a moderately strain hardening material (strain hardening exponent (n) of 10). These analyses suggest that β is an effective indicator of both the accuracy of T-MBL estimates of $J(o)$ and of applicability limits on evolving fracture analysis methodologies (i.e. T-MBL, J-Q, and $J/J(o)$). Specifically, when $|\beta| > 0.4$ these analyses show that the T-MBL approximation of $J(o)$ is accurate to within 20% of a detailed finite-element analysis. As "structural type" configurations, i.e. shallow cracks in tension, generally have $|\beta| > 0.4$, it appears that only an elastic analysis may be needed to determine reasonably accurate $J(o)$ values for structural conditions.

NUREG/CR-5971: CONTINUUM AND MICROMECHANICS TREATMENT OF CONSTRAINT IN FRACTURE. DODDS, R.H. Illinois, Univ. of, Urbana, IL. SHIH, C.F. Brown Univ., Providence, RI. ANDERSON, T.L. Texas A&M Univ., College Station, TX. July 1993. 47pp. 9308160294. ULLU-ENG92-2014. 76119:122.

Two complementary methodologies are described to quantify the effects of crack-tip stress triaxiality (constraint) on the macroscopic measures of elastic-plastic fracture toughness, J and CTOD. In the continuum mechanics methodology, two parameters, J and Q , suffice to characterize the full range of near-tip environments at the onset of fracture. A micromechanics methodology is described which predicts the toughness locus using crack-tip stress fields and critical J -values from a few fracture toughness tests. A robust micromechanics model for cleavage fracture has evolved from the observations of a strong, spatial self-similarity of crack-tip principal stresses under increased loading and across different fracture specimens. This report explores the fundamental concepts of the J-Q description of crack-tip fields, the fracture toughness locus and micromechanics approaches to predict the variability of macroscopic fracture toughness with constraint under elastic-plastic conditions. Computational results are presented for a surface cracked plate containing a 6:1 semi-elliptical, $a=t/4$ flaw subjected to remote uniaxial and biaxial tension.

NUREG/CR-5981: THE EFFECT OF ELECTRIC DISCHARGE MACHINED NOTCHES ON THE FRACTURE TOUGHNESS OF SEVERAL STRUCTURAL ALLOYS. JOYCE, J.A. U.S. Naval Academy, Annapolis, MD. LINK, R.E. Navy, Dept. of. September 1993. 90pp. 9310130032. 76744:290.

Recent computational studies of the stress and strain fields at the tip of very sharp notches have shown that the stress and strain fields are very weakly dependent on the initial geometry of the notch once the notch has been blunted to a radius that is 6 to 10 times the initial root radius. It follows that if the fracture toughness of a material is sufficiently high so that fracture initiation does not occur in a specimen until the crack-tip opening displacement (CTOD) reaches a value from 6 to 10 times the size of the initial notch tip diameter, then the fracture toughness will be independent of whether a fatigue crack or a machined notch served as the initial crack. In this experimental program the fracture toughness ($J(Ic)$ and J resistance ($J-R$) curve, and CTOD) for several structural alloys was measured using specimens with conventional fatigue cracks and with EDM machined notches. The results of this program have shown, in fact, that most structural materials do not achieve initiation CTOD values on the order of 6 to 10 times the radius of even the smallest EDM notch tip presently achievable. It is found furthermore that tougher materials do not seem to be less dependent on the type of notch tip present. Some materials are shown to be much more dependent on the type of initial notch tip used, but no simple pattern is found that relates this observed dependence to the material strength, toughness, or strain hardening rate.

NUREG/CR-5989: PERFORMANCE TESTING OF EXTREMITY DOSIMETERS--PILOT TEST. FOX, R.A.; HARTY, R.; MCDONALD, J.C. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1993. 51pp. 9308160263. PNL-8467. 76119:169.

A working group of the Health Physics Society Standards Committee (HPSSC) has issued a draft standard for extremity dosimeters. To determine the appropriateness of the proposed standard, Pacific Northwest Laboratory (PNL) has conducted three separate evaluations of the performance by processors of extremity dosimeters. The dosimeters were tested in each of the irradiation categories specified in the draft standard: high-energy photons (general and accident dosimetry), low-energy photons (general and accident dosimetry), beta particles, neutrons (first and second evaluations only), and a mixture category. In the first evaluation only about 60% of the processors met the draft standard's performance criteria for accuracy and precision. The second evaluation showed an overall improvement of 15% to 18%, but most processors were still unable to meet the performance criteria consistently in all irradiation categories. After these evaluations, PNL suggested several changes to the draft standard, including redefining some of the test categories and making the tolerance levels of the criteria more consistent with those in the standard for whole-body dosimetry. This report summarizes the third evaluation, which yielded an overall pass rate of 87%. Suggestions are given to render the draft standard generally more consistent with the criteria for whole-body dosimetry.

NUREG/CR-5993 V01: METHODS FOR DEPENDENCY ESTIMATION AND SYSTEM UNAVAILABILITY EVALUATION BASED ON FAILURE DATA STATISTICS. Summary Report. AZARM, M.A.; HSU, F.; MARTINEZ-GURIDI, et al. Brookhaven National Laboratory. July 1993. 37pp. 9307270039. BNL-NUREG-52362. 75803:032.

This report introduces a new perspective on the basic concept of dependent failures where the definition of dependency is based on clustering in failure times of similar components. This perspective has two significant implications: firstly, it relaxes the conventional assumption that dependent failures must be simultaneous and result from a severe shock; secondly, it allows the analyst to use all the failures in a time continuum to estimate the potential for multiple failures in a window of time (e.g., a test interval), therefore arriving at a more accurate value for system unavailability. In addition, the models developed here provide a method for plant-specific analysis of dependency, reflecting the plant-specific maintenance practices that reduce or

increase the contribution of dependent failures to system unavailability. The proposed methodology can be used for screening analysis of failure data to estimate the fraction of dependent failures among the failures. In addition, the proposed method can evaluate the impact of the observed dependency on system unavailability and plant risk. The formulations derived in this report have undergone various levels of validations through computer simulation studies and pilot applications. The pilot applications of these methodologies showed that the contribution of dependent failures of diesel generators in one plant was negligible, while in another plant, it was quite significant. It also showed that in the plant with significant contribution of dependency to Emergency Power System (EPS) unavailability, the contribution changed with time. Similar findings were reported for the Containment Fan Cooler breakers. Drawing such conclusions about system performance would not have been possible with any other reported dependency methodologies.

NUREG/CR-5993 V02: METHODS FOR DEPENDENCY ESTIMATION AND SYSTEM UNAVAILABILITY EVALUATION BASED ON FAILURE DATA STATISTICS. Detailed Description And Applications. AZARM, M.A.; HSU, F.; MARTINEZ-GURIDI, et al. Brookhaven National Laboratory. July 1993. 78pp. 9307270044. BNL-NUREG-52362. 75802:143.

This report introduces a new perspective on the basic concept of dependent failures where the definition of dependency is based on clustering in failure times of similar components. This perspective has two significant implications: firstly, it relaxes the conventional assumption that dependent failures must be simultaneous and result from a severe shock; secondly, it allows the analyst to use all the failures in a time continuum to estimate the potential for multiple failures in a window of time (e.g., a test interval), therefore arriving at a more accurate value for system unavailability. In addition, the models developed here provide a method for plant-specific analysis of dependency, reflecting the plant-specific maintenance practices that reduce or increase the contribution of dependent failures to system unavailability. The proposed methodology can be used for screening analysis of failure data to estimate the fraction of dependent failures among the failures. In addition, the proposed method can evaluate the impact of the observed dependency on system unavailability and plant risk. The formulations derived in this report have undergone various levels of validations through computer simulation studies and pilot applications. The pilot applications of these methodologies showed that the contribution of dependent failures of diesel generators in one plant was negligible, while in another plant, it was quite significant. It also showed that in the plant with significant contribution of dependency to Emergency Power System (EPS) unavailability, the contribution changed with time. Similar findings were reported for the Containment Fan Cooler breakers. Drawing such conclusions about system performance would not have been possible with any other reported dependency methodologies.

NUREG/CR-6015: STRUCTURAL AGING PROGRAM TECHNICAL PROGRESS FOR PERIOD JANUARY - DECEMBER 1992. NAUS, D.J.; OLAND, C.B. Oak Ridge National Laboratory. July 1993. 164pp. 9309030224. ORNL/TM-12342. 76328:021.

The Structural Aging (SAG) Program is conducted for the Nuclear Regulatory Commission (NRC) by the Oak Ridge National Laboratory (ORNL). The program has the overall objective of preparing an expandable handbook or report which will provide potential structural safety issues and acceptance criteria for use by the NRC in nuclear power plant evaluations of continued service. Initial focus of the program is on concrete and concrete-related materials which comprise safety-related (Category I) structures in light-water reactor facilities. The SAG Program is organized into four tasks: Task S.1-Program Management, Task S.2-Materials Property Data Base, Task S.3-Structural Component Assessment/Repair Technology, and Task S.4-Quantitative Methodology for Continued Service Determinations. In meeting the individual objectives of these tasks resources are drawn from ORNL with subcontract support from universities and other

research laboratories. This report provides an overview of principal developments in each of the four program tasks from January 1, 1992 to December 31, 1992. Planned activities under each of these tasks are also presented.

NUREG/CR-6021: A LITERATURE REVIEW OF COUPLED THERMAL-HYDROLOGIC-MECHANICAL-CHEMICAL PROCESSES PERTINENT TO THE PROPOSED HIGH-LEVEL WASTE REPOSITORY AT YUCCA MOUNTAIN. MANTEUFEL, R.D.; AHOLA, M.P.; TURNER, D.R.; et al. Center for Nuclear Waste Regulatory Analyses. July 1993. 240pp. 9308160184. CNWRA92-011. 76117:040.

A literature review has been conducted to determine the state of knowledge available in the modeling of coupled thermal (T), hydrologic (H), mechanical (M), and chemical (C) processes relevant to the design and/or performance of the proposed high-level waste (HLW) repository at Yucca Mountain, Nevada. The review focuses on identifying coupling mechanisms between individual processes and assessing their importance (i.e., if the coupling is either important, potentially important, or negligible). The significance of considering THMC-coupled processes lies in whether or not the processes impact the design and/or performance objectives of the repository. A review, such as reported here, is useful in identifying which coupled effects will be important, hence which coupled effects will need to be investigated by the U.S. Nuclear Regulatory Commission in order to assess the assumptions, data, analyses, and conclusions in the design and performance assessment of a geologic repository. Although this work stems from regulatory interest in the design of the geologic repository, it should be emphasized that the repository design implicitly considers all of the repository performance objectives, including those associated with the time after permanent closure. The scope of this review is considered beyond previous assessments in that it attempts (with the current state-of-knowledge) to determine which couplings are important, and identify which computer codes are currently available to model coupled processes.

NUREG/CR-6023: GENERIC ANALYSIS FOR EVALUATION OF LOW CHARPY UPPER-SHELF ENERGY EFFECTS ON SAFETY MARGINS AGAINST FRACTURE OF REACTOR PRESSURE VESSEL MATERIALS. DICKSON, T.L. Oak Ridge National Laboratory. July 1993. 82pp. 9308160110. ORNL/TM-12340. 76120:215.

Appendix G to 10 CFR Part 50 requires that reactor pressure vessel beltline materials maintain Charpy upper-shelf energies of no less than 50 ft-lb during the plant operating life, unless it is demonstrated in a manner approved by the Nuclear Regulatory Commission (NRC), that lower values of Charpy upper-shelf energy provide margins of safety against fracture equivalent to those in Appendix G to Section XI of the ASME Code. Analyses based on acceptance criteria and analysis methods adopted in the ASME Code Case N-512 are described herein. Additional information on material properties was provided by the NRC, Office of Nuclear Regulatory Research, Materials Engineering Branch. These cases, specified by the NRC, represent generic applications to boiling water reactor and pressurized water reactor vessels. This report is designated as HSST Report No. 140.

NUREG/CR-6026: THEORETICAL AND EXPERIMENTAL INVESTIGATION OF THERMOHYDROLOGIC PROCESSES IN A PARTIALLY SATURATED, FRACTURED POROUS MEDIUM. GREEN, R.T.; MANTEUFEL, R.D. Center for Nuclear Waste Regulatory Analyses. DODGE, F.T.; et al. Southwest Research Institute. July 1993. 225pp. 9309090010. CNWRA 92-006. 76390:064.

The performance of a geologic repository for high-level nuclear waste will be influenced to a large degree by thermohydrologic phenomena created by the emplacement of heat-generating radioactive waste. The importance of these phenomena is manifest in that they can greatly affect the movement of moisture and the resulting transport of radionuclides from the repository.

10 Main Citations and Abstracts

Thus, these phenomena must be well understood prior to a definitive assessment of a potential repository site. An investigation has been undertaken along three separate avenues of analysis: (1) laboratory experiments; (2) mathematical models; and (3) similitude analysis. A summary of accomplishments to date is as follows: (1) A review of the literature on the theory of heat and mass transfer in partially saturated porous medium; (2) A development of the governing conservation and constitutive equations; (3) A development of a dimensionless form of the governing equation; (4) A numerical study of the importance and sensitivity of flow to a set of dimensionless groups; (5) A survey and evaluation of experimental measurement techniques; (6) Execution of laboratory experiments of nonisothermal flow in a porous medium with a simulated fracture

NUREG/CR-6029 V01: AGING ASSESSMENT OF NUCLEAR AIR-TREATMENT SYSTEM HEPA FILTERS AND ADSORBERS. Phase I. WINEGARDNER, W. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1993. 48pp. 9309210238. PNL-8594. 76486:075.

A Phase I aging assessment of high-efficiency particulate air (HEPA) filters and activated carbon gas adsorption units (adsorbers) was performed by the Pacific Northwest Laboratory (PNL) as part of the U.S. Nuclear Regulatory Commission's (NRC) Nuclear Plant Aging Research (NPAR) Program. Information concerning design features; failure experience; aging mechanisms, effects, and stressors; and surveillance and monitoring methods for these key air-treatment system components was compiled. Over 1100 failures, or 12 percent of the filter installations, were reported as part of a Department of Energy (DOE) survey. Investigators from other national laboratories have suggested that aging effects could have contributed to over 80 percent of these failures. Tensile strength tests on aged filter media specimens indicated a decrease in strength. Filter aging mechanisms range from those associated with particle loading to reactions that alter properties of sealants and gaskets. Low radioiodine decontamination factors associated with the Three Mile Island (TMI) accident were attributed to the premature aging of the carbon in the adsorbers. Mechanisms that can lead to impaired adsorber performance include oxidation as well as the loss of potentially available active sites as a result of the adsorption of pollutants. Stressors include heat, moisture, radiation, and airborne particles and contaminants.

NUREG/CR-6034: OKLAHOMA SEISMIC NETWORK. Final Report. LUZA, K.V.; LAWSON, J.E. Oklahoma, Univ. of, Norman, OK. July 1993. 42pp. 9308160178. 76122:314.

The Nemaha uplift is composed of a number of crustal blocks typically 3 to 5 miles (5 to 8 km) wide and 5 to 20 miles (8 to 32 km) long. In Oklahoma, several discontinuous uplifts, such as the Oklahoma City, Lovell, Garber, and Crescent uplifts, occur along the main axis of the Nemaha uplift. A statewide network of 12 stations records seismological data in Oklahoma. Six semipermanent seismograph stations, four radiotelemetry stations, the Oklahoma Geophysical Observatory's seismograph station, and a borehole seismograph station at the Observatory comprise the Oklahoma Geological Survey's seismic network. From January 1, 1987, through December 31, 1992, 373 earthquakes were located by the Oklahoma seismic network. The distribution of the earthquakes by state is as follows: 315 in Oklahoma, 28 in Texas, 14 in Kansas, 8 in Arkansas, 7 in Missouri, and 1 in Nebraska. Of the 352 earthquakes, 23 were reported felt. The earthquake epicentral data produces at least three seismic trends. These trends are located in north-central Oklahoma, at the eastern margin of the Anadarko basin, and in the Arkoma basin-Ouachita Mountains area.

NUREG/CR-6043 V01: AGING ASSESSMENT OF ESSENTIAL HVAC CHILLERS USED IN NUCLEAR POWER PLANTS. Phase I. BLAHNIK, D.E.; KLEIN, R.F. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1993. 101pp. 9310120252. PNL-8614. 76741:146.

The Pacific Northwest Laboratory conducted a Phase I aging assessment of chillers used in the essential safety air-conditioning systems of nuclear power plants. Centrifugal chillers in the 75- to 750-ton refrigeration capacity range are the predominant type used. The chillers used, and air-conditioning systems served, vary in design from plant-to-plant. It is crucial to keep chiller internals very clean and to prevent the leakage of water, air, and other contaminants into the refrigerant containment system. Periodic operation on a weekly or monthly basis is necessary to remove moisture and non-condensable gases that gradually build up inside the chiller. This is especially desirable if a chiller is required to operate only as an emergency standby unit. The primary stressors and aging mechanisms that affect chillers include vibration, excessive temperatures and pressures, thermal cycling, chemical attack, and poor quality cooling water. Aging is accelerated by moisture, non-condensable gases (e.g., air), dirt, and other contamination within the refrigerant containment system, excessive start/stop cycling, and operating below the rated capacity. Aging is also accelerated by corrosion and fouling of the condenser and evaporator tubes. The principal cause of chiller failures is lack of adequate monitoring. Lack of performing scheduled maintenance and human errors also contribute to failures.

NUREG/CR-6048: PRESSURIZED-WATER REACTOR INTERNALS AGING DEGRADATION STUDY. Phase 1. LUK, K.H. Oak Ridge National Laboratory. September 1993. 66pp. 9310120328. ORNL/TM-12371. 76740:266.

This report documents the results of a Phase 1 study on the effects of aging degradations on pressurized-water reactor (PWR) internals. Primary stressors for internals are generated by the primary coolant flow in the reactor vessel, and they include unsteady hydrodynamic forces and pump-generated pressure pulsations. Other stressors are applied loads, manufacturing processes, impurities in the coolant and exposures to fast neutron fluxes. A survey of reported aging-related failure information indicates that fatigue, stress corrosion cracking (SCC) and mechanical wear are the three major aging-related degradation mechanisms for PWR internals. Significant reported failures include thermal shield flow-induced vibration problems, SCC in guide tube support pins and core support structure bolts, fatigue-induced core baffle water-jet impingement problems and excess wear in flux thimbles. Many of the reported problems have been resolved by accepted engineering practices. Uncertainties remain in the assessment of long-term neutron irradiation effects and environmental factors in high-cycle fatigue failures. Reactor internals are examined by visual inspections and the technique is access limited. Improved inspection methods, especially one with an early failure detection capability, can enhance the safety and efficiency of reactor operations.

NUREG/CR-6049: PIPING BENCHMARK PROBLEMS FOR THE GENERAL ELECTRIC ADVANCED BOILING WATER REACTOR. BEZLER, P.; DEGRASSI, G.; BRAVERMAN, J.; et al. Brookhaven National Laboratory. August 1993. 178pp. 9309030213. BNL-NUREG-52377. 76327:127.

To satisfy the need for verification of the computer programs and modeling techniques that will be used to perform the final piping analyses for an advanced boiling water reactor standard design, three benchmark problems were developed. The problems are representative piping systems subjected to representative dynamic loads with solutions developed using the methods being proposed for analysis for the advanced reactor standard design. It will be required that the combined license holders demonstrate that their solutions to these problems are in agreement with the benchmark problem set.

NUREG/CR-6050: RADIATION EXPOSURE MONITORING AND INFORMATION TRANSMITTAL (REMIT) SYSTEM. User's Manual. CALE, R.; CLARK, T.; DIXSON, F.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). June 1993. 300pp. 9307220201. SAIC-93/13'0-01. 75742:001.

The Radiation Exposure Monitoring and Information Transmittal (REMIT) system is designed to assist U.S. Nuclear Regulatory Commission (NRC) licensees in meeting the reporting requirements of the revised 10CFR20 and in agreement with the guidance contained in R.G. 8.7, Rev. 1, "Instructions for Recording and Reporting Occupational Exposure Data." REMIT is a personal computer (PC) based menu driven system that facilitates the manipulation of data base files to record and report radiation exposure information. REMIT is designed to be user-friendly and contains the full text of R.G. 8.7, Rev. 1, on-line as well as context-sensitive help throughout the program. The user can enter data directly from NRC Forms 4 or 5, REMIT allows the user to view the individual's exposure in relation to regulatory or administrative limits and alerts the user to exposures in excess of these limits. The system also provides for the calculation and summation of dose from intakes and the determination of the dose to the maximally exposed extremity for the monitoring year. REMIT can produce NRC Forms 4 and 5 in paper and electronic format and can import/export data from ASCII and data base files.

NUREG/CR-6052: METHODOLOGY FOR RELIABILITY BASED CONDITION ASSESSMENT. Application To Concrete Structures In Nuclear Plants. MORI, Y.; ELLINGWOOD, B. Johns Hopkins Univ., Baltimore, MD. * Oak Ridge National Laboratory. August 1993. 164pp. 9308200285. ORNL SUB93-SD684. 76159:048.

Structures in nuclear power plants may be exposed to aggressive environmental effects that cause their strength to decrease over an extended period of service. A major concern in evaluating the continued service of such structures is to ensure that in their current condition they are able to withstand future extreme load events during the intended service life with a level of reliability sufficient for public safety. This report describes a methodology to facilitate quantitative assessments of current and future structural reliability and performance of structures in nuclear power plants. This methodology takes into account the nature of past and future loads, and randomness in strength and in degradation resulting from environmental factors. An adaptive Monte Carlo simulation procedure is used to evaluate time-dependent system reliability. The time-dependent reliability is sensitive to the time-varying load characteristics and to the choice of initial strength and strength degradation models but not to correlation in component strengths within a system. Inspection/maintenance strategies are identified that minimize the expected future costs of keeping the failure probability of a structure at or below an established target failure probability during its anticipated service period.

NUREG/CR-6056: A FRAMEWORK FOR THE ASSESSMENT OF SEVERE ACCIDENT MANAGEMENT STRATEGIES. KASTENBERG, W.E.; APOSTOLAKIS, G.; DHIR, V.K.; et al. California, Univ. of, Los Angeles, CA. September 1993. 350pp. 9310130038. 76743:231.

Severe accident management can be defined as the use of existing and/or alternative resources, systems and actors to prevent or mitigate a core-melt accident. For each accident sequence and each combination of severe accident management strategies, there may be several options available to the operator, and each involves phenomenological and operational considerations regarding uncertainty. Operational uncertainties include operator, system and instrumentation behavior during an accident. A framework based on decision trees and influence diagrams has been developed which incorporates such criteria as feasibility, effectiveness, and adverse effects, for evaluating potential severe accident management strategies. The framework is also capable of propagating both data and model uncertainty. It is applied to several potential strategies including PWR

cavity flooding, BWR drywell flooding, PWR depressurization, and PWR feed and bleed.

NUREG/CR-6058: VIRGINIA REGIONAL SEISMIC NETWORK. Final Report (1986 -1992). BOLLINGER, G.A.; SIBOL, M.S.; CHAPMAN, M.C.; et al. Virginia Polytechnic Institute & State Univ., Blacksburg, VA. July 1993. 115pp. 9308160175. 76122:084.

In 1986, the Virginia Regional Seismic Network was one of the few fully calibrated digital seismic networks in the United States. Continued operation has resulted in the archival of signals from 2000+ local, regional and teleseismic sources. Seismotectonic studies of the central Virginia seismic zone showed the activity in the western part to be related to a large antiformal structure while seismicity in the eastern portion is associated spatially with dike swarms. The eastern Tennessee seismic zone extends over a 300x50 km area and is the result of a compressive stress field acting at the intersection between two large crustal blocks. Hydroseismicity, which proposes a significant role for meteoric water in intraplate seismogenesis, found support in the observation of common cyclicities between streamflow and earthquake strain data. Seismic hazard studies have provided the following results: 1) Damage areas in the eastern United States are three to five times larger than those observed in the west; 2) Judged solely on the basis of cataloged earthquake recurrence rates, the next major shock in the southeast region will probably occur outside the Charleston, South Carolina, area; and 3) Investigations yielded necessary hazard parameters (for example, maximum magnitudes) for several sites in the southeast. Basic to these investigations was the development and maintenance of several seismological data bases.

NUREG/CR-6060: HYDROGEN MIXING STUDIES (HMS) ASSESSMENT MANUAL. LAM, K.L.; WILSON, T.L. Los Alamos National Laboratory. TRAVIS, J.R. Science Applications International Corp. (formerly Science Applications, Inc.). June 1993. 94pp. 9308160049. LA-12593-M. 76121:148.

This report documents some calculations performed to assess the Hydrogen Mixing Studies (HMS) code. Results are presented first for some analytical test problems, including laminar flow and mass diffusion. The von Karman vortex street problem and the Sandia FLAME Facility and Heiss Dampf Reaktor (HDR) containment facility test problems are then discussed. For the analytical problems, the code gave results that agree exceptionally well with the analytical solutions. Calculations for the von Karman vortex street problem were performed at selected Reynolds numbers for several obstacle types. The computed flow patterns agree well with experimental observations--specifically the occurrence of a vortex street (double row of vortices) above a critical Reynolds number. Calculations for the von Karman vortex street problem were performed at selected Reynolds numbers for several obstacle types. The computed flow patterns agree well with experimental observations--specifically the occurrence of a vortex street (double row of vortices) above a critical Reynolds number. The last assessment problem involves modeling the experiment T31.5. The experiment was carried out in the HDR containment building which is a large, multi-compartment facility (11 300 m³) free volume in 72 compartments). In the experiment, a steam-water mixture was first injected into the containment to simulate a large-break blow-down of a pressure vessel, and then superheated steam was injected that was followed by a release of helium-hydrogen light gas. The calculated results (pressure, temperature, and gas concentrations) agree reasonably well with the experimental data.

NUREG/CR-6065: SYSTEMS ANALYSIS OF THE CANDU 3 REACTOR. WOLFGONG, J.R.; LINN, M.A.; WRIGHT, A.L.; et al. Oak Ridge National Laboratory. July 1993. 334pp. 9308160298. ORNL/TM-12396. 76118:001.

12 Main Citations and Abstracts

This report presents the results of a systems failure analysis study of the CANDU 3 reactor design; the study was performed for the U.S. Nuclear Regulatory Commission. As part of the study a review of the CANDU 3 design documentation was performed, a plant assessment methodology was developed, representative plant initiating events were identified for detailed analysis, and a plant assessment was performed. The results of the plant assessment included classification of the CANDU 3 event sequences that were analyzed, determination of CANDU 3 systems that are "significant to safety," and identification of key operator actions for the analyzed events.

NUREG/CR-6072: EXPERIMENTAL STUDY ON THE COMBUSTION BEHAVIOR OF HYDROGEN-AIR MIXTURES WITH TURBULENT JET IGNITION AT LARGE SCALE. DOROFEEV, S.B.; BEZMELNITSIN, A.; EFIMENKO, A.A.; et al. Russia. June 1993. 83pp. 9308160272. RRCKI-80-05/3. 76117:272.

This report describes research carried out in the KOPER facility on spontaneous detonation ignition in hydrogen-air mixtures by turbulent jet ignition. The KOPER facility is a semi-confined volume of 47.7 m(3) and consists of a steel canyon and a robust frame placed above it. The frame sides are sealed with thin polyethylene sheet. A "jet" chamber of 0.55 m(3), located on the bottom of the canyon was used to produce a jet of hot gases, which was vented into the hydrogen-air mixture. The effects of three variables were investigated: hydrogen concentration (18-30% vol.); jet orifice diameter (100-400 mm); and the composition of combustion products in the turbulent jet (by varying the hydrogen mole fraction in the "jet"-chamber from 25 to 50% vol.). The possibility of initiation of turbulent combustion and local detonation was demonstrated. Local detonation develops after a delay of 10-25 ms from ignition. For spontaneous detonation initiation, the minimum hydrogen concentration is within the range of 20 to 25% vol., and the minimum jet orifice diameter lies in the range of 100 to 200 mm for the KOPER facility. A minimum ratio of turbulent jet size L and mixture detonation cell λ , $L/\lambda = 12-13$ is required for detonation initiation which is supported by other type of turbulent jet initiation experiments (closed volume and continuous venting) and by theoretical analysis.

NUREG/CR-6073: LYSIMETER LITERATURE REVIEW. ROGERS, R.D.; MCCONNELL, J.W. EG&G Idaho, Inc. August 1993. 75pp. 9309210201. EGG-2706. 76484:257.

Many reports have been published concerning the use of lysimeters to obtain data on the performance of buried radioactive waste. This document presents a review of some of those reports. This review includes lysimeter studies using radioactive waste forms at Savannah River Site, Hanford Site, Argonne National Laboratory, and Oak Ridge National Laboratory; radionuclide tracer studies at Whiteshell Nuclear Research Establishment and Los Alamos National Laboratory; and water movement studies at the Nuclear Regulatory Commission's Beltsville, Maryland site, at the Hanford Site, and at New Mexico State University. The tests, results, and conclusions of each report are summarized, and conclusions concerning lysimeter technology are presented from an overall analysis of the literature.

NUREG/CR-6078: ANALYSIS OF CRACK INITIATION AND GROWTH IN THE HIGH LEVEL VIBRATION TEST AT TADOTSU. KASSIR, M.K.; PARK, Y.J.; HOFMAYER, C.H.; et al. Brookhaven National Laboratory. August 1993. 82pp. 9309210185. BNL-NUREG-52383. 76484:075.

The High Level Vibration Test data are used to assess the accuracy and usefulness of current engineering methodologies for predicting crack initiation and growth in a cast stainless steel pipe elbow under complex, large amplitude loading. The data were obtained by testing at room temperature a large scale modified model of one loop of a PWR primary coolant system at the Tadotsu Engineering Laboratory in Japan. Fatigue crack initiation time is reasonably predicted by applying a modified local strain approach (Coffin-Manson-Goodman equation) in conjunction with Miner's rule of cumulative damage. Three fracture me-

chanics methodologies are applied to investigate the crack growth behavior observed in the hot leg of the model. These are: the ΔK methodology (Paris law), ΔJ concepts and a recently developed limit load stress-range criterion. The report includes a discussion on the pros and cons of the analysis involved in each of the methods, the role played by the key parameters influencing the formulation and a comparison of the results with the actual crack growth behavior observed in the vibration test program. Some conclusions and recommendations for improvement of the methodologies are also provided.

NUREG/CR-6079: SEISMOLOGICAL INVESTIGATION OF EARTHQUAKES IN THE NEW MADRID SEISMIC ZONE. Final Report. September 1986 - December 1992. HERRMANN, R.B.; NGUYEN, B. St. Louis Univ., St. Louis, MO. August 1993. 75pp. 9309030148. 76325:084.

Earthquake activity in the New Madrid Seismic Zone had been monitored by regional seismic networks since 1975. During this time period, over 3700 earthquakes have been located within the region bounded by latitudes 35 degrees-39 degrees N and longitudes 87 degrees-92 degrees W. Most of these earthquakes occur within a 1.5 degree x 2 degree zone centered on the Missouri Bootheel. Source parameters of larger earthquakes in the zone and in eastern North America are determined using surface-wave spectral amplitudes and broadband waveforms for the purpose of determining the focal mechanism, source depth and seismic moment. Waveform modeling of broadband data is shown to be a powerful tool in defining these source parameters when used complementary with regional seismic network data, and in addition, in verifying the correctness of previously published focal mechanism solutions.

NUREG/CR-6081: ENHANCED REMOVAL OF RADIOACTIVE PARTICLES BY FLUOROCARBON SURFACTANT SOLUTIONS. KAISER, R.; HARLING, O.K. Affiliation Not Assigned. August 1993. 97pp. 9309210192. 76484:156.

The proposed research addressed the application of ESI's particle removal process to the non-destructive decontamination of nuclear equipment. The cleaning medium used in this process is a solution of a high molecular weight fluorocarbon surfactant in an inert perfluorinated liquid which results in enhanced particle removal. The perfluorinated liquids of interest, which are recycled in the process, are nontoxic, nonflammable, and environmentally compatible, and do not present a hazard to the ozone layer. The information obtained in the Phase I program indicated that the proposed ESI process is technically effective and economically attractive. The fluorocarbon surfactant solutions used as working media in the ESI process survived exposure of up to 10 Mrad doses of gamma rays, and are considered sufficiently radiation resistant for the proposed process. Ultrasonic cleaning in perfluorinated surfactant solutions was found to be an effective method of removing radioactive iron (Fe 59) oxide particles from contaminated test pieces. Radioactive particles suspended in the process liquids could be quantitatively removed by filtration through a 0.1 μ m membrane filter. Projected economics indicate a pre-tax pay back time of 1 month for a commercial scale system.

NUREG/CR-6082: DATA COMMUNICATIONS. PRECKSHOT, G.G. Lawrence Livermore National Laboratory. August 1993. 96pp. 9309030171. UCRL ID-114567. 76326:219.

The purpose of this paper is to recommend regulatory guidance for reviewers examining computer communication systems used in nuclear power plants. The recommendations cover three areas important to these communications systems: system design, communication protocols, and communication media. The first area, system design, considers three aspects of system design--questions about architecture, specific risky design elements or omissions to look for in designs being reviewed, and recommendations for multiplexed data communication systems used in safety systems. The second area reviews pertinent aspects of communication protocol design and makes

recommendations for newly designed protocols or the selection of existing protocols for safety system, information display, and non-safety control system use. The third area covers communication media selection, which differs significantly from traditional wire and cable. The recommendations for communication media extend or enhance the concerns of published IEEE standards about three subjects: data rate, imported hazards and maintainability.

NUREG/CR-6083: REVIEWING REAL-TIME PERFORMANCE OF NUCLEAR REACTOR SAFETY SYSTEMS. PRECKSHOT, G.G. Lawrence Livermore National Laboratory. August 1993. 88pp. 9309030112. UCRL-ID-114565. 76325:001.

The purpose of this paper is to recommend regulatory guidance for reviewers examining real-time performance of computer-based safety systems used in nuclear power plants. Three areas of guidance are covered in this report. The first area covers how to determine if, when, and what prototypes should be required of developers to make a convincing demonstration that specific problems have been solved or that performance goals have been met. The second area has recommendations for timing analyses that will prove that the real-time system will meet its safety-imposed deadlines. The third area has descriptions of means for assessing expected or actual real-time performance before, during, and after development is completed. To ensure that the delivered real-time software product meets performance goals, the paper recommends certain types of code-execution and communications scheduling. Technical background is provided in the appendix on methods of timing analysis, scheduling real-time computations, prototyping, real-time software development approaches, modeling and measurement, and real-time operating systems.

NUREG/CR-6085: UNITED STATES SEISMOGRAPHIC NETWORK. BULAND, R. Interior, Dept. of, Geological Survey. September 1993. 83pp. 9310120339. 76741:028.

The concept of a United States National Seismograph Network (USNSN) dates back nearly 30 years. The idea was revived several times over the decades, but never funded. For example, a national network was proposed and discussed at great length in the so called Bolt Report (U.S. Earthquake Observatories: Recommendations for a New National Network, National Academy Press, Washington, D.C., 1980, 122 pp). From the beginning, a national network was viewed as augmenting and complementing the relatively dense, predominantly short-period vertical coverage of selected areas provided by the Regional Seismograph Networks (RSN's) with a sparse, well-distributed network of three-component, observatory quality, permanent stations. The opportunity finally to begin developing a national network arose in 1986 with discussions between the U.S. Geological Survey (USGS) and the Nuclear Regulatory Commission (NRC). Under the agreement signed in 1987, the NRC has provided \$5 M in new funding for capital equipment (over the period 1987-1992) and the USGS has provided personnel and facilities to develop, deploy, and operate the network. Because the NRC funding was earmarked for the eastern United States, new USNSN station deployments are mostly east of 105 degrees West longitude while the network in the western United States is mostly made up of cooperating stations (stations meeting USNSN design goals, but deployed and operated by other institutions which provide a logical extension to the USNSN).

NUREG/CR-6090: THE PROGRAMMABLE LOGIC CONTROLLER AND ITS APPLICATION IN NUCLEAR REACTOR SYSTEMS. PALOMAR, J.; WYMAN, R. Lawrence Livermore National Laboratory. September 1993. 96pp. 9310120250. UCRL-ID-112900. 76740:170.

This document provides recommendations to guide reviewers in the application of Programmable Logic Controllers (PLCs) to the control, monitoring and protection of nuclear reactors. The first topics addressed are system-level design issues, specifically including safety. The document then discusses concerns

about the PLC manufacturing organization and the protection system engineering organization. Supplementing this document are two appendices. Appendix A summarizes PLC characteristics. Specifically addressed are those characteristics that make the PLC more suitable for emergency shutdown systems than other electrical/electronic-based systems, as well as characteristics that improve reliability of a system. Also covered are PLC characteristics that may create an unsafe operating environment. Appendix B provides an overview of the use of programmable logic controllers in emergency shutdown systems. The intent is to familiarize the reader with the design, development, test, and maintenance phases of applying a PLC to an ESD system. Each phase is described in detail and information pertinent to the application of a PLC is pointed out.

NUREG/GR-0005 V02 P1: RISK-BASED INSPECTION-DEVELOPMENT OF GUIDELINES. Light Water Reactor (LWR) Nuclear Power Plant Components. * American Society of Mechanical Engineers. July 1993. 173pp. 9308200281. CRTD-VOL.20-2. 76159:209.

Effective inservice inspection programs can play a significant role in minimizing equipment and structural failures. Most of the current inservice inspection programs for light water reactor (LWR) nuclear power plant components are based on experience and engineers' qualitative judgment. These programs include only an implicit consideration of risk, which combines the probability of failure of a component under its operation and loading conditions and the consequences of such failure, if it occurs. This document recommends appropriate methods for establishing a risk-based inspection program for LWR nuclear power plant components. The process has been built from a general methodology (Volume 1) and has been expanded to involve five major steps: defining the system; evaluating qualitative risk assessment results; using this and information from plant probabilistic risk assessments to perform a quantitative risk analysis; selecting target failure probabilities; and developing an inspection program for components using economic decision analysis and structural reliability assessment methods. Included: extensive bibliography. Companion Volume 2 - Part 2 document will recommend risk-based inspection program for consideration by Section XI of the ASME Boiler and Pressure Vessel Code.

NUREG/IA-0085: ASSESSMENT OF FULL POWER TURBINE TRIP START-UP TEST FOR C. TRILLO I WITH RELAP5/MOD2. LOZANO, M.F.; MORENO, P.; DE LA CAL, C.; et al. Spain, Govt. of. July 1993. 73pp. 9308160158. ICSP-TR-TTRIP-R. 76119:218.

C. Trillo I has developed a model of the plant with RELAP5/MOD2/36.04. This model will be validated against a selected set of start-up tests. One of the transients selected to that aim is the turbine trip, which presents very specific characteristics that make it significantly different from the same transient in other PWRs of different design, the main difference being that the reactor is not tripped: a reduction in primary power is carried out instead. Pre-test calculations were done of the Turbine Trip Test and compared against the actual test. Minor problems in the first model, specially in the Control and Limitation Systems, were identified and post-test calculations had been carried out. The results show a good agreement with data for all the compared variables.

NUREG/IA-0091: ASSESSMENT OF RELAP5/MOD2 AGAINST A NATURAL CIRCULATION EXPERIMENT IN NUCLEAR POWER PLANT BORSSELE. WINTERS, L. Netherlands, Govt. of. July 1993. 66pp. 9308160307. ECN-89-91. 76122:247.

As part of the ICAP (International Code Assessment and Applications Program) agreement between ECN (Netherlands Energy Research Foundation) and USNRC, ECN has performed a number of assessment calculations for the thermohydraulic system analysis code RELAP5/MOD2/36.05. This document describes the assessment of this computer program versus a

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natural circulation experiment as conducted at the Borssele Nuclear Power Plant. The results of this comparison show that the code RELAP5/MOD2 predicts well the natural circulation behavior of Nuclear Power Plant Borssele. The work has been sponsored by the Dutch Licensing Authority and ECN.

NUREG/IA-0096: NUMERICS AND IMPLEMENTATION OF THE UK HORIZONTAL STRATIFICATION ENTRAINMENT OFF-TAKE MODEL INTO RELAP5/MOD3. BRYCE, W.M. Winfrith Technology Centre (United Kingdom). June 1993. 47pp. 9307220307. AEA-TRS-1050. 75743:253.

This report presents the numerics and implementation details to add the same improved discharge quality correlations into RELAP5/MOD3. In the light of experience with the modified RELAP5/MOD2 code, some of the numerics have been slightly changed for RELAP5/MOD3. The description is quite detailed in order to facilitate change by some future code developer. A simple test calculation was performed to confirm the coding of the correlations implemented in RELAP5/MOD3.

NUREG/IA-0100: ASSESSMENT OF CCFL MODEL OF RELAP5/MOD3 AGAINST SIMPLE VERTICAL TUBES AND ROD BUNDLE TESTS. CHO, S.; ARNE, N. Korea Electric Power Corp. CHUNG, B.-D.; et al. Korea Institute of Nuclear Safety. June 1993. 122pp. 9307220299. 75744:001.

The CCFL model used in RELAP5/MOD3 version 5m5 has been assessed against simple vertical tubes and rod bundle tests performed at a facility of Korea Atomic Energy Research Institute. The effect of changes in tube diameter and nodalization of tube section were investigated. The roles of interfacial drags on the flooding characteristics are discussed. Differences between the calculation and the experiment are also discussed. A comparison between model assessment results and the test data showed that the calculated value lay well on the experimental flooding curve specified by user, but the pressure jump before onset of flooding was not calculated.

NUREG/IA-0103: ASSESSMENT OF BETHSY TEST 9.1.B USING RELAP5/MOD3. LEE, S.; CHUNG, B.-D.; KIM, H.-J. Korea Institute of Nuclear Safety. June 1993. 250pp. 9307220272. 75743:001.

2" cold leg break test 9.1.b, conducted at the BETHSY facility was analyzed using the RELAP5/MOD3 Version 5m5 code. The test 9.1.b was conducted with the main objective being the investigation of the thermal-hydraulic mechanisms responsible for the large core uncover and fuel heat-up, requiring the implementation of an ultimate procedure. The present analysis demonstrates the code's capability to predict, with sufficient accuracy, the main phenomena occurring in the depressurization transient, both from a qualitative and quantitative point of view. Nevertheless, several differences regarding the evolution of phenomena and affecting the timing order have to be pointed out in the base calculation. Three calculations were carried out to study the sensitivity to change of the nodalization in the components of the loop seal cross-over legs, and of the auxiliary feedwater control logics, and of the break discharge coefficient.

NUREG/IA-0107: ASSESSMENT OF RELAP5/MOD2 AGAINST A LOAD REJECTION FROM 100% TO 50% POWER IN THE VANDELLOS II NUCLEAR POWER PLANT. LLOPIS, C.; MENDIZABAL, R.; PEREZ, J. Spain, Govt. of. June 1993. 53pp. 9307220264. ICSP-V2-R50-R. 75742:241.

An assessment of RELAP5/MOD2 cycle 36.04 against a load rejection from 100% to 50% power in Vandellos II NPP (Spain) is presented. The work is inscribed in the framework of the Spanish contribution to ICAP Project. The model used in the simulation consists of a single loop, a steam generator and a steam line up to the steam header, all of them enlarged on a scale of 3:1; and full-scaled reactor vessel and pressurizer. The results of the calculations have been in reasonable agreement with plant measurements.

NUREG/IA-0112: ASSESSMENT OF RELAP5/MOD2 AGAINST ECN-REFLOOD EXPERIMENTS. WOODSTRA, A.; VANDEBOGAARD, J.; STOOP, P.M. Netherlands, Govt. of. July 1993. 87pp. 9308160054. ECN-C-92-008. 76121:243.

As part of the ICAP (International Code Assessment and Applications Program) agreement between ECN (Netherlands Energy Research Foundation) and USNRC, ECN has performed a number of assessment calculations with the computer program RELAP5. This report describes the results as obtained by ECN from the assessment of the thermohydraulic computer program RELAP5/MOD2/CY 36.05 versus a series of reflood experiments in a bundle geometry. A total number of seven selected experiments have been analyzed, from the reflood experimental program as previously conducted by ECN under contract of the Commission of the European Communities (CEC). In this document, the results of the analyses are presented and a comparison with the experimental data is provided.

NUREG/IA-0113: PRELIMINARY ASSESSMENT OF PWR STEAM GENERATOR MODELLING IN RELAP5/MOD3. PREECE, R.J.; PUTNEY, J.M. National Power (United Kingdom). July 1993. 26pp. 9309090021. 76381:154.

A preliminary assessment of Steam Generator (SG) modelling in the PWR thermal-hydraulic code RELAP5/MOD3 is presented. The study is based on calculations against a series of steady-state commissioning tests carried out on the Wolf Creek PWR over a range of load conditions. Data from the tests are used to assess the modelling of primary to secondary side heat transfer and, in particular, to examine the effect of reverting to the standard form of the Chen heat transfer correlation in place of the modified form applied in RELAP5/MOD2. Comparisons between the two versions of the code are also used to show how the new interphase drag model in RELAP5/MOD3 affects the calculation of SG liquid inventory and the void fraction profile in the riser.

NUREG/IA-0126: 2D/3D PROGRAM WORK SUMMARY REPORT. DAMERELL, P.S.; SIMONS, J.W. MPR Associates, Inc. *; et al. Japan Atomic Energy Research Institute. June 1993. 400pp. 9307220220. GRS-100. 75745:042.

The 2D/3D Program was carried out by Germany, Japan and the United States to investigate the thermal-hydraulics of a PWR large-break LOCA. A contributory approach was utilized in which each country contributed significant effort to the program and all three countries shared the research results. Germany constructed and operated the Upper Plenum Test Facility (UPTF), and Japan constructed and operated the Cylindrical Core Test Facility (CCTF) and the Slab Core Test Facility (SCTF). The US contribution consisted of provision of advanced instrumentation to each of the three test facilities, and assessment of the TRAC computer code against the test results. Evaluations of the test results were carried out in all three countries. This report summarizes the 2D/3D Program in terms of the contributing efforts of the participants.

NUREG/IA-0127: REACTOR SAFETY ISSUES RESOLVED BY THE 2D/3D PROGRAM. DAMERELL, P.S.; SIMONS, J.W. MPR Associates, Inc. * Japan Atomic Energy Research Institute. July 1993. 400pp. 9308160173. GRS-101. 76116:001.

The 2D/3D Program studied multidimensional thermal-hydraulics in a PWR core and primary system during the end-of-blowdown and post-blowdown phases of a large-break LOCA (LBLOCA), and during selected small-break LOCA (SBLOCA) transients. The program included tests at the Cylindrical Core Test Facility (CCTF), the Slab Core Test Facility (SCTF), and the Upper Plenum Test Facility (UPTF), and computer analyses using TRAC. Tests at CCTF investigated core thermal-hydraulics and overall system behavior while tests at SCTF concentrated on multidimensional core thermal-hydraulics. The UPTF tests investigated two-phase flow behavior in the downcomer, upper plenum, tie plate region, and primary loops. TRAC analyses evaluated thermal-hydraulic behavior throughout the primary

system in tests as well as in PWRs. This report summarizes the test and analysis results in each of the main areas where improved information was obtained in the 2D/3D Program. The

discussion is organized in terms of the reactor safety issues investigated.

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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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NUREG/IA-0085: ASSESSMENT OF FULL POWER TURBINE TRIP START-UP TEST FOR C. TRILLO I WITH RELAP5/MOD2
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 NUREG/IA-0107: ASSESSMENT OF RELAP5/MOD2 AGAINST A LOAD REJECTION FROM 100% TO 50% POWER IN THE VANDELLOS II NUCLEAR POWER PLANT.
 NUREG/IA-0112: ASSESSMENT OF RELAP5/MOD2 AGAINST ECN-RE-FLOOD EXPERIMENTS

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 NUREG/IA-0100: ASSESSMENT OF CCFL MODEL OF RELAP5/MOD3 AGAINST SIMPLE VERTICAL TUBES AND ROD BUNDLE TESTS.
 NUREG/IA-0103: ASSESSMENT OF BETHSY TEST 9.1B USING RELAP5/MOD3.
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NUREG/CR-6050: RADIATION EXPOSURE MONITORING AND INFORMATION TRANSMITTAL (REMIT) SYSTEM User's Manual.

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NUREG-1179: RESULTS FROM TWO WORKSHOPS STATE RADIATION CONTROL PROGRAMS DEVELOPING AND AMENDING REGULATIONS AND FUNDING

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NUREG/CR-3469 V07: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.

Radioactive Particle

NUREG/CR-6081: ENHANCED REMOVAL OF RADIOACTIVE PARTICLES BY FLUOROCARBON SURFACTANT SOLUTIONS

Radioactive Waste

NUREG/CR-6073: LYSIMETER LITERATURE REVIEW.

Radionuclide Migration

NUREG/CR-5943: SENSITIVITY ANALYSIS AND BENCHMARKING OF THE BLT LOW-LEVEL WASTE SOURCE TERM CODE.

Reactor Component

NUREG/CR-5993 V01: METHODS FOR DEPENDENCY ESTIMATION AND SYSTEM UNAVAILABILITY EVALUATION BASED ON FAILURE DATA STATISTICS Summary Report.
 NUREG/CR-5993 V02: METHODS FOR DEPENDENCY ESTIMATION AND SYSTEM UNAVAILABILITY EVALUATION BASED ON FAILURE DATA STATISTICS Detailed Description And Applications.

Reactor Pressure Vessel

NUREG/CR-5782: PRESSURIZED THERMAL SHOCK PROBABILISTIC FRACTURE MECHANICS SENSITIVITY ANALYSIS FOR YANKEE ROWE REACTOR PRESSURE VESSEL.
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Reactor Safety

NUREG/CR-6049: PIPING BENCHMARK PROBLEMS FOR THE GENERAL ELECTRIC ADVANCED BOILING WATER REACTOR.

Reactor Safety System

NUREG/CR-6083: REVIEWING REAL-TIME PERFORMANCE OF NUCLEAR REACTOR SAFETY SYSTEMS.

Real-Time Performance

NUREG/CR-6083: REVIEWING REAL-TIME PERFORMANCE OF NUCLEAR REACTOR SAFETY SYSTEMS.

Regulatory Agenda

NUREG-0936 V12 N02: NRC REGULATORY AGENDA Quarterly Report, April-June 1993

Regulatory And Technical Report

NUREG-0304 V18 N02: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Second Quarter 1993, April-June

Regulatory Review And Improvement

NUREG/CP-0129: PROCEEDINGS OF THE WORKSHOP ON PROGRAM FOR ELIMINATION OF REQUIREMENTS MARGINAL TO SAFETY

Reliability

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

Report To Congress

NUREG-0090 V16 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES January-March 1993

Ring Dosimeter

NUREG/CR-5989: PERFORMANCE TESTING OF EXTREMITY DOSIMETERS--PILOT TEST

Risk

NUREG-1449: SHUTDOWN AND LOW-POWER OPERATION AT NUCLEAR POWER PLANTS IN THE UNITED STATES Final Report

Risk Analysis

NUREG/CR-6056: A FRAMEWORK FOR THE ASSESSMENT OF SEVERE ACCIDENT MANAGEMENT STRATEGIES

Risk Management

NUREG/CR-6056: A FRAMEWORK FOR THE ASSESSMENT OF SEVERE ACCIDENT MANAGEMENT STRATEGIES.

Risk-Based Inspection

NUREG/GR-0005 V02 P1: RISK-BASED INSPECTION-DEVELOPMENT OF GUIDELINES Light Water Reactor (LWR) Nuclear Power Plant Components

Risk-Based Regulation

NUREG/CP-0129: PROCEEDINGS OF THE WORKSHOP ON PROGRAM FOR ELIMINATION OF REQUIREMENTS MARGINAL TO SAFETY

Rod Bundle

NUREG/IA-0100: ASSESSMENT OF CCFL MODEL OF RELAP5/MOD3 AGAINST SIMPLE VERTICAL TUBES AND ROD BUNDLE TESTS

Rules

NUREG-0936 V12 N02: NRC REGULATORY AGENDA Quarterly Report, April-June 1993.

Rules Of Practice

NUREG-0386 D06 R07: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST Commission, Appeal Board And Licensing Board Decisions July 1972 - September 1992

Safeguards Summary Event List

NUREG-0525 V02 R01: SAFEGUARDS SUMMARY EVENT LIST (SSEL). January 1, 1990 Through December 31, 1992.

Satellite Telemetry

NUREG/CR-6085: UNITED STATES SEISMOGRAPHIC NETWORK

Seismic Effect

NUREG/CR-6078: ANALYSIS OF CRACK INITIATION AND GROWTH IN THE HIGH LEVEL VIBRATION TEST AT TADOTSU.

Seismic Network

NUREG/CR-5778 V03: NEW YORK/NEW JERSEY REGIONAL SEISMIC NETWORK Final Report For April 1985 - September 1992.
 NUREG/CR-6034: OKLAHOMA SEISMIC NETWORK Final Report
 NUREG/CR-6058: VIRGINIA REGIONAL SEISMIC NETWORK Final Report (1986 -1992)
 NUREG/CR-6079: SEISMOLOGICAL INVESTIGATION OF EARTHQUAKES IN THE NEW MADRID SEISMIC ZONE Final Report September 1986 - December 1992.

Seismographic Network

NUREG/CR-6085: UNITED STATES SEISMOGRAPHIC NETWORK.

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Severe Accident

- NUREG/CR-5942: SEVERE ACCIDENT SOURCE TERM CHARACTERISTICS FOR SELECTED PEACH BOTTOM SEQUENCES PREDICTED BY THE MELCOR CODE
NUREG/CR-6056: A FRAMEWORK FOR THE ASSESSMENT OF SEVERE ACCIDENT MANAGEMENT STRATEGIES

Shallow Crack

- NUREG/CR-5969: J AND CTOD ESTIMATION EQUATIONS FOR SHALLOW CRACKS IN SINGLE EDGE NOTCH BEND SPECIMENS

Shutdown

- NUREG-1449: SHUTDOWN AND LOW-POWER OPERATION AT NUCLEAR POWER PLANTS IN THE UNITED STATES *Final Report*

Source Term

- NUREG/CR-5943: SENSITIVITY ANALYSIS AND BENCHMARKING OF THE BLT LOW-LEVEL WASTE SOURCE TERM CODE

Stainless Steel

- NUREG/CR-4744: V07: N2: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS *Semiannual Report, April-September 1992*

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- NUREG-1479: RESULTS FROM TWO WORKSHOPS: STATE RADIATION CONTROL PROGRAMS DEVELOPING AND AMENDING REGULATIONS AND FUNDING

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- NUREG/IA-0113: PRELIMINARY ASSESSMENT OF PWR STEAM GENERATOR MODELLING IN RELAP5/MOD3

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- NUREG/CR-5754: BOILING-WATER REACTOR INTERNALS AGING DEGRADATION STUDY: Phase 1
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- NUREG/CR-5981: THE EFFECT OF ELECTRIC DISCHARGE MACHINED NOTCHES ON THE FRACTURE TOUGHNESS OF SEVERAL STRUCTURAL ALLOYS

Structural Assessment

- NUREG/CR-5970: APPROXIMATE TECHNIQUES FOR PREDICTING SIZE EFFECTS ON CLEAVAGE FRACTURE TOUGHNESS (JC)

Structural Reliability

- NUREG/CR-6015: STRUCTURAL AGING PROGRAM TECHNICAL PROGRESS FOR PERIOD JANUARY - DECEMBER 1992

Substance Abuse

- NUREG/CR-5758: V03: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY: *Annual Summary Of Program Performance Reports, CY 1992*

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- NUREG/CR-5971: CONTINUUM AND MICROMECHANICS TREATMENT OF CONSTRAINT IN FRACTURE

Surfactant Waste Reduction

- NUREG/CR-6081: ENHANCED REMOVAL OF RADIOACTIVE PARTICLES BY FLUOROCARBON SURFACTANT SOLUTIONS

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- NUREG/CR-6065: SYSTEMS ANALYSIS OF THE CANDU 3 REACTOR

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- NUREG/CR-5993: V02: METHODS FOR DEPENDENCY ESTIMATION AND SYSTEM UNAVAILABILITY EVALUATION BASED ON FAILURE DATA STATISTICS *Detailed Description And Applications*

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- NUREG/CR-6026: 2D/3D PROGRAM WORK SUMMARY REPORT
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- NUREG/CR-6021: A LITERATURE REVIEW OF COUPLED THERMAL-HYDROLOGIC-MECHANICAL-CHEMICAL PROCESSES PERTINENT TO THE PROPOSED HIGH-LEVEL WASTE REPOSITORY AT YUCCA MOUNTAIN

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- NUREG/IA-0112: ASSESSMENT OF RELAP5/MOD2 AGAINST ECN-REFLOOD EXPERIMENTS

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- NUREG/CR-6026: THEORETICAL AND EXPERIMENTAL INVESTIGATION OF THERMOHYDROLOGIC PROCESSES IN A PARTIALLY SATURATED, FRACTURED POROUS MEDIUM

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- NUREG-0837: V13: N02: NRC TLD DIRECT RADIATION MONITORING NETWORK: *Progress Report, April-June 1993*

Title List

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NUREG-0540: V15: N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE: *June 1-30, 1993*
NUREG-0540: V15: N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE: *July 1-31, 1993*

Turbine Trip

- NUREG/IA-0085: ASSESSMENT OF FULL POWER TURBINE TRIP START-UP TEST FOR C. TRILLO I WITH RELAP5/MOD2

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- NUREG/CR-6072: EXPERIMENTAL STUDY ON THE COMBUSTION BEHAVIOR OF HYDROGEN-AIR MIXTURES WITH TURBULENT JET IGNITION AT LARGE SCALE

UK Numerics And Implementation

- NUREG/IA-0096: NUMERICS AND IMPLEMENTATION OF THE UK HORIZONTAL STRATIFICATION ENTRAINMENT OFF-TAKE MODEL INTO RELAP5/MOD3

Vendor Inspection

- NUREG-0040: V17: N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT: *Quarterly Report, April-June 1993 (White Book)*

Vertical Tube

- NUREG/IA-6021: ASSESSMENT OF CCFL MODEL OF RELAP5/MOD3 AGAINST SIMPLE VERTICAL TUBES AND ROD BUNDLE TESTS

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- NUREG/CR-6078: ANALYSIS OF CRACK INITIATION AND GROWTH IN THE HIGH LEVEL VIBRATION TEST AT TADOTSU

Yucca Mountain

- NUREG/CR-6021: A LITERATURE REVIEW OF COUPLED THERMAL-HYDROLOGIC-MECHANICAL-CHEMICAL PROCESSES PERTINENT TO THE PROPOSED HIGH-LEVEL WASTE REPOSITORY AT YUCCA MOUNTAIN

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.

ADVISORY COMMITTEE(S)

ADVISORY COMMITTEE ON NUCLEAR WASTE
NUREG-1423 V04: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON NUCLEAR WASTE July 1992 - June 1993

ATOMIC SAFETY BOARD(S) & PANEL(S)

ATOMIC SAFETY & LICENSING BOARD PANEL
NUREG-1363 V05: ATOMIC SAFETY AND LICENSING BOARD PANEL ANNUAL REPORT, Fiscal Year 1992.

OFFICE OF EXECUTIVE DIRECTOR FOR OPERATIONS (EDO)

REGION 1 (POST 820201)
NUREG-0837 V13 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK Progress Report, April-June 1993.
REGION 2 (POST 820201)
NUREG/CR-5833: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE H.B. ROBINSON NUCLEAR POWER PLANT
OFC OF ENFORCEMENT (POST 870413)
NUREG-0940 V12 N02: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED Quarterly Progress Report, April-June 1993

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

OFFICE OF ADMINISTRATION (POST 890205)
NUREG-1145 V09: U.S. NUCLEAR REGULATORY COMMISSION 1992 ANNUAL REPORT.
DIVISION OF FREEDOM OF INFORMATION & PUBLICATIONS SERVICES (POST 890205)
NUREG-0304 V18 N02: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL) Compilation For Second Quarter 1993, April-June.
NUREG-0540 V15 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE May 1-31, 1993.
NUREG-0540 V15 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE June 1-30, 1993.
NUREG-0540 V15 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE July 1-31, 1993.
NUREG-0750 V37 N01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES January-March 1993.
NUREG-0750 V37 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1993 Pages 251-354.
NUREG-0750 V37 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1993 Pages 355-418.
NUREG-0750 V37 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1993 Pages 419-515.
NUREG-0750 V38 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1993 Pages 1-24.
NUREG-0936 V12 N02: NRC REGULATORY AGENDA Quarterly Report, April-June 1993.

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

OFFICE OF THE CONTROLLER (POST 890205)
NUREG-1470 V02: CHIEF FINANCIAL OFFICER'S ANNUAL REPORT - 1993.

OFFICE OF STATE PROGRAMS (PRE 870413 & POST 911117)

OFFICE OF STATE PROGRAMS (POST 911117)
NUREG-1479: RESULTS FROM TWO WORKSHOPS: STATE RADIATION CONTROL PROGRAMS DEVELOPING AND AMENDING REGULATIONS AND FUNDING.

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA

OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA, DIRECTOR
NUREG-0090 V16 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES January-March 1993.

NUREG-1272 V07 N01: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA 1992 Annual Report - Power Reactors

EDO - OFFICE OF INFORMATION RESOURCES MANAGEMENT & ARM (POST 881109)

DIVISION OF INFORMATION SUPPORT SERVICES (POST 890205)
NUREG-0910 R02 S01: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE

EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS

DIVISION OF LOW-LEVEL WASTE MANAGEMENT & DECOMMISSIONING (POST 870413)
NUREG-1476: FINAL ENVIRONMENTAL IMPACT STATEMENT TO CONSTRUCT AND OPERATE A FACILITY TO RECEIVE, STORE, AND DISPOSE OF 11E.(2) BYPRODUCT MATERIAL NEAR CLIVE, UTAH, Docket No. 40-8989, Envirocare Of Utah, Inc.
OPERATIONS BRANCH
NUREG-0525 V02 R01: SAFEGUARDS SUMMARY EVENT LIST (SSEL) January 1, 1990 Through December 31, 1992

U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF THE GENERAL COUNSEL (POST 860701)
NUREG-0386 D06 R07: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST Commission, Appeal Board And Licensing Board Decisions July 1972 - September 1992

EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)

DIVISION OF REGULATORY APPLICATIONS (POST 870413)
NUREG-0713 V13: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES, 1991 Twenty-Fourth Annual Report.
NUREG-1400: AIR SAMPLING IN THE WORKPLACE Final Report.
DIVISION OF SAFETY ISSUE RESOLUTION (POST 880717)
NUREG-1453: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC ISSUE 142: LEAKAGE THROUGH ELECTRICAL ISOLATORS IN INSTRUMENTATION CIRCUITS.
NUREG-1461: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC ISSUE 153: LOSS OF ESSENTIAL SERVICE WATER IN LWRS.
NUREG-1463: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC SAFETY ISSUE 105: INTERFACING SYSTEM LOSS-OF-COOLANT ACCIDENT IN LIGHT-WATER REACTORS
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EDO - OFFICE OF NUCLEAR REACTOR REGULATION (POST 800428)

PROJECT DIRECTORATE III-3
NUREG/CR-5829: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDANCE FOR THE DAVIS-BESSE NUCLEAR POWER PLANT.
DIVISION OF SYSTEMS SAFETY & ANALYSIS (POST 921004)
NUREG-1449: SHUTDOWN AND LOW-POWER OPERATION AT NUCLEAR POWER PLANTS IN THE UNITED STATES Final Report.
DIVISION OF REACTOR INSPECTION & LICENSEE PERFORMANCE (POST 921004)
NUREG-0040 V17 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, April-June 1993 (White Book)
NUREG-1214 R12: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE.

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.

EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)

OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 880720)
NUREG/IA-0091 ASSESSMENT OF RELAP5/MOD2 AGAINST A
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NUREG/IA-0085 ASSESSMENT OF FULL POWER TURBINE TRIP
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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 OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS

DIVISION OF HIGH-LEVEL WASTE MANAGEMENT (POST 870413)

NUREG/CR-6021: A LITERATURE REVIEW OF COUPLED THERMAL-HYDROLOGIC-MECHANICAL-CHEMICAL PROCESSES PERTINENT TO THE PROPOSED HIGH-LEVEL WASTE REPOSITORY AT YUCCA MOUNTAIN

EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)

DIVISION OF ENGINEERING (POST 870413)

NUREG/CR-4667 V16: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report, October 1992 - March 1993

NUREG/CR-4744 V07 N2: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS. Semiannual Report, April-September 1992

NUREG/CR-5404 V02: AUXILIARY FEEDWATER SYSTEM AGING STUDY. Phase I Follow-On Study

NUREG/CR-5754: BOILING-WATER REACTOR INTERNALS AGING DEGRADATION STUDY. Phase I

NUREG/CR-5778 V03: NEW YORK/NEW JERSEY REGIONAL SEISMIC NETWORK. Final Report For April 1985 - September 1992

NUREG/CR-5782: PRESSURIZED THERMAL SHOCK PROBABILISTIC FRACTURE MECHANICS SENSITIVITY ANALYSIS FOR YANKEE ROWE REACTOR PRESSURE VESSEL

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

NUREG/CR-5969: J AND CTOD ESTIMATION EQUATIONS FOR SHALLOW CRACKS IN SINGLE EDGE NOTCH BEND SPECIMENS

NUREG/CR-5970: APPROXIMATE TECHNIQUES FOR PREDICTING SIZE EFFECTS ON CLEAVAGE FRACTURE TOUGHNESS (JC)

NUREG/CR-5971: CONTINUUM AND MICROMECHANICS TREATMENT OF CONSTRAINT IN FRACTURE

NUREG/CR-5981: THE EFFECT OF ELECTRIC DISCHARGE MACHINED NOTCHES ON THE FRACTURE TOUGHNESS OF SEVERAL STRUCTURAL ALLOYS

NUREG/CR-6015: STRUCTURAL AGING PROGRAM TECHNICAL PROGRESS FOR PERIOD JANUARY - DECEMBER 1992

NUREG/CR-6023: GENERIC ANALYSIS FOR EVALUATION OF LOW CHARPY UPPER-SHELF ENERGY EFFECTS ON SAFETY MARGINS AGAINST FRACTURE OF REACTOR PRESSURE VESSEL MATERIALS

NUREG/CR-6029 V01: AGING ASSESSMENT OF NUCLEAR AIR-TREATMENT SYSTEM HEPA FILTERS AND ADSORBERS. Phase I

NUREG/CR-6034: OKLAHOMA SEISMIC NETWORK. Final Report

NUREG/CR-6043 V01: AGING ASSESSMENT OF ESSENTIAL HVAC CHILLERS USED IN NUCLEAR POWER PLANTS. Phase I

NUREG/CR-6048: PRESSURIZED-WATER REACTOR INTERNALS AGING DEGRADATION STUDY. Phase I

NUREG/CR-6052: METHODOLOGY FOR RELIABILITY BASED CONDITION ASSESSMENT. Application To Concrete Structures In Nuclear Plants

NUREG/CR-6058: VIRGINIA REGIONAL SEISMIC NETWORK. Final Report (1986 - 1992)

NUREG/CR-6078: ANALYSIS OF CRACK INITIATION AND GROWTH IN THE HIGH LEVEL VIBRATION TEST AT TADOTSU

NUREG/CR-6079: SEISMOLOGICAL INVESTIGATION OF EARTHQUAKES IN THE NEW MADRID SEISMIC ZONE. Final Report. September 1986 - December 1992

NUREG/CR-6085: UNITED STATES SEISMOGRAPHIC NETWORK. DIVISION OF REGULATORY APPLICATIONS (POST 870413)

NUREG/CR-3469 V07: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS. ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA

NUREG/CR-5817 V03 N2: NRC HIGH-LEVEL RADIOACTIVE WASTE RESEARCH AT CNWRA. July-December 1992

NUREG/CR-5927 V01: EVALUATION OF A PERFORMANCE ASSESSMENT METHODOLOGY FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES. Evaluation Of Modeling Approaches

NUREG/CR-5943: SENSITIVITY ANALYSIS AND BENCHMARKING OF THE BLT LOW-LEVEL WASTE SOURCE TERM CODE

NUREG/CR-5989: PERFORMANCE TESTING OF EXTREMITY DOSIMETERS-PILOT TEST

NUREG/CR-6026: THEORETICAL AND EXPERIMENTAL INVESTIGATION OF THERMOHYDROLOGIC PROCESSES IN A PARTIALLY SATURATED, FRACTURED POROUS MEDIUM

NUREG/CR-6050: RADIATION EXPOSURE MONITORING AND INFORMATION TRANSMITTAL (REMIT) SYSTEM. User's Manual

NUREG/CR-6073: LYSIMETER LITERATURE REVIEW

NUREG/CR-6081: ENHANCED REMOVAL OF RADIOACTIVE PARTICLES BY FLUOROCARBON SURFACTANT SOLUTIONS

DIVISION OF SAFETY ISSUE RESOLUTION (POST 880717)

NUREG/CR-5928: ISLOCA RESEARCH PROGRAM. Final Report

NUREG/CR-5942: SEVERE ACCIDENT SOURCE TERM CHARACTERISTICS FOR SELECTED PEACH BOTTOM SEQUENCES PREDICTED BY THE MELCOR CODE

DIVISION OF SYSTEMS RESEARCH (POST 880717)

NUREG/CR-5993 V01: METHODS FOR DEPENDENCY ESTIMATION AND SYSTEM UNAVAILABILITY EVALUATION BASED ON FAILURE DATA STATISTICS. Summary Report

NUREG/CR-5993 V02: METHODS FOR DEPENDENCY ESTIMATION AND SYSTEM UNAVAILABILITY EVALUATION BASED ON FAILURE DATA STATISTICS. Detailed Description And Applications

NUREG/CR-6056: A FRAMEWORK FOR THE ASSESSMENT OF SEVERE ACCIDENT MANAGEMENT STRATEGIES

NUREG/CR-6060: HYDROGEN MIXING STUDIES (HMS) ASSESSMENT MANUAL

NUREG/CR-6065: SYSTEMS ANALYSIS OF THE CANDU 3 REACTOR

NUREG/CR-6072: EXPERIMENTAL STUDY ON THE COMBUSTION BEHAVIOR OF HYDROGEN-AIR MIXTURES WITH TURBULENT JET IGNITION AT LARGE SCALE

EDO - OFFICE OF NUCLEAR REACTOR REGULATION (POST 800428)

DIVISION OF REACTOR CONTROLS & HUMAN FACTORS (POST 921004)

NUREG/CR-6082: DATA COMMUNICATIONS

NUREG/CR-6083: REVIEWING REAL-TIME PERFORMANCE OF NUCLEAR REACTOR SAFETY SYSTEMS

NUREG/CR-6090: THE PROGRAMMABLE LOGIC CONTROLLER AND ITS APPLICATION IN NUCLEAR REACTOR SYSTEMS

DIVISION OF SYSTEMS SAFETY & ANALYSIS (POST 921004)

NUREG/CR-5829: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDANCE FOR THE DAVIS-BESSE NUCLEAR POWER PLANT

NUREG/CR-5833: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE H.B. ROBINSON NUCLEAR POWER PLANT

DIVISION OF RADIATION SAFETY & SAFEGUARDS (POST 921004)

NUREG/CR-5758 V03: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY. Annual Summary Of Program Performance Reports, CY 1992

DIVISION OF ENGINEERING (POST 921004)

NUREG/CR-6049: PIPING BENCHMARK PROBLEMS FOR THE GENERAL ELECTRIC ADVANCED BOILING WATER REACTOR

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.

ADVANCED SYSTEMS TECHNOLOGY, INC.

NUREG-1400: AIR SAMPLING IN THE WORKPLACE.Final Report.

AMERICAN SOCIETY OF MECHANICAL ENGINEERS

NUREG/GR-0005 V02 P1: RISK-BASED INSPECTION-DEVELOPMENT OF GUIDELINES.Light Water Reactor (LWR) Nuclear Power Plant Components.

ARGONNE NATIONAL LABORATORY

NUREG/CR-4667 V16: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1992 - March 1993.

NUREG/CR-4744 V07 N2: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,April-September 1992.

ARIZONA, UNIV. OF, TUCSON, AZ

NUREG/CP-0040: PROCEEDINGS OF WORKSHOP V: FLOW AND TRANSPORT THROUGH UNSATURATED FRACTURED ROCK -- RELATED TO HIGH-LEVEL RADIOACTIVE WASTE DISPOSAL.Held At Radisson Suite Hotel, Tucson, Arizona,January 7-10,1991.

BATTELLE HUMAN AFFAIRS RESEARCH CENTERS

NUREG/CR-5758 V03: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports,CY 1992.

BATTELLE MEMORIAL INSTITUTE, PACIFIC NORTHWEST LABORATORY

NUREG-1400: AIR SAMPLING IN THE WORKPLACE.Final Report.
NUREG/CR-5758 V03: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports,CY 1992.

NUREG/CR-5829: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDANCE FOR THE DAVIS-BESSE NUCLEAR POWER PLANT.

NUREG/CR-5833: AUXILIARY FEEDWATER SYSTEM RISK-BASED INSPECTION GUIDE FOR THE H.B. ROBINSON NUCLEAR POWER PLANT.

NUREG/CR-5989: PERFORMANCE TESTING OF EXTREMITY DOSIMETERS--PILOT TEST.

NUREG/CR-6029 V01: AGING ASSESSMENT OF NUCLEAR AIR-TREATMENT SYSTEM HEPA FILTERS AND ADSORBERS.Phase I.

NUREG/CR-6043 V01: AGING ASSESSMENT OF ESSENTIAL HVAC CHILLERS USED IN NUCLEAR POWER PLANTS.Phase I.

BROOKHAVEN NATIONAL LABORATORY

NUREG/CR-3469 V07: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.

NUREG/CR-5943: SENSITIVITY ANALYSIS AND BENCHMARKING OF THE BLT LOW-LEVEL WASTE SOURCE TERM CODE.

NUREG/CR-5993 V01: METHODS FOR DEPENDENCY ESTIMATION AND SYSTEM UNAVAILABILITY EVALUATION BASED ON FAILURE DATA STATISTICS.Summary Report.

NUREG/CR-5993 V02: METHODS FOR DEPENDENCY ESTIMATION AND SYSTEM UNAVAILABILITY EVALUATION BASED ON FAILURE DATA STATISTICS.Detailed Description And Applications.

NUREG/CR-6049: PIPING BENCHMARK PROBLEMS FOR THE GENERAL ELECTRIC ADVANCED BOILING WATER REACTOR.

NUREG/CR-6078: ANALYSIS OF CRACK INITIATION AND GROWTH IN THE HIGH LEVEL VIBRATION TEST AT TADOTSU.

BROWN UNIV., PROVIDENCE, RI

NUREG/CR-5971: CONTINUUM AND MICROMECHANICS TREATMENT OF CONSTRAINT IN FRACTURE.

CALIFORNIA, UNIV. OF, LOS ANGELES, CA

NUREG/CR-6056: A FRAMEWORK FOR THE ASSESSMENT OF SEVERE ACCIDENT MANAGEMENT STRATEGIES.

CENTER FOR NUCLEAR WASTE REGULATORY ANALYSES

NUREG/CR-5817 V03 N2: NRC HIGH-LEVEL RADIOACTIVE WASTE RESEARCH AT CNWRA,July-December 1992.

NUREG/CR-6021: A LITERATURE REVIEW OF COUPLED THERMAL-HYDROLOGIC-MECHANICAL-CHEMICAL PROCESSES PERTINENT TO THE PROPOSED HIGH-LEVEL WASTE REPOSITORY AT YUCCA MOUNTAIN.

NUREG/CR-6026: THEORETICAL AND EXPERIMENTAL INVESTIGATION OF THERMOHYDROLOGIC PROCESSES IN A PARTIALLY SATURATED, FRACTURED POROUS MEDIUM.

EG&G IDAHO, INC.

NUREG/CR-5928: ISLOCA RESEARCH PROGRAM.Final Report.

NUREG/CR-6073: LYSIMETER LITERATURE REVIEW.

HARVARD SCHOOL OF PUBLIC HEALTH, BOSTON, MA

NUREG/CP-0130 V01: PROCEEDINGS OF THE 22ND DOE/NRC NUCLEAR AIR CLEANING CONFERENCE.Sessions 1-8.Held In Denver,Colorado,August 24-27,1992.

NUREG/CP-0130 V02: PROCEEDINGS OF THE 22ND DOE/NRC NUCLEAR AIR CLEANING CONFERENCE.Sessions 9-16.Held In Denver,Colorado,August 24-27, 1992.

ILLINOIS, UNIV. OF, URBANA, IL

NUREG/CR-5969: J AND CTOD ESTIMATION EQUATIONS FOR SHALLOW CRACKS IN SINGLE EDGE NOTCH BEND SPECIMENS.

NUREG/CR-5970: APPROXIMATE TECHNIQUES FOR PREDICTING SIZE EFFECTS ON CLEAVAGE FRACTURE TOUGHNESS (JC).

NUREG/CR-5971: CONTINUUM AND MICROMECHANICS TREATMENT OF CONSTRAINT IN FRACTURE.

INTERIOR, DEPT. OF, GEOLOGICAL SURVEY

NUREG/CR-6085: UNITED STATES SEISMOGRAPHIC NETWORK.

JOHNS HOPKINS UNIV., BALTIMORE, MD

NUREG/CR-6052: METHODOLOGY FOR RELIABILITY BASED CONDITION ASSESSMENT: Application To Concrete Structures In Nuclear Plants.

LAMONT-DOHERTY GEOLOGICAL OBSERVATORY

NUREG/CR-5778 V03: NEW YORK/NEW JERSEY REGIONAL SEISMIC NETWORK.Final Report For April 1985 - September 1992.

LAWRENCE LIVERMORE NATIONAL LABORATORY

NUREG/CR-6082: DATA COMMUNICATIONS.

NUREG/CR-6083: REVIEWING REAL-TIME PERFORMANCE OF NUCLEAR REACTOR SAFETY SYSTEMS.

NUREG/CR-6090: THE PROGRAMMABLE LOGIC CONTROLLER AND ITS APPLICATION IN NUCLEAR REACTOR SYSTEMS.

LOS ALAMOS NATIONAL LABORATORY

NUREG/CR-6060: HYDROGEN MIXING STUDIES (HMS) ASSESSMENT MANUAL.

NAVY, DEPT. OF

NUREG/CR-5969: J AND CTOD ESTIMATION EQUATIONS FOR SHALLOW CRACKS IN SINGLE EDGE NOTCH BEND SPECIMENS.

NUREG/CR-5970: APPROXIMATE TECHNIQUES FOR PREDICTING SIZE EFFECTS ON CLEAVAGE FRACTURE TOUGHNESS (JC).

NUREG/CR-5981: THE EFFECT OF ELECTRIC DISCHARGE MACHINED NOTCHES ON THE FRACTURE TOUGHNESS OF SEVERAL STRUCTURAL ALLOYS.

OAK RIDGE NATIONAL LABORATORY

NUREG/CR-5404 V02: AUXILIARY FEEDWATER SYSTEM AGING STUDY.Phase I Follow-On Study.

NUREG/CR-5754: BOILING-WATER REACTOR INTERNALS AGING DEGRADATION STUDY. Phase 1.

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NUREG/CR-5782: PRESSURIZED THERMAL SHOCK PROBABILISTIC FRACTURE MECHANICS SENSITIVITY ANALYSIS FOR YANKEE ROWE REACTOR PRESSURE VESSEL.
NUREG/CR-5942: SEVERE ACCIDENT SOURCE TERM CHARACTERISTICS FOR SELECTED PEACH BOTTOM SEQUENCES PREDICTED BY THE MELCOR CODE.
NUREG/CR-5944: A CHARACTERIZATION OF CHECK VALVE DEGRADATION AND FAILURE EXPERIENCE IN THE NUCLEAR POWER INDUSTRY.
NUREG/CR-6015: STRUCTURAL AGING PROGRAM TECHNICAL PROGRESS FOR PERIOD JANUARY - DECEMBER 1992.
NUREG/CR-6023: GENERIC ANALYSIS FOR EVALUATION OF LOW CHARPY UPPER-SHELF ENERGY EFFECTS ON SAFETY MARGINS AGAINST FRACTURE OF REACTOR PRESSURE VESSEL MATERIALS.
NUREG/CR-6048: PRESSURIZED-WATER REACTOR INTERNALS AGING DEGRADATION STUDY, Phase 1.
NUREG/CR-6052: METHODOLOGY FOR RELIABILITY BASED CONDITION ASSESSMENT. Application To Concrete Structures In Nuclear Plants.
NUREG/CR-6065: SYSTEMS ANALYSIS OF THE CANDU 3 REACTOR.

OKLAHOMA, UNIV. OF, NORMAN, OK

NUREG/CR-6034: OKLAHOMA SEISMIC NETWORK. Final Report.

RUSSIA

NUREG/CR-6072: EXPERIMENTAL STUDY ON THE COMBUSTION BEHAVIOR OF HYDROGEN-AIR MIXTURES WITH TURBULENT JET IGNITION AT LARGE SCALE.

SANDIA NATIONAL LABORATORIES

NUREG/CR-5927 V01: EVALUATION OF A PERFORMANCE ASSESSMENT METHODOLOGY FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES. Evaluation Of Modeling Approaches.

SCIENCE APPLICATIONS INTERNATIONAL CORP. (FORMERLY SCIENCE APPLICATIONS,

NUREG-0713 V13: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES, 1991. Twenty-Fourth Annual Report.

NUREG/CR-5993 V01: METHODS FOR DEPENDENCY ESTIMATION AND SYSTEM UNAVAILABILITY EVALUATION BASED ON FAILURE DATA STATISTICS. Summary Report.

NUREG/CR-5993 V02: METHODS FOR DEPENDENCY ESTIMATION AND SYSTEM UNAVAILABILITY EVALUATION BASED ON FAILURE DATA STATISTICS. Detailed Description And Applications.

NUREG/CR-6050: RADIATION EXPOSURE MONITORING AND INFORMATION TRANSMITTAL (REMIT) SYSTEM. User's Manual.

NUREG/CR-6060: HYDROGEN MIXING STUDIES (HMS) ASSESSMENT MANUAL.

SCIENTECH, INC.

NUREG/CR-0129: PROCEEDINGS OF THE WORKSHOP ON PROGRAM FOR ELIMINATION OF REQUIREMENTS MARGINAL TO SAFETY.

SOUTHWEST RESEARCH INSTITUTE

NUREG/CR-6026: THEORETICAL AND EXPERIMENTAL INVESTIGATION OF THERMOHYDROLOGIC PROCESSES IN A PARTIALLY SATURATED, FRACTURED POROUS MEDIUM.

ST. LOUIS UNIV., ST. LOUIS, MO

NUREG/CR-6079: SEISMOLOGICAL INVESTIGATION OF EARTHQUAKES IN THE NEW MADRID SEISMIC ZONE. Final Report. September 1986 - December 1992.

TEXAS A&M UNIV., COLLEGE STATION, TX

NUREG/CR-5971: CONTINUUM AND MICROMECHANICS TREATMENT OF CONSTRAINT IN FRACTURE.

U.S. NAVAL ACADEMY, ANNAPOLIS, MD

NUREG/CR-5981: THE EFFECT OF ELECTRIC DISCHARGE MACHINED NOTCHES ON THE FRACTURE TOUGHNESS OF SEVERAL STRUCTURAL ALLOYS.

VIRGINIA POLYTECHNIC INSTITUTE & STATE UNIV., BLACKSBURG, VA

NUREG/CR-6058: VIRGINIA REGIONAL SEISMIC NETWORK. Final Report (1986 - 1992).

International Organization Index

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

JAPAN

JAPAN ATOMIC ENERGY RESEARCH INSTITUTE
NUREG/IA-0126: 2D/3D PROGRAM WORK SUMMARY REPORT.
NUREG/IA-0127: REACTOR SAFETY ISSUES RESOLVED BY THE
2D/3D PROGRAM.

REPUBLIC OF KOREA

KOREA ELECTRIC POWER CORPORATION
NUREG/IA-0100: ASSESSMENT OF CCFL MODEL OF RELAP5/
MOD3 AGAINST SIMPLE VERTICAL TUBES AND ROD BUNDLE
TESTS.
KOREA INSTITUTE OF NUCLEAR SAFETY
NUREG/IA-0103: ASSESSMENT OF BETHSY TEST 9.1.B USING
RELAP5/MOD3.

SPAIN

ASOCIACION NUCLEAR DE VANDELLOS
NUREG/IA-0107: ASSESSMENT OF RELAP5/MOD2 AGAINST A
LOAD REJECTION FROM 100% TO 50% POWER IN THE VAN-
DELLOS II NUCLEAR POWER PLANT.

CONSEJO DE SEGURIDAD NUCLEAR

NUREG/IA-0085: ASSESSMENT OF FULL POWER TURBINE TRIP
START-UP TEST FOR C. TRILLO I WITH RELAP5/MOD2.

THE NETHERLANDS

NETHERLANDS ENERGY RESEARCH FOUNDATION ECN
NUREG/IA-0091: ASSESSMENT OF RELAP5/MOD2 AGAINST A
NATURAL CIRCULATION EXPERIMENT IN NUCLEAR POWER
PLANT BORSSELE.
NUREG/IA-0112: ASSESSMENT OF RELAP5/MOD2 AGAINST ECN-
REFLOOD EXPERIMENTS.

UNITED KINGDOM

NATIONAL POWER
NUREG/IA-0113: PRELIMINARY ASSESSMENT OF PWR STEAM
GENERATOR MODELLING IN RELAP5/MOD3.
WINFRITH TECHNOLOGY CENTRE
NUREG/IA-0096: NUMERICS AND IMPLEMENTATION OF THE UK
HORIZONTAL STRATIFICATION ENTRAINMENT OFF-TAKE
MODEL INTO RELAP5/MOD3.

Licensed Facility Index

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

50-346	Davis-Besse Nuclear Power Station, Unit 1, Toledo Edison Co.	NUREG/CR-5829	50-277	Peach Bottom Atomic Power Station, Unit 2, Philadelphia Electric Co.	NUREG/CR-5942
40-8989	Envirocare of Utah, Inc., Salt Lake City, UT,	NUREG-1476	50-278	Peach Bottom Atomic Power Station, Unit 3, Philadelphia Electric Co.	NUREG/CR-5942
50-261	H.B. Robinson Plant, Unit 2, Carolina Power & Light Co.	NUREG/CR-5833	50-29	Yankee-Rose Nuclear Power Station, Yankee Atomic Electric Co.	NUREG/CR-5782
50-171	Peach Bottom Atomic Power Station, Unit 1, Philadelphia Electric Co.	NUREG/CR-5942			

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