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Vol. 12, No. 2

Regulatory and Technical Reports (Abstract Index Journal)

Compilation for
Second Quarter 1987
April - June

U.S. Nuclear Regulatory Commission

Office of Administration and Resources Management

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Office of Administration and Resources Management
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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:

Division of Publications Services
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Woodmont 537
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Washington, D.C. 20555

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
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A detailed explanation of the entries precedes each index.

The bibliographic elements of the main citations are the following:

Staff Report

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

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

Conference Report

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

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

Contractor Report

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

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

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International Agreement Report

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

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

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

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

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NRC Report Codes

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

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

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

Main Citations and Abstracts

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

NUREG-0020 V10 N11: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of October 31,1986.(Gray Book I) ROSS,P.A.; BEEBE,M.R. Division of Computer & Telecommunications Services (Post 870413). April 1987. 471pp. 8705120010. 40901:348.

The OPERATING UNITS STATUS REPORT - LICENSED OPERATING REACTORS provides data on the operation of nuclear units as timely and accurately as possible. This information is collected by the Office of Resource Management from the Headquarters staff of NRC's Office of Inspection and Enforcement, from NRC's Regional Offices, and from utilities. The three sections of the report are: monthly highlights and statistics for commercial operating units, and errata from previously reported data; a compilation of detailed information on each unit, provided by NRC's Regional Offices, IE Headquarters and the utilities; and an appendix for miscellaneous information such as spent fuel storage capability, reactor-years of experience and non-power reactors in the U.S. It is hoped the report is helpful to all agencies and individuals interested in maintaining an awareness of the U.S. energy situation as a whole.

NUREG-0020 V10 N12: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of November 30,1986.(Gray Book I) ROSS,P.A. Division of Computer & Telecommunications Services (Post 870413). June 1987. 450pp. 8707060427. 41582:185.

See NUREG-0020,V10,N11 abstract.

NUREG-0040 V11 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1987.(White Book) * Division of Reactor Inspection & Safeguards (Post 870411). May 1987. 235pp. 8706160106. 41311:032.

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 January 1987 through March 1987. Also, included in this issue are the results of certain inspections performed prior to January 1987 that were not included in previous issues of NUREG-0040.

NUREG-0090 V09 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1986. * Office for Analysis & Evaluation of Operational Data, Director. April 1987. 60pp. 8705290301. 41115:296.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which 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 July 1 to September 30, 1986. During the report period, there were four abnormal occurrences at the nuclear power plants licensed to operate. The events were (1) a differential pressure switch problem in safety systems at LaSalle facility, (2) abnormal cooldown and depressurization transient at Catawba Unit 2, (3) significant safeguards deficiencies at Wolf Creek and Fort St. Vrain, and (4) significant deficiencies in access controls at River Bend Station. There was one abnormal occurrence at the other NRC licensees; it involved a therapeutic medical misadministration. There was one abnormal occurrence reported by an Agreement State; it involved a therapeutic medical misadministration. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0304 V12 N01: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1987,January-March. * Division of Publication Services (Post 870413). May 1987. 62pp. 8706030076. 41166:112.

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-0332: POTENTIAL HEALTH AND ENVIRONMENTAL IMPACTS ATTRIBUTABLE TO THE NUCLEAR AND COAL FUEL CYCLES.Final Report. GOTCHY,R.L. Office of Nuclear Reactor Regulation, Director (Post 870411). June 1987. 73pp. 8707070538. 41600:001.

Estimates of mortality and morbidity are presented based on present-day knowledge of health effects resulting from current component designs and operations of the nuclear and coal fuel cycles, and anticipated emission rates and occupational exposure for the various fuel cycle facilities expected to go into operation during the next decade. The author concluded that, although there are large uncertainties in the estimates of potential health effects, the coal fuel cycle alternative has a greater health impact on man than the uranium fuel cycle. However, the increased risk of health effects for either fuel cycle represents a very small incremental risk to the average individual in the public for the balance of this century. The potential for large impacts exists in both fuel cycles, but the potential impacts associated with a runaway Greenhouse Effect from combustion of fossil fuels, such as coal, cannot yet be reasonably quantified. Some of the potential environmental impacts of the coal fuel cycle cannot currently be realistically estimated, but those that can appear greater than those from the nuclear fuel cycle.

NUREG-0386 D04 R05: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.July 1972 - September 1986. * Office of the General Counsel. June 1987. 300pp. 8707130184. 41682:228.

This Revision 5 of the fourth edition of the NRC Staff 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 July 1, 1972 to September 30, 1986, interpreting the NRC Rules of Practice in 10 CFR Part 2. This Revision 5 replaces in part earlier editions and supplements and includes appropriate changes reflecting the amendments to the Rules of Practice effective September 30, 1986.

NUREG-0540 V09 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. February 1-28,1987. * Division of Publication Services (Post 870413). April 1987. 400pp. 8706240149. 41440:175.

This document is a monthly publication containing descriptions of information received and generated by the U.S. 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 Index, Corporate Source Index, Report Number Index and Cross Reference to Principal Documents Index.

NUREG-0540 V09 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. March 1-31,1987. * Division of Publication Services (Post 870413). May 1987. 469pp 8706160119. 41315:256.

See NUREG-0540,V09,N02 abstract.

NUREG-0540 V09 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30,1987. * Division of Publication Services (Post 870413). June 1987. 450pp. 8706230495. 41434:117.

See NUREG-0540,V09,N02 abstract.

NUREG-0683 S02: PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT RELATED TO DECONTAMINATION AND DISPOSAL OF RADIOACTIVE WASTES RESULTING FROM MARCH 28,1979 ACCIDENT AT THREE MILE ISLAND NUCLEAR STATION,UNIT 2.Final Supplement Dealing With Disposal Of.... * TMI-2 Cleanup Project Directorate. June 1987. 300pp. 8707080318. 41616:213.

In accordance with the National Environmental Policy Act, the Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Waste for the 1979 Accident at Three Mile Island Nuclear Station, Unit 2 (PEIS) has been supplemented. This supplement updates the environmental evaluation of accident-generated water disposal alternatives published in the PEIS, utilizing more complete and current information, and covering the licensee's proposal to dispose of the water by evaporation to the atmosphere. The staff concludes that the water can be disposed of without incurring significant environmental impact. The staff's evaluation of a number of disposal alternatives indicates that no alternative is clearly preferable to the others, and that the licensee's proposal method is satisfactory. The risks to the general public from exposure to radioactive effluents from any alternative have been quantitatively estimated and are very small fractions of the estimated normal incidence of cancer fatalities and genetic disorders. The most significant potential impact associated with any disposal alternative is the risk of physical injury associated with transportation accidents. Additionally, no significant impacts to aquatic or terrestrial biotic from any disposal alternative are expected.

NUREG-0728 R02: NRC INCIDENT RESPONSE PLAN. * Office for Analysis & Evaluation of Operational Data, Director. June 1987. 29pp. 8706240202. 41448:219.

The Nuclear Regulatory Commission (NRC) regulates civilian nuclear activities to protect the public health and safety and to preserve environmental quality. An Incident Response Plan had been developed and has now been revised for the second time to reflect current Commission policy. NUREG-0728, Rev 2 assigns responsibilities for responding to any potentially threatening incident involving NRC licensed activities and for assuring that the NRC will fulfill its statutory mission. Revision 2 was necessary to reflect organizational changes.

NUREG-0750 V24 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-September 1986. * Division of Publication Services (Post 870413). June 1987. 61pp. 8707020110. 41569:166.

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

NUREG-0750 V24 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1986.Pages 489-679. * Division of Publication Services (Post 870413). April 1987. 202pp. 8705120123. 40899:257.

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

NUREG-0750 V24 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1986.Pages 681-768. * Division of Publication Services (Post 870413). May 1987. 99pp. 8706040313. 41180:238.

See NUREG-0750,V24,N04 abstract.

NUREG-0750 V24 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1986.Pages 769-930. * Division of Publication Services (Post 870413). June 1987. 175pp. 8707090378. 41633:182.

See NUREG-0750,V24,N04 abstract.

NUREG-0781 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SOUTH TEXAS PROJECT,UNITS 1 AND 2.Docket Nos. 50-498 And 50-499. (Houston Lighting And Power Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). May 1987. 172pp. 8706240260. 41441:226.

The Safety Evaluation Report issued in April 1986 provided the results of the NRC staff's review of the Houston Lighting and Power Company's application for licenses to operate the South Texas Project. The facility consists of two pressurized water nuclear reactors located in Matagorda County, Texas. Supplement No. 1 issued in September 1986 updated the information contained in the Safety Evaluation Report and addressed the ACRS Report issued on June 10, 1986. Supplement No. 2 issued in January 1987 addressed and resolved some of the outstanding issues remaining after issuance of the Safety Evaluation Report and Supplement No. 1. This Supplement No. 3 also addresses and resolves some of the outstanding issues remaining after issuance of the SER and Supplements 1 and 2.

NUREG-0800 06.5.2 R2: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Proposed Revision 2 To Section 6.5.2, "Containment Spray As A Fission Product Cleanup System." For Comment. * Office of Nuclear Reactor Regulation, Director (Post 870411). April 1987. 80pp. 8704270036. 40682:016.

Proposed revision 2 to SRP Section 6.5.2 would incorporate changes in the requirements for containment spray chemical additive systems, and explicitly states computational models which had only appeared in references in previous revisions. The requirement for immediate initiation of caustic addition to the spray would be deleted, and the minimum pH to be achieved would be reduced from 8.5 to 7. If adopted, this revision would be required to be used for future plants, and would be optional for present licensees. The proposed revision is accompanied by a regulatory analysis and two supporting technical documents.

NUREG-0800 06.5.5 R0: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Proposed Revision 0 To New SRP Section 6.5.5, "Pressure Suppression Pools As Fission Product Clean-Up Systems." For Comment. * Office of Nuclear Reactor Regulation, Director (Post 870411). April 1987. 51pp. 8704280166. 40716:350.

Proposed new SRP Section 6.5.5 would provide acceptance criteria and review procedures to be used in assessing the role of pressure suppression pools as fission product cleanup systems following potential reactor accidents. A calculational model to account for drywell bypass is given, and minimum fission product decontamination factors are listed for use in instances in which no detailed calculations of pool scrubbing have been performed. The proposed section is accompanied by a regulatory analysis and a supporting technical report.

NUREG-0837 V06 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, October-December 1986. JANG,J.; COHEN,L. Region 1, Office of Director. April 1987. 324pp. 8704280116. 40712:118.

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 facility sites throughout the country for the fourth quarter of 1986.

NUREG-0904 S01: DRAFT SUPPLEMENT TO THE FINAL ENVIRONMENTAL STATEMENT RELATED TO THE DECOMMISSIONING OF THE RARE EARTHS FACILITY, WEST CHICAGO,ILLINOIS.Docket No. 40-2061.(Kerr-McGee) * Division of Fuel Cycle, Medical, Academic & Commercial Use Safety (Post 870413). June 1987. 475pp. 8707060327. 41584:239.

This Draft Supplement to the Final Environmental Statement is issued by the U.S. Nuclear Regulatory Commission in response to the Atomic Safety and Licensing Board's ruling that the staff must supplement the Final Environmental Statement in order to evaluate the impact of permanent disposal of the Kerr-McGee Rare Earths Facility wastes located at West Chicago, Illinois. The statement considers the Kerr-McGee preferred plan and various alternatives to the plan. The action proposed by the Commission is the renewal of the Kerr-McGee license to allow disposal of wastes onsite and for possession of the wastes under license for an indeterminate time. The license could be terminated at a later date if certain specified requirements were met.

NUREG-0936 V05 N04: NRC REGULATORY AGENDA.Quarterly Report,October-December 1986. * Division of Rules & Records (Post 870413). May 1987. 160pp. 8706150112. 41300:247.

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

NUREG-0936 V06 N01: NRC REGULATORY AGENDA.Quarterly Report,January-March 1987. * Division of Rules & Records (Post 870413). June 1987. 139pp. 8706300230. 41491:320.

See NUREG-0936,V05,N04 abstract.

NUREG-0940 V06 N01: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,January-March 1987. * Office of Enforcement (Post 870413). June 1987. 400pp. 8706240320. 41450:266.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January - March 1987) 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-0980 R03: NUCLEAR REGULATORY LEGISLATION. HOSPODOR,S. Office of the General Counsel. April 1987. 115pp. 8705290061. 41116:266.

NUREG-0980 is a compilation of nuclear regulatory legislation and other relevant material through the 99th Congress, 2nd Session. This compilation has been prepared for use as a resource document, which the NRC intends to update at the end of every Congress. Contents of NUREG-0980 include: The Atomic Energy Act of 1954, as amended; Energy Reorganization Act of 1974, as amended; Uranium Mill Tailings Radiation Control Act of 1978; Low-Level Radioactive Waste Policy Act; Nuclear Waste Policy Act of 1982; and NRC Authorization and Appropriations Acts. Other materials included are statutes and treaties on export licensing, nuclear non-proliferation, and environmental protection.

NUREG-1002 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BRAIDWOOD STATION,UNITS 1 AND 2.Docket Nos. 50-456 And 50-457.(Commonwealth Edison Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). May 1987. 35pp. 8706150208. 41300:212.

In November 1983, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1002) regarding the application filed by the Commonwealth Edison Company, as applicant and owner, for a license to operate Braidwood Station, Units 1 and 2 (Docket No. 50-456 and 50-457). The first supplement to NUREG-1002 was issued in September 1986; the second supplement to NUREG-1002 was issued in October 1986. This third supplement to NUREG-1002 reports the status of certain items that remained unresolved at the time Supplement 2 was published. The facility is located in Reed Township, Will County, Illinois.

NUREG-1021 R04: OPERATOR LICENSING EXAMINER STANDARDS. * Operator Licensing Branch. May 1987. 200pp. 8706030301. 41165:001.

The Operator Licensing Examiners Standards provide policy and guidance to NRC examiners and establishes the procedures and practices for examining and licensing of applicants for NRC operator licenses pursuant to Part 55 of Title 10 of the Code of Federal Regulations (10 CFR 55). It is intended to assist NRC examiners and facility licensees to understand the examination process better and to provide for equitable and consistent administration of examinations to all applicants by NRC examiners. This standard is not a substitute for the Operator Licensing Regulations. As appropriate, this standard will be periodically revised to accommodate comments and reflect new information or experience.

NUREG-1057 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BEAVER VALLEY POWER STATION,UNIT 2.Docket No. 50-412.(Duquesne Light Company,et al) * Division of Reactor Projects - I/II (Post 870411). May 1987. 176pp. 8706120079. 41186:104.

Supplement No. 5 to the Safety Evaluation Report for the application filed by Duquesne Light Company, et al., for license to operate the Beaver Valley Power Station, Unit 2 (Docket No. 50-412), located in Beaver County, Pennsylvania, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation of (1) additional information submitted by the applicants since Supplement No. 4 was issued, and (2) matters that the staff had under review when Supplement No. 4 was issued.

NUREG-1100 V03 ADD: BUDGET ESTIMATES.Fiscal Years 1988-1989. * Division of Budget & Analysis (Post 870413). May 1987. 44pp. 8706300181. 41491:060.

This report contains the fiscal year budget justifications to Congress. The budget provides estimates for salaries and expenses for fiscal years 1988-1989. This addendum is required due to the NRC reorganization of April 12, 1987.

NUREG-1122 S01: KNOWLEDGES AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS. Pressurized Water Reactors. * Office of Nuclear Reactor Regulation, Director (Post 870411). April 1987. 240pp. 8705290304. 41116:001.

This document catalogs roughly 5300 knowledges and abilities of reactor operators and senior reactor operators. It results from a reanalysis of a much larger job-task analysis data base compiled by the Institute of Nuclear Power Operations (INPO). Knowledges and abilities are cataloged for 45 major power plant systems and 38 emergency evolutions, grouped according to 11 fundamental safety functions (e.g., reactivity control and reactor coolant system inventory control). Supplemental pages have been added to conform to NUREG-1123, "Knowledges and Abilities Catalog for Nuclear Power Plant Operators: Boiling Water Reactors," September, 1986. A structured sampling procedure for both catalogs is under development by the Nuclear Regulatory Commission (NRC) and will be published as a companion document, "Examiners' Handbook for Developing Operator Licensing Examinations" (NUREG-1121). With appropriate sampling from these catalogs, operator licensing examinations having content validity can be developed. The examinations developed by using the catalogs and handbook will cover those topics listed under Title 10, "Code of Federal Regulations," Part 55.

NUREG-1125 V08: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1986. * ACRS - Advisory Committee on Reactor Safeguards. April 1987. 217pp. 8705200003. 40987:013.

This compilation contains 58 ACRS reports submitted to the Commission or to the Executive Director for Operations during the calendar year 1986. All reports have been made available to the public through the NRC Public Document Room and the U.S. Library of Congress. The reports are divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports alphabetized by project name and within project name by chronological order. Part 2 categorizes the reports by the most appropriate generic subject area and within subject area by chronological order.

NUREG-1145 V03: U.S. NUCLEAR REGULATORY COMMISSION 1986 ANNUAL REPORT. * Office of Administration & Resources Management, Director (Post 870413). June 1987. 266pp. 8707020359. 41567:001.

This report covers the major activities, events, decisions and planning that took place during fiscal year 1986 within the NRC or involving the NRC.

NUREG-1147 R01: SEISMIC SAFETY RESEARCH PROGRAM PLAN. * Division of Engineering (Post 870413). May 1987. 200pp. 8706240291. 41438:334.

This document presents a plan for seismic research to be performed by the Structural and Seismic Engineering Branch in the Office of Nuclear Regulatory Research. The plan describes the regulatory needs and related research necessary to address the following issues: uncertainties in seismic hazard, earthquakes larger than the design basis, seismic vulnerabilities, shifts in building frequency, piping design, and the adequacy of current criteria and methods. In addition to presenting current and proposed research within the NRC, the plan discusses research sponsored by other domestic and foreign sources.

NUREG-1166: FINAL ENVIRONMENTAL STATEMENT FOR DECOMMISSIONING HUMBOLDT BAY POWER PLANT,UNIT 3.Docket No. 50-133.(Pacific Gas And Electric Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). April 1987. 164pp. 8705190499. 40978:319.

The Final Environmental Statement contains the assessment of the environmental impact associated with decommissioning the Humboldt Bay Power Plant Unit 3 pursuant to the National Environmental Policy Act of 1969 (NEPA) and Title 10 of the Code of Federal Regulations, Part 51, as amended, of the Nuclear Regulatory Commission regulations. The proposed decommissioning would involve safe storage of the facility for about 30 years, after which the residual radioactivity would be removed so that the facility would be at levels of radioactivity acceptable for release of the facility to unrestricted access.

NUREG-1184 DRFT: INTEGRATED SAFETY ASSESSMENT REPORT,INTEGRATED SAFETY ASSESSMENT PROGRAM - MILLSTONE NUCLEAR POWER STATION,UNIT 1. Docket No. 50-245.(Northeast Nuclear Energy Co).Draft Report. * Office of Nuclear Reactor Regulation, Director (Post 870411). April 1987. 524pp. 8704270155. 40632:334.

The Integrated Safety Assessment Program (ISAP) was initiated in November 1984, by the U.S. Nuclear Regulatory Commission to conduct integrated assessments for operating nuclear power reactors. The integrated assessment is conducted on a plant-specific basis to evaluate all licensing actions, licensee initiated plant improvements and selected unresolved generic/safety issues to establish implementation schedules for each item. In addition, procedures will be established to allow for a periodic updating of the schedules to account for licensing issues that arise in the future. This report documents the review of Millstone Nuclear Power Station, Unit No. 1, operated by Northeast Nuclear Energy Company, which is one of two plants being reviewed under the pilot program for ISAP. This report indicates how 85 topics selected for review were addressed and presents the staff's recommendations regarding the corrective actions to resolve the 85 topics and other actions to enhance plant safety. The report is being issued in draft form to obtain comments from the licensee, nuclear safety experts, and the Advisory Committee for Reactor Safeguards. Once those comments have been resolved, the staff will present its positions, along with a long-term implementation schedule from the licensee, in the final version of this report.

NUREG-1214 R01: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. JOHNSON,M.R.; WATERMAN,D.K. Division of Inspection Programs (850212-870411). April 4, 1987. 69pp. 8704270378. 40704:218.

This report presents a history of Systematic Assessment of Licensee Performance (SALP) ratings for nuclear power plant facilities in operation and under construction. The SALP results are listed by NRC region in three sections: the most recent report, operating facilities, and facilities under construction. The historical data summary report has been prepared by the NRC Office of Inspection and Enforcement (IE). Information contained in this report has been updated to include those published SALP reports received before March 19, 1987.

NUREG-1230 DRFT FC: COMPENDIUM OF ECCS RESEARCH FOR REALISTIC LOCA ANALYSIS. Draft Report For Comment. * Division of Reactor & Plant Systems (Post 870413). April 1987. 1,348pp. 8705290033. 41101:238.

Emergency Core Cooling Systems (ECCS) are required on all light water reactors (LWRs) in the United States to provide cooling of the reactor core in the event of a break in the reactor piping. These accidents are called loss-of-coolant accidents (LOCA), and they range from small leaks to a postulated full break of the largest pipe in the reactor cooling system. Federal government regulations require that calculations of the LOCA be performed to show that the ECCS will maintain fuel rod cladding temperatures, cladding oxidation, and hydrogen production within certain limits. The Nuclear Regulatory Commission (NRC) and others have completed an extensive investigation of fuel rod behavior and ECCS performance. The technology has been advanced to the point that is now possible to make a realistic estimate of ECCS performance during a LOCA and to quantify the uncertainty of this calculation. This report serves as a general reference for ECCS research. The report (1) summarizes the understanding of LOCA phenomena in 1974, (2) reviews experimental and analytical programs developed to address the phenomena, (3) describes best-estimate computer codes developed by the NRC, (4) discusses the salient technical aspects of LOCA phenomena and our current understanding of them, (5) discusses probabilistic risk studies, and (6) examines the impact of research on the ECCS regulations.

NUREG-1235: TECHNICAL SPECIFICATIONS FOR CLINTON POWER STATION,UNIT 1. Docket No. 50-461.(Illinois Power Company) * Office of Nuclear Reactor Regulation, Director (Post 870411). April 1987. 562pp. 8705120069. 40897:147.

The Clinton Power Station, Unit No. 1 Technical Specifications were prepared by the U.S. Nuclear Regulatory Commission to set forth limits, operating conditions, and other requirements applicable to a nuclear reactor facility as set forth in Section 50.36 of 10 CFR 50 for the protection of the health and safety of the public.

NUREG-1239: ENVIRONMENTAL ASSESSMENT FOR RENEWAL OF SOURCE MATERIAL LICENSE NO. STB-401.Docket No. 40-6563.(Columbium-Tantalum Division,Mallinckrodt,Inc.) * Division of Fuel Cycle, Medical, Academic & Commercial Use Safety (Post 870413). April 1987. 69pp. 8705190502. 40980:267.

In response to an application for renewal of Materials License No. STB-401 for the Columbium-Tantalum Division of Mallinckrodt, Inc., St. Louis, Missouri, the NRC staff prepared this Environmental Assessment. The Environmental Assessment includes discussions of the need for the proposed renewal, alternatives to the action, and the environmental impacts of proposed action.

NUREG-1244: PLAN FOR INTEGRATING TECHNICAL ACTIVITIES WITHIN THE U.S. NRC AND ITS CONTRACTORS IN THE AREA OF THERMAL HYDRAULICS. * Office of Nuclear Regulatory Research, Director (Post 860720). April 1987. 42pp. 8705120074. 40896:162.

The Executive Director for Operations (EDO) directed the NRC staff to prepare a coordinated plan for the integration of technical activities within the agency and specified a number of issues to be addressed. This report summarizes the status of agency programs involved in thermal hydraulic research and proposes management methods to accomplish the EDO's directives.

NUREG-1245 V01: RADIOACTIVE WASTE MANAGEMENT RESEARCH PROGRAM PLAN FOR HIGH-LEVEL WASTE - 1987. * Division of Engineering (Post 870413). May 1987. 62pp. 8706120346. 41280:026.

The program of research described in this plan is intended to identify and resolve technical and scientific issues involved in the NRC's licensing and regulation of disposal systems intended to isolate high-level hazardous radioactive wastes (HLW) from the human environment. The Plan describes the program goals, discusses the research approach to be used, lays out peer review procedures, discusses the history and development of the high-level radioactive waste problem and the research effort to date and describes study objectives and research programs in the areas of: (a) materials and engineering, (b) hydrology and geochemistry, and (c) compliance with international waste management research programs. In addition, a proposed Earth Science Seismotectonic Research Program plan for radioactive waste facilities is appended.

NUREG-1258 DRFT: EVALUATION PROCEDURE FOR SIMULATION FACILITIES CERTIFIED UNDER 10CFR55.Draft Report. LAUGHERY,K.R.; PLOTT,C.; WACHTEL,J. Division of Human Factors Technology (851125-870411). March 1987. 141pp. 8704270073. 40682:096.

This document describes the process to be followed by the NRC for the inspection of simulation facilities certified by facility licensees in accordance with 10CFR55. Such inspections are divided into four major technical areas: performance testing; physical fidelity/human factors; control capabilities; and design, updating, modification and testing. Inspections will be performed by NRC staff with interdisciplinary skills including license examiner, operations specialist and human factors expert. Inspections may consist of off- site and/or on-site phases. The off-site phase consists of an examination of simulation facility documentation, and the identification of those operations that may be considered for use in on-site performance testing. In the on-site phase, the staff will work with the facility licensee to conduct a review of the four technical areas, and to evaluate the results of tests that are conducted. Findings will be based upon the staff's judgment of the degree of compliance of the simulation facility with 10CFR55 in terms of its suitability for the conduct of operating examinations.

NUREG-1259: TECHNICAL SPECIFICATIONS FOR BEAVER VALLEY POWER STATION, UNIT 2.Docket No. 50-412.(Duquesne Light Company) * Division of Reactor Projects - I/II (Post 870411). May 1987. 429pp. 8706120083. 41282:007.

The Beaver Valley Power Station, Unit 2, Technical Specifications were prepared by the U.S. Nuclear Regulatory Commission to set forth the limits, operating conditions, and other requirements applicable to a nuclear reactor facility as set forth in Section 50.36 of 10 CFR 50 for the protection of the health and safety of the public.

NUREG-1261: TECHNICAL SPECIFICATIONS FOR BRAIDWOOD STATION,UNITS 1 AND 2.Docket Nos. 50-456 And 50-457.(Commonwealth Edison Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). May 1987. 225pp. 8707140211. 41692:120.

The Braidwood Station, Unit Nos. 1 and 2, Technical Specifications were prepared by the U.S. Nuclear Regulatory Commission to set forth the limits, operating conditions, and other requirements applicable to a nuclear reactor facility as set forth in Section 50.36 of 10 CFR Part 50 for the protection of the health and safety of the public.

NUREG-1265: UNCERTAINTY PAPERS ON SEVERE ACCIDENT SOURCE TERMS. * Office of Nuclear Regulatory Research, Director (Post 860720). May 1987. 180pp. 8707140200. 41691:216.

An assessment of the severe accident source term technology was recently published by the NRC in NUREG-0956. State-of-the-art methods described in NUREG-0956 are now being used in risk assessments and as the basis for implementing the NRC's Severe Accident Policy Statement and its Safety Goal. Notwithstanding major advances in source term technology resulting from recent severe accident research programs, NUREG-0956 identified eight technical areas where uncertainties remain large and where our near-term research efforts should be focused. Individual programs within the severe accident research program are being adjusted to address these eight areas of uncertainty with a concentrated effort. To plan for these program changes, NRC research program managers have reviewed the nature of the uncertainties in their respective subject areas and prepared background papers. These background papers (or uncertainty papers) are presented in this report.

NUREG-1269: LOSS OF RESIDUAL HEAT REMOVAL SYSTEM.Diablo Canyon Unit 2, April 10,1987. CREWS,J.L.; TRAMMELL,C.M.; LYON,W.C.; et al. Region 5, Office of Director. June 1987. 105pp. 8707060049. 41584:040.

This report presents the findings of an NRC Augmented Inspection Team (AIT) investigation into the circumstances associated with the loss of residual heat removal (RHR) system capability for a period of approximately one and one-half hours at the Diablo Canyon, Unit 2 reactor facility on April 10, 1987. This event occurred while the Diablo Canyon, Unit 2, a pressurized water reactor, was shutdown with the reactor coolant system (RCS) water level drained to approximately mid-level of the hot leg piping. The reactor containment building equipment hatch was removed at the time of the event, and plant personnel were in the process of removing the primary side manways to gain access into the steam generator channel head areas. Thus, two fission product barriers were breached throughout the event. The RCS temperature increased from approximately 87 degrees F to bulk boiling conditions without RCS temperature indication available to the plant operators. The RCS was subsequently pressurized to approximately 7-10 psig. The NRC AIT members concluded that the Diablo Canyon, Unit 2 plant was, the time of the event, in a condition not previously analyzed by the NRC staff. The AIT findings from this event appear significant and generic to other pressurized water reactor facilities licensed by the NRC.

NUREG-1270 V01: INTERNATIONAL CODE ASSESSMENT AND APPLICATIONS PROGRAM. Annual Report. TING,P.; HANSON,R.; JENKS,R. Office of Nuclear Regulatory Research, Director (Post 860720). March 1987. 298pp. 8704270543. 40713:223.

The first ICAP Annual Report is devoted to coverage of program activities and accomplishments during the period between April 1985 and March 1987. The ICAP was organized by the Office of Nuclear Regulatory Research, United States Nuclear Regulatory Commission in 1985. The ICAP is an international cooperative reactor safety research program planned to provide independent assessment of the NRC computer codes developed for analysis of reactor transients and loss- of-coolant accidents.

NUREG-1271: GUIDELINES AND PROCEDURES FOR THE INTERNATIONAL CODE ASSESSMENT AND APPLICATIONS PROGRAM. TING,P.; BESSETTE,D.; HANSON,R. Division of Reactor & Plant Systems (Post 870413). April 1987. 80pp. 8705120086. 40896:073.

This document presents the guidelines and procedures by which the International Code Assessment and Applications Program (ICAP) will be conducted. The document summarizes the management structure of the program and the relationships between and responsibilities of the United States Nuclear Regulatory Commission (USNRC) and the international participants. The procedures for code maintenance and necessary documentation are described. Guidelines for the performance and documentation of code assessment studies are presented. An overview of an effort to quantify code uncertainty, which the ICAP supports, is included.

NUREG-1272: REPORT TO THE U.S. NUCLEAR REGULATORY COMMISSION ON ANALYSIS AND EVALUATION OF OPERATIONAL DATA - 1986. * Office for Analysis & Evaluation of Operational Data, Director. May 1987. 249pp. 8706160095. 41315:007.

This 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 the calendar year 1986. Comments and observations are provided on operating experience at nuclear power plants and other NRC licensees, including results from selected AEOD studies; summaries of abnormal occurrences involving U.S. nuclear plants; reviews of licensee event reports and their quality, reactor scram experience from 1984 to 1986, engineered safety features actuations, and the trends and patterns analysis program; and assessments of nonreactor and medical misadministration events. In addition, the report provides the year-end status of all recommendations included in AEOD studies, and listings of all AEOD reports issued from 1980 through 1986.

NUREG-1300: ENVIRONMENTAL STANDARD REVIEW PLAN FOR THE REVIEW OF A LICENSE APPLICATION FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY. PANGBURN,G.C. Office of Nuclear Material Safety & Safeguards,Director. April 1987. 251pp. 8705120121. 40899:001.

The Environmental Standard Review Plan (ESRP) (NUREG-1300) provides guidance to staff reviewers in the Office of Nuclear Material Safety and Safeguards who perform environmental reviews of Applicant's environmental reports prepared in support of license applications to construct and operate new low-level radioactive waste disposal facilities. The individual ESRP's which make up this document identify the information considered necessary to conduct the review, the purpose and scope of the review, the analysis procedure and evaluation, the formal inputs made to the Environmental Statement and the references considered appropriate for each review. By providing this information to the staff, the ESRP is intended to assure quality and uniformity of approach in individual reviews as well as compliance with the National Environmental Policy Act of 1969. In addition, the ESRP will make information about the environmental component of the licensing process more readily available and thereby improve the understanding of this process among the public, States and Regional Compacts and the regulated community.

NUREG/CP-0078: PROCEEDINGS OF THE SYMPOSIUM ON CHEMICAL PHENOMENA ASSOCIATED WITH RADIOACTIVITY RELEASES DURING SEVERE NUCLEAR PLANT ACCIDENTS. NIEMCZYK,S.J. American Chemical Society. June 1987. 700pp. 8706250118. 41465:079.

The Symposium on Chemical Phenomena Associated with Radioactivity Releases During Severe Nuclear Plant Accidents was held during the American Chemical Society National Meeting in Anaheim, California, September 9-12, 1986. The purpose of the symposium was to provide a forum for discussion of chemical processes and phenomena potentially occurring during severe nuclear reactor accidents. The symposium included an overview session designed to help place chemical issues in context in a severe accident perspective, as well as six sessions devoted to a variety of severe accident chemistry topics. Those topics included releases of radioactive and nonradioactive species from core materials in the reactor vessel; transport and behavior of those species in the reactor coolant system; transport and behavior of released species within the containment; releases during core-concrete interactions, as well as other aspects of such interactions; effects of engineered safety features and other plant systems; effects of extreme in-plant environments (such as high radiation fields and combustion zones); and potentially disruptive phenomena (such as hig.-pressure ejection of the melt from the vessel). The proceedings represent the compilation of all the papers presented at the symposium.

NUREG/CP-0086 V01: PROCEEDINGS OF THE 19TH DOE/NRC NUCLEAR AIR CLEANING CONFERENCE.Held In Seattle,Washington,August 18-21,1986. FIRST,M.W. Harvard School of Public Health, Boston, MA. May 1987. 650pp. 8706250268. CONF-860820. 41463:073.

This document contains the papers and the associated discussions of the 19th DOE/NRC Nuclear Air Cleaning Conference. Sessions were devoted to (1) fire, explosion and accident analysis, (2) adsorption and iodine retention, (3) filters and filter testing, (4) standards and regulation, (5) treatment of radon, krypton, tritium and carbon- 14, (6) ventilation and air cleaning in reactor operations, (7) disolver off-gas cleaning, (8) adsorber fires, (9) nuclear grade carbon testing, (10) sampling and monitoring, and (11) field test experience.

NUREG/CP-0086 V02: PROCEEDINGS OF THE 19TH DOE/NRC NUCLEAR AIR CLEANING CONFERENCE.Held In Seattle,Washington,August 18-21,1986. FIRST,M.W. Harvard School of Public Health, Boston, MA. May 1987. 635pp. 8706240014. CONF-860820. 41442:037.

See NUREG/CP-0086,V01 abstract.

NUREG/CP-0087: SUMMARY REPORT OF THE SYMPOSIUM ON SEISMIC AND GEOLOGIC SITING CRITERIA FOR NUCLEAR POWER PLANTS. CUMMINGS,G.E.; BERNREUTER,D.L.; MURRAY,R.C.; et al. Lawrence Livermore National Laboratory. June 1987. 49pp. 8706300026. UCID-21039. 41515:242.

This is a summary report of the Symposium on Seismic and Geologic Siting Criteria for Nuclear Power Plants held in Rockville, Maryland, on October 7-9, 1986. The purpose of the symposium was to provide a forum to delineate and examine the issues relating to a proposed revision to Appendix A of 10 CFR Part 100. Appendix A to Part 100 is the basic U.S. Nuclear Regulatory Commission regulation concerning seismic siting criteria for nuclear power plants. Some of the issues discussed at the symposium pertained to incorporating current probabilistic concepts into the regulation, decoupling the OBE and SSE, and revising the definition of basic concepts and parameters. The extent of any revision is yet to be determined.

NUREG/CR-2000 V06 N2: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of February 1987. * Oak Ridge National Laboratory. March 1987. 123pp. 8704270093. ORNL/NSIC-2000. 40678:297.

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-1061, "Instructions for Preparation of Data Entry Sheets for Licensee Event Reports." For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73 - Licensee Event Report System) which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022, "Licensee Event Report System - Description of Systems and Guidelines for Reporting," provides supporting guidance and information on the revised LER rule. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, keyword, and component vendor indexes follow the summaries. Vendors are those identified by the utility when the LER form is initiated; the keywords for the component, system, and general keyword indexes are assigned by the computer using correlation tables from the Sequence Coding and Search System.

NUREG/CR-2000 V06 N3: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of March 1987. * Oak Ridge National Laboratory. April 1987. 107pp. 8705190343. ORNL/NSIC-2000. 40975:324.

See NUREG/CR-2000,V06,N02 abstract.

NUREG/CR-2000 V06 N4: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of April 1987. * Oak Ridge National Laboratory. May 1987. 150pp. 8706230059. ORNL/NSIC-2000. 41421:297.

See NUREG/CR-2000,V06,N02 abstract.

NUREG/CR-2000 V06 N5: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of May 1987. * Oak Ridge National Laboratory. June 1987. 110pp. 8707090381. ORNL/NSIC-2000. 41633:074.

See NUREG/CR-2000,V06,N02 abstract.

NUREG/CR-2331 V06 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, July-September 1986. WEISS,A.J. Brookhaven National Laboratory. March 1987. 114pp. 8706120037. BNL-NUREG-51454. 41187:087.

This progress report will describe current activities and technical progress in the programs at Brookhaven National Laboratory sponsored by the Division of Accident Evaluation, Division of Engineering Technology, and Division of Risk Analysis & Operations of the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. The projects reported are the following: High Temperature Reactor Research, SSC Code Improvements, Thermal-Hydraulic Reactor Safety Experiments, Thermodynamic Core-Concrete Interaction Experiments and Analysis, Plant Analyzer, Code Assessment and Application, Code Maintenance (RAMONA-3B), MELCOR Verification and Benchmarking, Source Term Code Package Verification and Benchmarking, Uncertainty Analysis of the Source Term; Stress Corrosion Cracking of PWR Steam Generator Tubing, Soil-Structure Interaction Evaluation and Structural Benchmarks, Identification of Age Related Failure Modes; Application of HRA/PRA Results to Support Resolution of Generic Safety Issues Involving Human Performance, Protective Action Decisionmaking, Rebaselining of Risk for Zion, Containment Performance Design Objective, and Operational Safety Reliability Research.

NUREG/CR-2850 V05: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1983. BAKER,D.A.; PELOQUIN,R.A. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1987. 141pp. 8704280126. PNL-4221. 40713:082.

Population radiation dose commitments have been estimated from reported radionuclide releases from commercial power reactors operating during 1983. Fifty-year dose commitments from a one-year exposure were calculated from both liquid and atmospheric releases for four population groups (infant, child, teen-ager and adult) residing between 2 and 80 km from each of 52 sites. This report tabulates the results of these calculations, showing the dose commitments for both liquid and airborne pathways for each age group and organ. Also included for each of the sites is a histogram showing the fraction of the total population within 2 to 80 km around each site receiving various average dose commitments from the airborne pathways. The total dose commitments (from both liquid and airborne pathways) for each site ranged from a high of 45 person-rem to a low of 0.002 person-rem for the sites with plants operating throughout the year with an arithmetic mean of 3 person-rem. The total population dose for all sites was estimated at 170 person-rem for the 100 million people considered at risk. The site average individual dose commitment from all pathways ranged from a low of 1 x 10(-6) mrem to a high of 0.06 mrem. No attempt was made in this study to determine the maximum dose commitment received by any one individual from the radionuclides released at any of the sites.

NUREG/CR-3231: PIPE-TO-PIPE IMPACT PROGRAM. ALZHEIMER,J.M.; BAMPTON,M.C.; FRILEY,J.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1987. 104pp. 8706240152. PNL-5779. 41452:059.

The objective of this research was to determine the extent of damage that occurs when two pipes experience an impact event due to one whipping against the other. The research was conducted through experimental and analytical approaches. The former required the development of a specialized impact machine that could accelerate a whipping pipe with sufficient energy to cause failure of a target pipe that was heated and pressurized to Pressurized Water Reactor (PWR) conditions. Damage was measured in terms of crushing, bending, and failure. The results of the tests permitted the correlation between pipes of a certain size and the damage they could cause when impacting with a certain amount of known energy. These results were used to evaluate the pipe whip criteria in the Standard Review Plan 3.6.2-4. It was established that the criteria conditions did not fully represent the results obtained experimentally. An analysis procedure to model the pipe whip event was developed and used to establish the test matrix for the experimental program. This analytical procedure can also be used to predict deformation and rupture for postulated pipe whip scenarios.

NUREG/CR-3319 R01: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium. MCELROY,W.N. Hanford Engineering Development Laboratory. May 1987. 86pp. 8706160066. HEDL-TME 85-3. 41310:239.

This NRC physics-dosimetry compendium (Sections 1.0 through 4.0) is a collation of information and data developed from available research and commercial light water reactor vessel surveillance program (RVSP) documents and related surveillance capsule reports. The Section 4.0 data represents the results of the HEDL least-squares FERRET-SAND II Code reevaluation of exposure units and values for 47 PWR and BWR surveillance capsules. Using a consistent set of auxiliary data and dosimetry-adjusted reactor physics results, the revised fluence values for E greater than 1 MeV averaged 25% higher than the originally reported values. The range of fluence values (new/old) was from a low of 0.80 to a high of 2.38. These HEDL- derived FERRET-SAND II exposure parameter values are being used for NRC-supported HEDL and other PWR and BWR trend curve data development and testing studies, which support Revision 2 of Regulatory Guide 1.99. These trend curves are used by the utilities and by the NRC to account for neutron radiation damage in setting pressure/temperature limits, in analyzing fractures, and in predicting neutron-induced changes in reactor PV steel fracture toughness and embrittlement during the vessel's service life. The status of the development and application of new advancements in LWR reactor surveillance programs is discussed, such as cavity physics-dosimetry for improving the reliability of current and end-of-life (EOL) predictions on the metallurgical conditions of pressure vessels and their support structures.

NUREG/CR-3925 REV: SWIFT II SELF-TEACHING CURRICULUM.Illustrative Problems For The Sandia Waste-Isolation Flow And Transport Model For Fractured Media. REEVES,M.; WARD,D.S.; DAVIS,P.A.; et al. Sandia National Laboratories. January 1987. 794pp. 8704300036. SAND84-1586. 40749:199.

Several documents have been written describing SWIFT II, the most current version of the SWIFT (Sandia Waste Isolation Flow and Transport) Model. Reeves et al. [1986a], describes the theory and implementation, and Reeves et al. [1986b], describes the required input of data and parameters. Ward et al. [1984a], and [1984b], describe the comparison of the results from the SWIFT code with field data and other existing codes. This document is devoted to assisting the analyst who desires to use the SWIFT II code. The analyst is referred to the User's Manual for SWIFT II (Reeves et al. [1986b]) for detailed data input instructions. Eight examples are presented to illustrate the use of SWIFT II. The implementation of the numerical simulation of the physical problem is described for each example. For each problem, a listing of the input data and a microfiche listing of the output are provided.

NUREG/CR-3956: IN SITU TESTING OF THE SHIPPINGPORT ATOMIC POWER STATION ELECTRICAL CIRCUITS. DINSEL,M.R.; DONALDSON,M.R.; SOBERANO,F.T. EG&G Idaho, Inc. (subs. of EG&G, Inc.). April 1987. 49pp. 8706120172. EGG-2443. 41281:001.

This report discusses the results of electrical in situ testing of selected circuits and components at the Shippingport Atomic Power Station in Shippingport, Pennsylvania. Testing was performed by EG&G Idaho in support of the United States Nuclear Regulatory Commission (USNRC) Nuclear Plant Aging Research (NPAR) Program. The goal was to determine the extent of aging or degradation of various circuits from the original plant, and the two major coreplant upgrades (representing three distinct age groups), as well as to evaluate previously developed surveillance technology. The electrical testing was performed using the Electrical Circuit Characterization and Diagnostic (ECCAD) system developed by EG&G for the U.S. Department of Energy to use at TMI-2. Testing included measurements of voltage, effective series inductance, impedance, effective series resistance, dc resistance, insulation resistance and time domain reflectometry (TDR) parameters. The circuits evaluated included pressurizer heaters, control rod position indicator cables, and safety injection system motor operated valves. It is to be noted that the operability of these circuits was tested after several years had elapsed because plant operations had concluded at Shippingport. There was no need following plant shutdown to retain the circuits in working condition, so no effort was expended for that purpose. The in situ measurements and analysis of the data confirmed the effectiveness of the ECCAD system for detecting degradation of circuit connections and splices because of high resistance paths, with most of the problems caused by corrosion. Results indicate a correlation between the chronological age of circuits and circuit degradation.

NUREG/CR-4082 V05: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, April-September 1986. WILKOWSKI,G.M.; AHMAD,J.; BARNES,C.R.; et al. Battelle Memorial Institute, Columbus Laboratories. April 1987. 238pp. 8704270130. BMI-2120. 40675:326.

Presented herein is the Fifth Semiannual Report of the U.S. NRC's Degraded Piping Program - Phase II. The intent of this program is to experimentally validate and enhance available analytical methods for evaluating the mechanical behavior of nuclear power plant piping containing circumferentially-oriented defects. Fifty-one pipe experiments have been conducted to date. These and approximately 42 additional pipe experiments from other programs have been analyzed. In the analytical effort, a screening criterion has been developed to show when the net-section-collapse analysis is valid. This shows that even tough materials such as stainless steel can fail at less than net-section-collapse loads if the pipe diameter is sufficiently large. Numerous predictive J-estimation schemes have been evaluated and modified. A finite length surface cracked pipe estimation scheme has also been developed. Finite element analyses of specimens with welds suggest that the size of the weld relative to the specimen or structure size can affect the deformation J values. Supporting research efforts involve investigating geometry effects on J-R curves, as well as characterizing the material properties for each pipe tested.

NUREG/CR-4098: SEISMIC-FRAGILITY TESTS OF NEW AND ACCELERATED-AGED CLASS 1E BATTERY CELLS. BONZON,L.L. Sandia National Laboratories. JANIS,W.J.; BLACK,D.A.; et al. Ontario Hydro. January 1987. 135pp. 8705200231. SAND84-2631. 40984:326.

The seismic-fragility response of naturally-aged nuclear station safety-related batteries is of interest for two reasons: (1) to determine actual failure modes and thresholds and (2) to determine the validity of using the electrical capacity of individual cells as an indicator of the potential survivability of a battery given a seismic event. Prior reports in this series discussed the seismic-fragility tests and results for three specific naturally-aged cell types: 12- year old NCX-2250, 10-year old LCU-13, and 10-year old FHC-19. This report focuses on the complementary approach, namely the seismic- fragility response of accelerated-aged batteries. Of particular interest is the degree to which such approaches accurately reproduce the actual failure modes and thresholds. In these tests the significant aging effects observed, in terms of seismic survivability, were: embrittlement of cell cases, positive bus material and positive plate active material causing hardening and expansion of positive plates. The IEEE Standard 535 accelerated aging method successfully reproduced seismically significant aging effects in new cells but accelerated grid embrittlement an estimated five years beyond the conditional age of other components.

NUREG/CR-4161 V02: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 2:Tuff. DIDWALL,E.M. Lawrence Livermore National Laboratory. BENSON,S.M.; BINNALL,E.P.; et al. Lawrence Berkeley Laboratory. May 1987. 106pp. 8706030120. UCID-20092. 41144:242.

This report addresses critical parameters specific to a repository in tuff, using the Yucca Mountain tuffs of Nevada as the principal example. For the purposes of this report, a parameter is considered to be a physical property whose value helps determine the characteristics or behavior of a repository system. Parameters which are defined as critical are those essential to evaluate and/or monitor leakage of radionuclides from the repository and to evaluate the need for retrieval. The parameters are considered with respect to the disciplines of geomechanics, geology, hydrology, and geochemistry and are rank ordered in terms of importance. The specific role of each parameter, specific factors affecting the measurement of each parameter, and the interrelationships between the parameters are considered in detail.

NUREG/CR-4165: SEVERE ACCIDENT SEQUENCE ANALYSIS PROGRAM - ANTICIPATED TRANSIENT WITHOUT SCRAM SIMULATIONS FOR BROWNS FERRY NUCLEAR PLANT UNIT 1. DALLMAN,R.J.; GOTTULA,R.C.; HOLCOMB,E.E.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1987. 92pp. 8707060336. EGG-2379. 41584:147.

An analysis of five anticipated transients without scram (ATWS) was conducted at the Idaho National Engineering Laboratory (INEL). The five detailed deterministic simulations of postulated ATWS sequences were initiated from a main steam line isolation valve (MSIV) closure. The subject of the analysis was the Browns Ferry Nuclear Plant Unit 1, a boiling water reactor (BWR) of the BWR/4 product line with a Mark I containment. The simulations yielded insights to the possible consequences resulting from a MSIV closure ATWS. An evaluation of the effects of plant safety systems and operator actions on accident progression and mitigation is presented.

NUREG/CR-4219 V03 N2: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For April-September 1986. PUGH,C.E. Oak Ridge National Laboratory. December 1986. 234pp. 8706030097. ORNL/TM-9593. 41151:228.

The Heavy-Section Steel Technology (HSST) Program is an engineering research activity conducted by the Oak Ridge National Laboratory for the Nuclear Regulatory Commission. The program comprises studies related to all areas of the technology of material fabricated into thick-section primary-coolant containment systems of light-water-cooled nuclear power reactors. The investigation focuses on the behavior and structural integrity of steel pressure vessels containing cracklike flaws. Current work is organized into ten tasks: (1) program management, (2) fracture-methodology and analysis, (3) material characterization and properties, (4) environmentally assisted crack-growth studies, (5) crack-arrest technology, (6) irradiation effects studies, (7) cladding evaluations, (8) intermediate vessel tests and analysis, (9) thermal-shock technology, and (10) pressurized thermal-shock technology.

NUREG/CR-4300 V04 N1: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Progress Report, October 1986 - March 1987. HUTTON,P.H.; FRIESEL,M.A. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1987. 20pp. 8707100289. PNL-5511. 41663:077.

This report discusses technical progress for the period October 1986 to March 1987 for the NRC-sponsored research program concerned with "Acoustic Emission/Flaw Relationships for Inservice Monitoring of Nuclear Reactor Pressure Boundaries." Topics discussed included AE monitoring of primary piping during reactor operation, development of AE/IGSCC relationships, evaluation of the effects of crack growth rate on detection of the associated AE, validation of the AE signal identification method, and progress in developing an ASME Section XI Code appendix for continuous AE monitoring of pressure boundary components.

NUREG/CR-4307 V03: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.1986 Annual Report,October 1985 - September 1986. MCELROY,W.N. Hanford Engineering Development Laboratory. April 1987. 232pp. 8705120103. HEDL-TME 86-2. 40896:256.

The Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) has been established by the U.S. Nuclear Regulatory Commission (NRC) to improve, test, verify, and standardize the physics-dosimetry-metallurgy, damage correlation, and associated reactor analysis methods, procedures and data used to predict the integrated effect of neutron exposure to LWR pressure vessels and their support structures. A vigorous research effort attacking the same measurement and analysis problems exists worldwide, and strong cooperative links between the U.S. NRC-supported activities at HEDL, ORNL, NBS, and MEA and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Julich, Germany), and several United Kingdom Laboratories have been extended to a number of other countries and laboratories. These cooperative links are strengthened by the active membership of the scientific staff from many participating countries and laboratories in the ASTM E10 Committee on Nuclear Technology and Applications. Several subcommittees of ASTM E10 are responsible for the preparation of LWR surveillance standards. Results of FY-86 research by a number of LWR-PV-SDIP participants are reported in this progress report.

NUREG/CR-4330 V03: REVIEW OF LIGHT WATER REACTOR REGULATORY REQUIREMENTS. Assessment Of Selected Regulatory Requirements That May Have Marginal Importance To Risk:Postaccident Sampling System, Turbine Missiles,Combustible Gas Control,Charcoal Filters. SCOTT,W.B.; JAMISON,J.D.; STOETZEL,G.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1987. 134pp. 8706150009. PNL-5809. 41301:193.

In a study commissioned by the Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory (PNL) evaluated the costs and benefits of streamlining regulatory requirements in the areas of postaccident sampling systems, turbine missiles, combustible gas control, and impregnated charcoal filters. The basic framework of the analyses was that presented in the Regulatory Analysis Guidelines (NUREG/BR-0058) and in the Handbook for Value-Impact Assessment (NUREG/CR-3568). The effects of streamlined regulations were evaluated in terms of such factors as population dose and costs to industry and NRC. The results indicate that streamlining regulatory requirements in three of the four areas, i.e., postaccident sampling systems, turbine missiles, and combustible gas control, would have little impact on public risk. Streamlined regulatory requirements in the fourth area, impregnated charcoal filters, might increase public risk. Cost evaluations indicate substantial savings by lengthening the inspection interval for low-pressure turbine rotors. Small-to-moderate saving may be realized through postulated modifications to the postaccident sampling system requirements and to the combustible gas control requirements for hydrogen recombiners in inerted BWR Mark I and II containments. The results indicate that the use of impregnated charcoal filters is the most cost-effective method of radioiodine removal from building ventilation systems.

NUREG/CR-4469 V05: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report, April-September 1986. DOCTOR,S.R.; BATES,D.J.; DEFFENBAUGH,J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1987. 39pp. 8705280412. PNL-5711. 41097:183.

Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors (NDE Reliability) Program at the Pacific Northwest Laboratory was established by the Nuclear Regulatory Commission to determine the reliability of current inservice inspection (ISI) techniques and to develop recommendations that will ensure a suitably high inspection reliability. The objectives of this program include determining the reliability of ISI performed on the primary systems of commercial light-water reactors (LWRs); using probabilistic fracture mechanics analysis to determine the impact of NDE unreliability on system safety; and evaluating reliability improvements that can be achieved with improved and advanced technology. A final objective is to formulate recommended revisions to ASME Code and regulatory requirements, based on material properties, service conditions, and NDE uncertainties. The program scope is limited to ISI of the primary systems including piping, vessel, and other inspected components. This is a progress report covering the programmatic work from April 1986 through September 1986.

NUREG/CR-4469 V06: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report, October 1986 - March 1987. DOCTOR,S.R.; DEFFENBAUGH,J.; GOOD,M.S.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1987. 63pp. 8707080329. PNL-5711. 41621:250.

The Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors (NDE Reliability) Program at the Pacific Northwest Laboratory was established by the Nuclear Regulatory Commission to determine the reliability of current inservice inspection (ISI) techniques and to develop recommendations that will ensure a suitably high inspection reliability. The objectives of this program include determining the reliability of ISI performed on the primary systems of commercial light-water reactors (LWRs); using probabilistic fracture mechanics analysis to determine the impact of NDE unreliability on system safety; and evaluating reliability improvements that can be achieved with improved and advanced technology. A final objective is to formulate recommended revisions to ASME Code and Regulatory requirements, based on material properties, service conditions, and NDE uncertainties. The program scope is limited to ISI of the primary systems including the piping, vessel, and other inspected components. This is a progress report covering the programmatic work from October 1986 through March 1987.

NUREG/CR-4527 V01: AN EXPERIMENTAL INVESTIGATION OF INTERNALLY IGNITIED FIRES IN NUCLEAR POWER PLANT CONTROL CABINETS.Part 1:Cabinet Effects Tests. CHAVEZ,J.M. Sandia National Laboratories. April 1987. 117pp. 8707060099. SAND86-0336. 41583:285.

A series of full-scale cabinet fire tests was conducted by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission. The cabinet fire tests were prompted by the potential threat to the safety of a nuclear power plant by a cabinet fire in either the control room or in a switchgear type room. The purpose of these cabinet fire tests was to characterize the development and effects of internally ignited cabinet fires as a function of several parameters believed to most influence the burning process. A primary goal of this test program was to test representative and credible configurations and materials. This series of 22 cabinet fire tests demonstrated that fires in either benchboard or vertical cabinets with either IEEE-383 qualified cable or unqualified cable can be ignited and propagate. However, fires with IEEE-383 qualified cable do not propagate as rapidly nor to the extent that unqualified cable does. Futhermore, the results showed that the thermal environment in the test enclosure and adjacent cabinets is not severe enough to result in autoignition of other combustibles; although in some of the larger fires melting of plastic materials may occur. Smoke accumulation in the room appeared to be the most significant problem, as smoke obscured the view in the enclosure within minutes after ignition. Essentially, a cabinet fire can propagate within a single cabinet; however, for the conditions tested it does not appear that the fire poses a threat outside the burning cabinet except the resulting smoke.

NUREG/CR-4550 V05: ANALYSIS OF CORE DAMAGE FREQUENCY FROM INTERNAL EVENTS: SEQUOYAH,UNIT 1. BERTUCIO,R.C.; MOORE,D.L.; HELD,J.T.; et al. Sandia National Laboratories. February 1987. 424pp. 8705200221. SAND86-2084. 40985:101.

This document contains the accident sequence analyses for Sequoyah, Unit 1; one of the reference plants being examined as part of the NUREG-1150 effort by the Nuclear Regulatory Commission (NRC). NUREG-1150 will document the risk of a selected group of nuclear power plants. As part of that work, this report contains the overall core damage frequency estimate for Sequoyah, Unit 1, and the accompanying plant damage state frequencies. Sensitivity and uncertainty analyses provide additional insights regarding the dominant contributors to the Sequoyah core damage frequency estimate. The mean core damage frequency at Sequoyah was calculated to be 1.0E-4 per year. Small LOCAs with failure of emergency coolant recirculation account for over half of core damage frequency. Loss of component cooling water, leading to a reactor coolant pump seal LOCA is the next largest contributor, accounting for nearly one third of core damage frequency. Station blackout sequences account for 5% of core damage frequency. No other sequence types account for more than 5% of core damage frequency. The numerical results are influenced by modeling assumptions and data selection for issues such as coolant pump seal LOCA, common cause failure probabilities, and operator response to emergency conditions such as small LOCA and station blackout. The sensitivity studies explore the impact of alternate theories and data on these issues. The results of the uncertainty and sensitivity analyses should be considered before any future actions are taken based on this analysis.

NUREG/CR-4550 V06PT1: ANALYSIS OF CORE DAMAGE FREQUENCY FROM INTERNAL EVENTS:GRAND GULF,UNIT 1.Main Report. DROUIN,M.T.; LACHANCE,J.L.; SHAPIRO,B.J.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). April 1987. 500pp. 8707080284. SAND86-2084. 41618:163.

This document contains the accident sequence analyses for Grand Gulf Unit 1, one of the reference plants being examined as part of the NUREG-1150 effort by the Nuclear Regulatory Commission. NUREG- 1150 will document the risk of a selected group of nuclear power plants. As part of that work, this report contains the overall core damage frequency estimate for Grand Gulf Unit 1 and the accompanying plant damage state frequencies. Sensitivity and uncertainty analyses provide additional insights regarding the dominant contributors to the Grand Gulf core damage frequency estimate. The mean core damage frequency at Grand Gulf was calculated to be 2.9E-5. Station blackout type accidents (loss of all AC power accidents) were found to dominate the overall results. Anticipated transient without scram accidents were also found to be contributors. The numerical results are largely driven by common mode failure probability estimates and, to some extent, human error. Because of significant data uncertainties in these two areas, it is recommended that the results of the uncertainty and sensitivity analyses be considered before any future actions are taken based on this analysis. In particular, the single most dominant scenario may require a more detailed data search and analysis before actions are implemented on the basis of this scenario.

NUREG/CR-4550 V06PT2: ANALYSIS OF CORE DAMAGE FREQUENCY FROM INTERNAL EVENTS:GRAND GULF,UNIT 1.Appendices. DROUIN,M.T.; LACHANCE,J.L.; SHAPIRO,B.J.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). April 1987. 600pp. 8707080295. SAND86-2084. 41619:317.

See NUREG/CR-4550,V06,PT01 abstract.

NUREG/CR-4551 V2 DRF: EVALUATION OF SEVERE ACCIDENT RISKS AND THE POTENTIAL FOR RISK REDUCTION:SEQUOYAH POWER STATION,UNIT 1.Draft For Comment. BENJAMIN,A.S.; KUNSMAN,D.M.; LEWIS,S.R.; et al. Sandia National Laboratories. February 1987. 516pp. 8706120125. SAND86-1309. 41283:076.

The Severe Accident Risk Reduction Program (SARRP) has completed a rebaselining of the risks to the public from a particular pressurized water reactor with an ice-condenser containment (Sequoyah, Unit 1). Emphasis was placed on determining the magnitude and character of the uncertainties, rather than focusing on a point estimate. The risk-reduction potential of a set of proposed safety option backfits was also studied, and their costs and benefits were also evaluated. It was found that the risks from internal events are generally comparable to those evaluated in the Reactor Safety Study for a different pressurized water reactor, although the calculated uncertainty bands indicate that the risk could be higher or lower by as much as an order of magnitude. Principle sources of uncertainty include the modeling of common-cause failure in the component cooling water system, the characteristics of hydrogen generation and burning, and the possibility of fission product releases that bypass the ice condenser. Most of the postulated safety options do not appear to be cost effective for the Sequoyah plant; however, certain relatively inexpensive hardware and procedural changes to prevent core damage appear to be marginally cost effective. It should be noted that this work is based on a draft report of the ASEP study of core-damage frequency for Sequoyah. The final ASEP results have lower frequencies for station-blackout sequences. These final ASEP results will be incorporated in the final version of this study. The differences are not expected to cause significant changes in the conclusions of the report because risk is influenced by a variety of accident sequences, not just station blackout. This work supports the Nuclear Regulatory Commission's assessment of severe accidents in NUREG-1150.

NUREG/CR-4551 V3 PT1: EVALUATION OF SEVERE ACCIDENT RISKS AND THE POTENTIAL FOR RISK REDUCTION:PEACH BOTTOM,UNIT 2.Main Report.Draft For Comment. AMOS,C.N.; BENJAMIN,A.S.; BOYD,G.J.; et al. Sandia National Laboratories. April 1987. 200pp. 8707060116. SAND86-1309. 41593:078.

The Severe Accident Risk Reduction Program (SARRP) has completed a rebaselining of the risks to the public from a boiling water reactor with a Mark I containment (Peach Bottom, Unit 2). Emphasis was placed on determining the magnitude and character of the uncertainties, rather than focusing on a point estimate. The risk-reduction potential of a set of proposed safety option backfits was also studied, and their costs and benefits were also evaluated. It was found that the risks from internal events are generally low relative to previous studies; for example, most of the uncertainty range is lower than the point estimate of risk for the Peach Bottom plant in the Reactor Safety Study (RSS). However, certain unresolved issues cause the top of the uncertainty band to appear at a level that is comparable with the RSS point estimate. These issues include the modeling of the common-mode failures for the dc power system, the likelihood of offsite power recovery versus time during a station blackout, the probability of drywell failure resulting from meltthrough of the drywell shell, the magnitude of the fission product releases during core-concrete interactions, and the decontamination effectiveness of the reactor enclosure building. Most of the postulated safety options do not appear to be cost effective, although some based on changes to procedures or inexpensive hardware additions may be marginally cost effective. This draft for comment of the SARRP report for Peach Bottom does not include detailed technical appendices, which are still in preparation. The appendices will be issued under separate cover when completed. This work supports the Nuclear Regulatory Commission's assessment of severe accidents in NUREG-1150.

NUREG/CR-4551 V3 PT2: EVALUATION OF SEVERE ACCIDENT RISKS AND THE POTENTIAL FOR RISK REDUCTION:PEACH BOTTOM,UNIT 2.Appendices.Draft For Comment. AMOS,C.N.; BENJAMIN,A.S.; BOYD,G.J.; et al. Sandia National Laboratories. May 1987. 500pp. 8707060135. SAND86-1309. 41591:267.

See NUREG/CR-4551,V03,PT01 abstract.

NUREG/CR-4551 V4 DRF: EVALUATION OF SEVERE ACCIDENT RISKS AND THE POTENTIAL FOR RISK REDUCTION:GRAND GULF,UNIT 1.Draft For Comment. AMOS,C.N.; BENJAMIN,A.S.; BOYD,G.J.; et al. Sandia National Laboratories. April 1987. 750pp. 8707060161. SAND86-1309. 41589:226.

The Severe Accident Risk Reduction Program (SARRP) has completed a rebaselining of the risks to the public from a boiling water reactor with a Mark III containment (Grand Gulf, Unit 1). Emphasis was placed on determining the magnitude and character of the uncertainties, rather than focusing on a point estimate. The risk-reduction potential of a set of proposed safety option backfits was also studied, and their costs and benefits were also evaluated. It was found that the risks from internal events are generally low relative to previous studies; for example, most of the uncertainty range is lower than the point estimate of risk for the Peach Bottom plant in the Reactor Safety Study (RSS). However, certain unresolved issues cause the top of the uncertainty band to appear at a level that is comparable with the RSS point estimate. These issues include the diesel generator failure rate, iodine and cesium reevolution after vessel breach, and the possibility of reactor vessel pedestal failure caused by core debris attack. Some of the postulated safety options appear to be potentially cost effective for the Grand Gulf power plant, particularly when onsite accident costs are included in the evaluation of benefits. Principally these include procedural modifications and relatively inexpensive hardware additions to insure core cooling in the event of a station blackout. This work supports the Nuclear Regulatory Commission's assessment of severe accidents in NUREG-1150.

NUREG/CR-4583 V02: DEVELOPMENT AND VALIDATION OF A REAL-TIME SAFT-UT SYSTEM FOR THE INSPECTION OF LIGHT WATER REACTOR COMPONENTS.Annual Report,October 1984 - September 1985. DOCTOR,S.R.; HALL,T.E.; REID,L.D.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1987. 85pp. 8707080345. PNL-5822. 41621:314.

The Pacific Northwest Laboratory is working to design, fabricate, and evaluate a real-time flaw detection and characterization system based on the synthetic aperture focusing technique for ultrasonic testing (SAFT-UT). The system is designed to perform inservice inspection of light-water reactor components. Included objectives of this program for the Nuclear Regulatory Commission are to develop procedures for system calibration and field operation, to validate the system through laboratory and field inspections, and to generate an engineering data base to support ASME Code acceptance of the technology. This progress report covers the programmatic work from October 1984 through September 1985.

NUREG/CR-4615 V02: MODELING STUDY OF SOLUTE TRANSPORT IN THE UNSATURATED ZONE. Workshop Proceedings. SPRINGER,E.P.; FUENTES,H.R. Los Alamos National Laboratory. April 1987. 250pp. 8705190513. LA-10730-MS. 40976:191.

These proceedings include the technical papers, a panel summary report, and discussions held at the workshop on Modeling of Solute Transport in the Unsaturated Zone held June 19-20, 1986, at Los Alamos, New Mexico. The central focus of the workshop was the analysis of data collected by Los Alamos under agreement with the U.S. Nuclear Regulatory Commission on intermediate-scale caisson experiments. Five different modeling approaches were used. The purpose was to evaluate models for near-surface waste disposal of low- level radioactive wastes. The workshop was part of a larger study being conducted by Los Alamos on transport in the unsaturated zone under agreement with the U.S. Nuclear Regulatory Commission.

NUREG/CR-4617: ONSITE ASSESSMENTS OF THE EFFECTIVENESS AND IMPACTS OF UPGRADED EMERGENCY OPERATING PROCEDURES. MEYER,O.R.; BLACKMAN,H.S.; FORD,R.E.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). March 1987. 157pp. 8706220014. EGG-2456. 41415:226.

The implementation of upgraded emergency operating procedures (EOPs) by the nuclear utilities supports Three Mile Island (TMI) Action Plan Items I.C.1 and I.C.9. This is the final report of a research project directed at assessing the costs and benefits of the resultant EOPs. A dual methodology was used to assess effectiveness; experimental and onsite data collection. A simulator-based, laboratory experiment was conducted, which was designed to be sensitive to changes in operator effectiveness using function oriented EOPs versus event based EOPs. The onsite data collected were related to the effectiveness of upgraded EOPs as implemented with all attributes (e.g., function oriented, human factored, etc.). The acceptance of the upgraded EOPs by the control room operators was also measured. A cost/benefit ratio for upgraded EOPs was estimated per regulatory analysis guidelines. Summary conclusions are disclosed as to the ultimate benefits of the upgraded EOP program for the regulation of the safety of commercial nuclear power plants.

NUREG/CR-4623: IN-SITU STRESS MEASUREMENTS IN THE EARTH'S CRUST IN THE EASTERN UNITED STATES. RUNDLE,T.A.; SINGH,M.M.; BAKER,C.H. Engineers International, Inc. April 1987. 609pp. 8705120101. 40900:099.

In-situ stress measurements were made in three seismic areas in the Eastern United States, using the hydraulic fracturing technique. The areas covered were (i) Moodus, Connecticut, (ii) the Ramapo fault system, and (iii) the Central Virginia seismic zone. At each location, one borehole was drilled within the seismic zone and the second outside it, so as to compare the results obtained. No geologic interpretation of the data was made during this project.

NUREG/CR-4651: DEVELOPMENT OF RIPRAP DESIGN CRITERIA BY RIPRAP TESTING IN FLUMES.Phase I. ABT,S.R.; KHATTAK,M.S.; NELSON,J.D.; et al. Colorado State Univ., Fort Collins, CO. May 1987. 120pp. 8705280379. ORNL/TM-10100. 41100:230.

Flume studies were conducted in which riprap embankments were subjected to overtopping flows. Embankment slopes of 1, 2, 8, 10 and 20% were protected with riprap layers with median stone sizes of 1, 2, 4, 5 and/or 6 inches. Riprap design criteria for overtopping flows were developed in terms of unit discharge at failure, interstitial velocities and discharges through the riprap layer, resistance to flow over the riprap surface, potential impacts of the filter blanket on the riprap layer stability, and the effects of flow concentration on the riprap stability. The resulting riprap design criteria were compared to the Stephenson, the U.S. Army Corps of Engineers, the U.S. Bureau of Reclamation, and the Safety Factors methods for riprap stone design; the Leps relation for interstitial velocities through riprap; and the Anderson et al. and Corps of Engineers relationships for estimating Manning's n values for resistance to flow.

NUREG/CR-4653: GASPAR II - TECHNICAL REFERENCE AND USER GUIDE. STRENGE,D.L.; BANDER,T.J.; SOLDAT,J.K. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1987. 600pp. 8707090389. PNL-5907. 41631:179.

The Nuclear Regulatory Commission's GASPAR II computer program performs environmental dose analyses from releases of radioactive effluents from nuclear power plants into the atmosphere. The analyses estimate radiation dose to individuals and population groups from inhalation, ingestion, and external exposure pathways. The estimated doses provide information for National Environmental Policy Act evaluations and for determining compliance with Appendix I of 10 CFR 50. This report describes the mathematical models used in the GASPAR II computer program, instructs the user in preparing input to the program, and supplies detailed information on program structure and parameters used to modify the program.

NUREG/CR-4655: UNSATURATED FLOW AND TRANSPORT THROUGH FRACTURED ROCK RELATED TO HIGH-LEVEL WASTE REPOSITORIES.Final Report - Phase II. RASMUSSEN,T.C.; EVANS,D.D. Arizona, Univ. of, Tucson, AZ. May 1987. 499pp. 8706150101. 41298:131.

In response to high-level radioactive waste repository licensing needs of the U.S. Nuclear Regulatory Commission, this report examines and provides insights into physical characteristics and methodologies for performance assessment of candidate sites in unsaturated fractured rock. The focus is on the ability of the geologic medium surrounding an underground repository to isolate radionuclides from the accessible environment. Media of interest are consolidated rocks with variable fracturing, rock matrix permeabilities, contained water under negative pressure, and air-filled voids. Temperature gradients are also of interest. Studies present conceptual and theoretical considerations, physical and geochemical characterization, computer modeling techniques, and parameter estimation procedures. Radionuclide transport pathways are as solutes in groundwater and as vapor through air-filled voids. The latter may be important near a heat source. Water flow and solute transport properties of a rock matrix may be quantified using rock core analyses. Natural spatial variation dictates many samples. Observed fractures can be characterized and combined to form a fracture network for hydraulic and transport assessments. Unresolved problems include the relation of network hydraulic conductivity to fluid pressure and to scale. Once characterized, the matrix and fracture network can be coupled. Reliable performance assessment requires additional studies.

NUREG/CR-4663: CLOSEOUT OF IE BULLETIN 83-01:FAILURE OF REACTOR TRIP BREAKERS (WESTINGHOUSE DB-50) TO OPEN ON AUTOMATIC TRIP SIGNAL. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. Parameter, Inc. May 1987. 34pp. 8706240093. IEB-83-01. 41450:074.

During startup of Salem Unit 1 on February 25, 1983, the undervoltage trip attachments (UVTAs) of both Westinghouse DB-50 circuit breakers failed to open automatically upon receipt of a valid trip signal from the Reactor Protection System (RPS) on low-low steam generator level. The reactor was tripped manually about 30 seconds later. The manual trip used shunt relays installed in the DB-50 breakers. Similar failures of only one of a pair of DB-50 breakers in series had been reported to the NRC/IE and Westinghouse. Because of concern about the event at Salem Unit 1 and previous single failures, the NRC/IE issued Bulletin 83-01 on February 25, 1983. Licensees of operating pressurized water reactors with Westinghouse Type DB breakers having UVTAs were required to take specific actions. The bulletin was issued for information to all other nuclear power facilities. Evaluation of utility responses and NRC/IE inspection reports shows that the bulletin can be closed out per specific criteria for all of the 50 facilities to which it was issued for action. Malfunctions of the UVTAs were reported for only two sites other than Salem. There are no remaining areas of concern for this bulletin.

NUREG/CR-4664: CLOSEOUT OF IE BULLETIN 83-04:FAILURE OF THE UNDERVOLTAGE TRIP FUNCTION OF REACTOR TRIP BREAKERS. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. Parameter, Inc. May 1987. 31pp. 8706240323. IEB-83-04. 41439:216.

During shutdown of San Onofre units 2 and 3 on March 3 and 8, 1983, four General Electric (GE) Type AK-2 circuit breakers in the reactor protection systems (RPSs) failed to open on activation of the undervoltage trip coil during testing. Since issuance of IE Bulletin 79-09 April 17, 1979 on failures of GE Type AK-2 breakers, additional failures had been reported before the tests at San Onofre. Because of concern about continued failures of the subject breakers in RPSs, the NRC/IE issued IE Bulletin 83-04 on March 11, 1983. All licensees of operating pressurized water power reactors, except those with Westinghouse Type DB breakers, were required to take five specific actions. The bulletin was issued for information to all other nuclear power facilities. Evaluation of utility responses and NRC/IE inspection reports indicates that the bulletin can be closed out for all of the 50 facilities to which it was issued for action. Six plants had breakers which failed to operate satisfactorily during tests for bulletin requirements. There are no remaining areas of concern for this bulletin.

NUREG/CR-4674 V03: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1984,A STATUS REPORT.Main Report And Appendixes A And B. MINARICK,J.W.; HARRIS,J.D.; AUSTIN,P.N.; et al. Oak Ridge National Laboratory. May 1987. 133pp. 8706160050. ORNL/NOAC-242. 41309:298.

Forty-eight operational events, reported in Licensee Event Reports (LERs) and occurring at commercial light-water reactors during 1984, are considered to be precursors to potential severe core damage. These are described along with associated significance estimates, categorization, and subsequent analyses. This study is a continuation of the work presented in earlier volumes in this series, which evaluated the 1969-1981 and 1985 events. The report sequentially discusses (1) the general rationale for this study, (2) the program methods for review and documentation of operational events as precursors, (3) the use of the conditional probability of subsequent severe core damage estimates to rank precursor events, and (4) initial conclusions from the assessment of 1984 events.

NUREG/CR-4674 V04: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1984,A STATUS REPORT.Appendixes C,D And E. MINARICK,J.W.; HARRIS,J.D.; AUSTIN,P.N.; et al. Oak Ridge National Laboratory. May 1987. 534pp. 8706160086. ORNL/NOAC-232. 41312:196.

See NUREG/CR-4674,V03 abstract.

NUREG/CR-4679: QUANTITATIVE DATA ON THE FIRE BEHAVIOR OF COMBUSTIBLE MATERIALS FOUND IN NUCLEAR POWER PLANTS:A Literature Review. NOWLEN,S.P. Sandia National Laboratories. February 1987. 166pp. 8706120299. SAND86-0311. 41285:215.

This report presents the findings of a task in which currently available fire research literature was reviewed for quantitative data on the burning characteristics of combustible materials that are found in nuclear power plants. The materials considered for which quantitative data were available include cable insulation materials, flammable liquids, furniture, trash and general refuse, and wood and wood products. A total of 59 figures and tables, taken primarily from the referenced works, which summarize the available quantitative fire characterization information for these materials is presented.

NUREG/CR-4681: ENCLOSURE ENVIRONMENT CHARACTERIZATION TESTING FOR THE BASE LINE VALIDATION OF COMPUTER FIRE SIMULATION CODES. NOWLEN,S.P. Sandia National Laboratories. March 1987. 82pp. 8706150161. SAND86-1296. 41300:047.

This report describes a series of fire tests conducted under the direction of Sandia National Laboratories for the U.S. Nuclear Regulatory Commission. The primary purpose of these tests was to provide data against which to validate computer fire environment simulation models to be used in the analysis of nuclear power plant enclosure fire situations. Examples of the data gathered during three of the tests are presented, though the primary objective of this report is to provide a timely description of the test effort itself. These tests were conducted in an enclosure measuring 60x40x20 feet. All of the tests utilized forced ventilation conditions typical of nuclear power plant installations. A total of 22 tests using simple gas burner, heptane pool, methanol pool, and PMMA solid fires was conducted. Four of these tests were conducted with a full-scale control room mockup in place. Parameters varied during testing were fire intensity, enclosure ventilation rate, and fire location. Data gathered included air temperatures, air velocities, radiative and convective heat flux levels, optical smoke densities, inner and outer enclosure surface temperatures, enclosure surface heat flux levels, and gas concentrations within the enclosure in the exhaust stream.

NUREG/CR-4700 V2 DRF: CONTAINMENT EVENT ANALYSIS FOR POSTULATED SEVERE ACCIDENTS: SEQUOYAH POWER STATION,UNIT 1.Draft Report For Comment. BEHR,V.L.; BENJAMIN,A.S.; KUNSMAN,D.M.; et al. Sandia National Laboratories. February 1987. 208pp. 8705200197. SAND86-1135. 40986:165.

A study has been performed as part of the Severe Accident Risk Reduction Program (SARRP) to investigate the response of a particular pressurized water reactor with an ice-condenser containment (Sequoyah Unit 1) to postulated severe accidents. A detailed containment event tree for the Sequoyah plant has been devised to describe the various possible accident pathways that can lead to radioactive releases from containment. Data and analyses from a large number of NRC and industry-sponsored programs have been reviewed and used as a basis for quantifying the event tree, i.e., determining the likelihood of each pathway for a variety of accident sequence initiators. A generalized containment event tree code, called ENVTRE, has been developed to facilitate the quantification. The uncertainty in the results has been examined by performing the quantification three times, using a different set of input each time to represent the variation of opinion in the reactor safety community. In the so-called "central" estimate, the likelihood of early containment failure (occurring before or at the time of reactor vessel breach) was found to be high for station blackout sequences but very low for other accident sequence initiators. Unavailability of igniters and air return fans was the principal reason for the high failure probability for station blackouts. The analysis also showed that melting or bypass of the ice before or within a short time after vessel breach can be expected to occur with moderate to high likelihood during station blackouts and during sequences initiated by very small LOCAs with failure of emergency core cooling in the recirculation phase after success in the injection phase. This work supports NRC's assessment of severe accident risks to be published in NUREG-1150.

NUREG/CR-4700 V4 DRF: CONTAINMENT EVENT ANALYSIS FOR POSTULATED SEVERE ACCIDENTS: GRAND GULF NUCLEAR STATION,UNIT 1.Draft For Comment. AMOS,C.N.; KOLACZKOWSKI,A. Sandia National Laboratories. April 1987. 448pp. 8706120033. SAND86-1135. 41287:115.

A study has been performed as part of the Severe Accident Risk Reduction Program (SARRP) to investigate the response of a particular boiling water reactor with a Mark III containment (Grand Gulf Unit 1) to postulated severe accidents. A detailed containment event tree for the Grand Gulf plant has been developed to describe the various possible accident pathways that can lead to radioactive releases from containment. Data and analyses from a large number of NRC and industry-sponsored programs have been reviewed and used as a basis for quantifying the event tree, i.e., determining the likelihood of the pathways at each branch point for a variety of accident sequence initiators. A generalized containment event tree code, called EVNTRE, has been developed to facilitate the quantification. The uncertainty in the results has been examined by performing the quantification three times, using a different set of input each time to represent the variation of opinion in the reactor safety community. In the so- called "central" estimate, the likelihood of early containment failure (occurring before or within a short time after reactor vessel breach) was found to be significant because of the possibility of hydrogen deflagrations or detonations that can threaten containment integrity. However, uncertainties surrounding these issues could cause the early failure likelihood to be significantly lower than in the central estimate. Further, radioactive releases following containment failure would most likely be scrubbed by water, which would lower the overall source term. This work supports NRC's assessment of severe accident risks to be published in NUREG-1150.

NUREG/CR-4719: COOLABILITY OF STRATIFIED UO2 DEBRIS IN SODIUM WITH DOWNWARD HEAT REMOVAL.The D13 Experiment. OTTINGER,C.A.; MITCHELL,G.W.; REED,A.W. Sandia National Laboratories. March 1987. 67pp. 8706150126. SAND86-1043. 41301:127.

The LMFBR Debris Coolability Program at Sandia National Laboratories investigates the coolability of particle beds that may form following a severe accident involving core disassembly in a nuclear reactor. The D series experiments utilize fission heating of fully enriched UO2 particles submerged in sodium to realistically simulate decay heating. The D13 experiment is the first in the series to study the effects of bottom cooling of stratified debris, which could be provided in an actual accident condition by structural materials onto which the debris might settle. Additionally the D13 experiment was designed to achieve maximum temperatures in the debris approaching the melting point of UO2. The experiment was operated for over 40 hours and investigated downward heat removal at specific powers of 0.22 and 2.58 W/g. Channeled dryout in the debris was achieved at powers from 0.94 to 2.58 W/g. Maximum temperatures approaching 2700 degrees C were attained. Bottom heat removal was up to 750 kW/m2 as compared to 450 kW/m2 in the D10 experiment.

NUREG/CR-4726: EVALUATION OF PROTECTIVE ACTION RISKS. WITZIG,W.F.; SHILLENN,J.K. Pennsylvania State Univ., University Park, PA. June 1987. 200pp. 8707070524. 41600:074.

Risk of death and injury due to evacuation was estimated by studying 902 evacuations that occurred in the United States between January 1, 1973 and April 30, 1986. The risk of death due to evacuation is quite small. It was estimated to be equal to the risk of exposure to radiation doses of several hundred to about a thousand millirems. Key factors for a successful evacuation were found to include: an emergency plan, good communications and coordination, practice drills, and defined authority. Few evacuations used the emergency broadcasting system or warning sirens. Reports of panic and traffic jams were very few. Traffic flow was usually described as light to moderate. The speed of vehicles at the height of the evacuation was most commonly reported to be in the 25 to 40 mph range.

NUREG/CR-4739: RAMONA-3B CALCULATIONS FOR BROWNS FERRY ATWS STUDY. SAHA,P.; SLOVIK,G.C.; NEYMOTIN,L.Y. Brookhaven National Laboratory. February 1987. 120pp. 8705190595. BNL-NUREG-52021. 40976:071.

Several aspects of the Anticipated Transient Without Scram (ATWS) initiated by an inadvertent closure of all Main Steam Isolation Valves (MSIV) in a typical BWR/4 are analyzed in the report. The analysis is performed using the Brookhaven National Laboratory code, RAMONA-3B, which employs a three-dimensional neutron kinetics model coupled with a parallel-channel thermal-hydraulics in representing a Boiling Water Reactor (BWR) Core. Four different transient scenarios have been investigated: a) downcomer water level and reactor pressure control, b) manual control rod insertion transient, c) high pressure boil-off, and d) recirculation pump trip failure. Results of these calculations should provide better understanding of mitigative effects of operator actions during ATWS, thus helping in the development of adequate Emergency Procedure Guidelines (EPG) required for the BWR plant safety. A few unresolved questions subject to future investigations are also discussed.

NUREG/CR-4758: A RETRAN MODEL OF THE CALVERT CLIFFS-1 PRESSURIZED WATER REACTOR FOR ASSESSING THE SAFETY IMPLICATIONS OF CONTROL SYSTEMS. RENIER,J.A.; SMITH,O.L. Oak Ridge National Laboratory. March 1987. 116pp. 8706030092. ORNL/TM-10236. 41161:126.

The failure mode and effects analysis of Calvert Cliffs-1 identified sequences of events judged sufficiently complex to merit further analysis in detailed dynamic simulations. This report describes the RETRAN model developed for this purpose and the results obtained. The mathematical tool was RETRAN/Mod3, the latest version of widely used and extensively validated thermal-hydraulic production code. RETRAN2 is based on a first-principles methodology that treats two-phase flow with slip. Thermal equilibrium of phases is assumed except in the pressurizer, where non-equilibrium processes are important and special methodology is used. Heat transfer in solids is obtained from the conventional conduction equation. Point or 1-D kinetics is available for the reactor core. The fundamental methodology is supplemented with a broad list of process submodels that calculate heat transfer coefficients, fluid and metal state properties, choked flow, form and wall friction losses, and other parameters. Also supplied are component submodels for various types of valves and pumps, the latter of which incorporate four-quadrant characteristics for components in which two-phase or reverse flow may be expected, and heat versus flow curves for others. Extensive input allows the code to be highly particulized to a specific plant. The major investment in time and manpower occurs in setting up the base case; changes are comparatively easy to implement.

NUREG/CR-4765: MXS CROSS-SECTION PREPROCESSOR USER'S MANUAL. PARKER,F.; LUCK,L. Los Alamos National Laboratory. ISHIKAWA,M. FBR Safety Laboratory, Ibaraki Prefecture, Japan. March 1987. 58pp. 8705190577. LA-10856-M. 40966:217.

The MXS preprocessor has been designed to reduce the execution time of programs using isotopic cross-section data and to both reduce the execution time and improve the accuracy of shielding-factor interpolation in the SIMMER-II accident analysis program. MXS is a dual-purpose preprocessing code to (1) mix isotopes into materials and (2) fit analytic functions to the self-shielding data. The program uses the isotope microscopic neutron cross-section data from the CCCC standard interface file ISOTXS and the isotope Bondarenko self- shielding data from the CCCC standard interface file BRKOXS to generate cross-section and self-shielding data for materials. The materials may be a mixture of several isotopes. The self-shielding data for the materials may be the actual shielding factors or a set of coefficients for functions representing the background dependence of the shielding factors. A set of additional data is given to describe the functions necessary to interpolate the shielding factors over temperature.

NUREG/CR-4772: ACCIDENT SEQUENCE EVALUATION PROGRAM HUMAN RELIABILITY ANALYSIS PROCEDURE. SWAIN,A.D. Sandia National Laboratories. February 1987. 167pp. 8706120044. SAND86-1996. 41186:280.

This document presents a shortened version of the procedure, models, and data for human reliability analysis (HRA) which are presented in the "Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications" (NUREG/CR-1278, August 1983). This shortened version was prepared and tried out as part of the Accident Sequence Evaluation Program (ASEP) funded by the U.S. Nuclear Regulatory Commission and managed by Sandia National Laboratories. The intent of this new HRA procedure, called the "ASEP HRA Procedure," is to enable systems analysts, with minimal support from experts in human reliability analysis, to make estimates of the human error probabilities and other human performance characteristics that are sufficiently accurate for many probabilistic risk assessments. The ASEP HRA Procedure consists of a Pre-Accident Screening HRA, a Pre- Accident Nominal HRA, a Post-Accident Screening HRA, and a Post- Accident Nominal HRA. The procedure in this document includes changes made after tryout and evaluation of the procedure in four nuclear power plants by four different systems analysts and related personnel, including human reliability specialists. The changes consist of some additional explanatory material (including examples), and more detailed definitions of some of the terms.

NUREG/CR-4773: DESIGN FEATURES TO FACILITATE INTERNATIONAL SAFEGUARDS AT MIXED-OXIDE CONVERSION FACILITIES. HARMS,N.L.; ROBERTS,F.P. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1987. 44pp. 8706300252. PNL-5894. 41491:001.

This study for the Nuclear Regulatory Commission identifies and analyzes facility designs that can facilitate International Atomic Energy Agency safeguards for mixed plutonium uranium oxide (MOX) conversion plants. A baseline facility is defined and the implementation of safeguards is analyzed. Areas are identified for which special facility design considerations can facilitate IAEA inspections for timely detection of possible diversion of nuclear material. Design features are proposed to enhance inspection capabilities for verification of nuclear material flows and inventories.

NUREG/CR-4779: NEW DATA FOR AEROSOLS GENERATED BY RELEASES OF PRESSURIZED POWDERS AND SOLUTIONS IN STATIC AIR. BALLINGER,M.Y.; SUTTER,S.L.; HODGSON,W.H. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1987. 51pp. 8706240046. PNL-6065. 41448:248.

Safety assessments and environmental impact statements for nuclear fuel cycle facilities require an estimate of potential airborne releases. Aerosols generated by accidents are being investigated by Pacific Northwest Laboratory to develop radioactive source-term estimation methods. Experiments measuring the mass airborne and particle size distribution of aerosols produced by pressurized releases were run. Carbon dioxide was used to pressurize uranine solutions to 50, 250, and 500 psig before release. The mass airborne from these experiments was higher than for comparable air- pressurized systems, but not as great as expected based on the amount of gas dissolved in the liquid and the volume of liquid ejected from the release equipment. Flashing sprays of uranine at 60, 125, and 240 psig produced a much larger source term than all other pressurized releases performed under this program. Low-pressure releases of depleted uranium dioxide at 9, 17.5, and 24.5 psig provided data in the energy region between 3-m spills and 50-psig pressurized releases.

NUREG/CR-4783: ANALYSIS OF BALANCE-OF-PLANT REGULATORY ISSUES.Final Report. LAY,R.; ETTLINGER,L.; SETH,S.; et al. Mitre Corp. June 1987. 157pp. 8706160138. MTR-86W00213. 41310:071.

The MITRE Corporation, under contract to the U.S Nuclear Regulatory Commission (NRC), has examined certain regulatory and performance aspects of the conventional, or power conversion, side of nuclear plants, often referred to as the Balance-of-Plant. This report includes: MITRE's characterization and analysis of the Balance- of-Plant failures; a perspective on the safety significance of these failures; a description of current NRC activities and industry initiatives that have the potential to reduce these failures; and the formulation of a set of recommendations for NRC's consideration in addressing the safety issues raised by the Balance-of-Plant. In general, the Balance-of-Plant problems represent the most frequent reason for unanticipated plant shutdowns, and they compromise safety. The NRC needs to look at the total power plant facility as a system, and provide the same level of attention to reducing challenges to the plant safety systems as it does to responding and mitigating those challenges. MITRE believes that this can be accomplished within the context of NRC's present regulatory posture.

NUREG/CR-4800: SIGPI:A USER'S MANUAL FOR FAST COMPUTATION OF THE PROBABILISTIC PERFORMANCE OF COMPLEX SYSTEMS. PATENAUDE,C.J. Lawrence Livermore National Laboratory. May 1987. 75pp. 8706150086. UCID-20679. 41304:154.

The SIGPI program computes the probability of complex systems as defined by cut sets or other binary product sets. The SIGPI program uses two fast complementary methods of computing the probabilistic performance of complex systems: the II method and the sigma method. The II algorithm exploits the fact that carefully defined system components are often statistically independent conditional to the environment in which they are embedded. The sigma algorithm computes the probability of combinations of components produced by the II algorithm by disjointing and partitioning such components, thereby allowing the exact computation of performance. The program assumes input of up to three data types: cut set data in disjoint normal form, basic component probabilities for independent basic components, and/or mean and covariance data for statistically dependent basic components.

NUREG/CR-4802: AN EVALUATION OF TRAC-PF1/MOD1 COMPUTER CODE PERFORMANCE DURING POSTTEST SIMULATIONS OF SEMISCALE MOD-2C FEEDWATER LINE BREAK TRANSIENTS. HALL,D.G.; WATKINS,J.C. EG&G Idaho, Inc. (subs. of EG&G, Inc.). February 1987. 204pp. 8704270202. EGG-2486. 40681:173.

This report documents an evaluation of the TRAC-PF1/MOD1 reactor safety analysis computer code during computer simulations of feedwater line break transients. The experimental data base for the evaluation included the results of three bottom feedwater line break tests performed in the Semiscale Mod-2C test facility. The tests modeled 14.3% (S-FS-7), 50% (S-FS-11), and 100% (S-FS-6B) breaks. The test facility and the TRAC-PF1/MOD1 model used in the calculations are described. Evaluations of the accuracy of the calculations are presented in the form of comparisons of measured and calculated histories of selected parameters associated with the primary and secondary systems. In addition to evaluating accuracy of the code calculations, the computational performance of the code during the simulations was assessed. A conclusion was reached that the code is capable of making feedwater line break transient calculations efficiently, but there is room for significant improvements in the simulations that were performed. Recommendations are made for follow- on investigations to determine how to improve future feedwater line break calculations and for code improvements to make the code easier to use.

NUREG/CR-4814: SOURCES OF CORRELATION BETWEEN EXPERTS.Empirical Results From Two Extremes. MEYER,M.A.; BOOKER,J.M. Los Alamos National Laboratory. April 1987. 61pp. 8705190636. LA-10918-MS. 40977:256.

Through two studies, this report seeks to identify the sources of correlation, or dependence, between experts' estimates. Expert estimates are relied upon as sources of data whenever experimental data is lacking such as in risk analysis and reliability assessments. Correlation between experts is a problem in the elicitation and subsequent use of subjective estimates. Until now, there has been no data confirming sources of correlation, although the experts' background is commonly speculated to be one. Two different populations of experts were administered questions in their areas of expertise. Data on their professional backgrounds and means of solving the questions were elicited using techniques from educational psychology and ethnography. The results from both studies indicate that the way in which an expert solves the problem is the major source of correlation. The experts' background can not be shown to be an important source of correlation nor to influence his choice of method for problem solution. From these results, some recommendations are given for the elicitation and use of expert opinion.

NUREG/CR-4815: DEMONSTRATION TESTING OF A SURVEILLANCE ROBOT AT BROWNS FERRY NUCLEAR PLANT.Analysis Of Costs And Benefits. WHITE,J.R.; HARVEY,H.W.; FARNSTROM,K.A.; et al. Remote Technology Corp. March 1987. 82pp. 8704270352. 40704:117.

This report presents the results of an NRC project to determine whether robotics equipment can be cost effective in performing surveillance and inspection work at existing nuclear power plants. A mobile surveillance robot, called SURBOT, was developed by the Remote Technology Corporation (REMOTEC) to perform visual, sound, and radiation surveillance within rooms designated as radiologically hazardous. SURBOT was tested in the turbine building of the Browns Ferry Nuclear Plant (BFNP) by TVA personnel for a five-month period. The results showed that SURBOT obtains higher quality data and can perform more thorough surveillance within radiation areas than workers wearing protective clothing. SURBOT can be transferred between rooms without releasing contamination in the hallways using a portable enclosure. TVA has estimated that over 100 person-rem exposure and $100,000 operating costs can be saved annually at the BFNP using SURBOT for surveillance in 54 turbine and reactor building rooms. TVA recommendations for improving the function, reliability, and maintainability have been incorporated into a production model of SURBOT which is now commercially available from REMOTEC along with other types of mobile robots and manipulators.

NUREG/CR-4819 V01: AGING AND SERVICE WEAR OF SOLENOID-OPERATED VALVES USED IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS.Volume 1.Operating Experience And Failure Identification. BACANSKAS,V.P.; ROBERTS,G.C.; TOMAN,G.J.; et al. Oak Ridge National Laboratory. March 1987. 69pp. 8706030204. 41147:287.

An assessment of the types and uses of solenoid operated valves (SOV) in nuclear power plant safety-related service is provided. Through a description of each SOV's operation, combined with knowledge of nuclear power plant applications and operational occurrences, the significant stressors responsible for degradation of SOV performance are identified. A review of actual operating experience (failure data) leads to identification of potential nondestructive in-situ testing which, if properly developed, could provide the methodology for deterioration monitoring of SOVs. Recommendations are provided for continuation of the study into the test methodology development phase.

NUREG/CR-4821: REACTOR COOLANT PUMP SHAFT SEAL STABILITY DURING STATION BLACKOUT. RHODES,D.B.; HILL,R.C.; WENSEL,R.G. AECL, Chalk River Nuclear Laboratories. May 1987. 60pp. 8706120189. AECL-9342. 41187:201.

Results are presented from an investigation into the behavior of Reactor Coolant Pump shaft seals during a potential station blackout (loss of all ac power) at a nuclear power plant. The investigation assumes loss of cooling to the seals and focuses on the effect of high temperature on polymer seals located in the shaft seal assemblies, and the identification of parameters having the most influence on overall hydraulic seal performance. Predicted seal failure thresholds are presented for a range of station blackout conditions and shaft seal geometries.

NUREG/CR-4822: BROAD BAND SEISMIC DATA ANALYSIS.September 1984 - September 1986. CARTER,J.A.; BARSTOW,N.; SUTTON,G.H.; et al. Rondout Associates, Inc. April 1987. 100pp. 8704270184. 40682:234.

This report contains a detailed description of the SRNY station including its response function, description of the techniques and software used to analyze the data, and evaluations of both the station and processing methods. The station was evaluated through noise studies under both quiet and noisy conditions; its detection and location thresholds; and the frequency, duration, and causes of data loss. The report has been divided into three main sections. The first section gives a description of the broad band digital seismic station SRNY installed near the Rondout Associates, Incorporated (RAI) offices in Stone Ridge, NY. Included in the discussion are the system response functions and an analysis of the causes of data loss. The second section gives details of the data analysis methods used in studying broad band and array data. Although several methods have been studied, much of this section is devoted to the adaptive polarization method which has shown promise as a single station location tool. The final section examines the noise characteristics at SRNY during both quiet and noisy conditions and compares these levels to the Regional Seismic Test Network station RSNY in the Adirondack Mountain region of northern New York. The detection and location capabilities of SRNY are also described and single-station locations at SRNY, as well as RSNY, are compared to network locations.

NUREG/CR-4830: MELCOR VALIDATION AND VERIFICATION 1986 PAPERS. LEIGH,C.D. Sandia National Laboratories. March 1987. 223pp. 8706120166. SAND86-2689. 41281:049.

MELCOR validation and verification results from 1986 are presented. Results of comparisons to analytic solutions and experiments are included. The major areas tested in these comparisons are the control volume hydrodynamics and thermodynamics, the heat transfer and the aerosol behavior in MELCOR. A set of nine standard tests is included.

NUREG/CR-4841: FRACTURE EVALUATION OF SURFACE CRACKS EMBEDDED IN REACTOR VESSEL CLADDING.Unirradiated Bend Specimen Results. MCCABE,D.E. Materials Engineering Associates, Inc. May 1987. 70pp. 8705190631. MEA-2200. 40977:187.

The surface crack embedded in the clad layer of a reactor vessel has been identified as a critical fracture safety assessment condition relative to the pressurized thermal shock accident scenario. This project was initiated to determine the severity of such cracks experimentally, using irradiated material, and to identify the material property and stress conditions in the local region of the crack that are significant to the analysis. Bend bar tests provide the experimental simulation of the subject RPV surface crack. This report describes the initial investigation using unirradiated material, addresses the analysis techniques, and presents the findings indicated by the experimental results. Characterization of the irradiated material will be presented in a subsequent report.

NUREG/CR-4842: A STUDY OF NATURAL GLASS ANALOGUES AS APPLIED TO ALTERATION OF NUCLEAR WASTE GLASS. BYERS,C.D.; JERCINOVIC,M.J.; EWING,R.C. Argonne National Laboratory. February 1987. 165pp. 8705200253. ANL-86-46. 40984:161.

A two-part study was undertaken on the alteration of natural basalt and nuclear waste glasses. In the first part, the University of New Mexico characterized a wide variety of natural basaltic glasses with respect to reaction products, reaction kinetics, and geologic history. The important outcome of this study was a description of the process whereby natural glass alters to palagonite and authigenic materials, including clays and zeolites. In the second part, ANL performed laboratory tests to simulate the natural alteration process with a synthetic basaltic glass. For laboratory tests in which the glass samples were exposed to water vapor at high temperature ($>$ 90 degrees C), a close similarity was found in the alteration of natural and synthetic glasses. The same alteration process was found for a nuclear waste glass (SRL 165). It was concluded that both the natural alteration of basaltic glass and water-vapor laboratory tests can be used as an analogue to assess the performance of nuclear waste glass under potential repository conditions.

NUREG/CR-4845: AN ANALYSIS OF THE SEMISCALE MOD-2C S-NH-3 TEST USING THE TRAC-PF1 COMPUTER PROGRAM. DRISKELL,W.E.; KULLBERG,C.M. EG&G Idaho, Inc. (subs. of EG&G, Inc.). March 1987. 45pp. 8706030180. EGG-2496. 41144:203.

A calculation was performed using the TRAC-PF1/MOD1 computer program to simulate a small break, loss-of-coolant experiment where the high-pressure injection was not used to mitigate the fuel rod temperature excursion. This experiment, designated the S-NH-3 Test, simulated a 0.5% cold-leg break in a PWR and was one of a series of tests conducted in the Semiscale Mod-2C test facility. The primary purpose for doing the calculation was to evaluate the capability of the TRAC-PF1 code to calculate the thermal-hydraulic response observed in the experiment. The evaluation employs the comparison of selected code-calculated system responses with the test data. Conclusions and recommendations on improving the quality of the calculation are included.

NUREG/CR-4848: STEAM GENERATOR GROUP PROJECT.Annual Report - 1985. KURTZ,R.J.; LEWIS,M.; CLARK,R.A. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1987. 122pp. 8704270138. PNL-5771. 40681:051.

This report is a summary of the Steam Generator Group Project progress for 1985. Statistical analyses of data from nondestructive examination (NDE) round robins performed on the Surry generator are presented. Criteria are listed for selection of tube specimens to be removed from the generator for validation of the NDE round robin results. A sampling plan is described along with the initial steps taken to implement this plan. Special tooling fabricated for specimen removals is discussed. Removal of three 9-tube sections of tube sheet is presented. Preliminary results from destructive analysis of one section are reported. Validation activities were initiated by metallurgical evaluation of two tubes pulled from the hot leg tube sheet. Results from destructive measurement of maximum defect depth from these two tubes indicated that in situ bobbin coil eddy current measurements consistently undersized the defects.

NUREG/CR-4866: AN ASSESSMENT OF HYDROGEN GENERATION FOR THE PBF SEVERE FUEL DAMAGE SCOPING AND 1-1 TESTS. CRONENBERG,A.W.; MILLER,R.W.; OSETEK,D.J. EG&G Idaho, Inc. (subs. of EG&G, Inc.). April 1987. 95pp. 8706120168. EGG-2499. 41281:272.

An evaluation of zircaloy oxidation and hydrogen generation data is presented for the first two severe fuel damage (SFD) tests, SFD-ST and SFD 1-1, conducted in the PBF at the INEL. The report presents an assessment of data in terms of the influence of zircaloy melting on oxidation behavior and fuel bundle reconfiguration effects which may alter steam flow and hydrogen generation characteristics. A comparison of the H(2) generation and cladding thermocouple data indicates that a significant amount of hydrogen was produced after the initiation of zircaloy melt-induced fuel dissolution (greater than or equal to 2150 K). Posttest metallographic observations corroborate the trend of the on-line data. Analyses also indicate that essentially complete flow area blockage (> 98%) would be required to diminish steam flow through the degraded test bundle, reducing hydrogen production. Neither on-line data nor posttest examination of the SFD-ST and SFD 1-1 fuel bundles indicates that such extreme flow area blockages occurred. For the steam-rich SFD-ST experiment, UO(2) fuel oxidation was also observed, possibly accounting for approximately 20% of the total hydrogen production. Fuel oxidation has also been noted from retrieved TMI-2 core debris samples. Thus, oxidation of UO(2) to a hypostoichiometric condition may add to the total hydrogen burden for severe accidents.

NUREG/CR-4872: EXPERIMENTAL AND ANALYTICAL ASSESSMENT OF CIRCUMFERENTIALLY SURFACE-CRACKED PIPES UNDER BENDING. SCOTT,P.M.; AHMAD,J. Battelle Memorial Institute, Columbus Laboratories. April 1987. 167pp. 8705190623. BMI-2149. 40978:152.

This study was performed to assess the validity of various techniques to predict maximum loads for circumferentially surface- cracked pipes under bending. Experimental data were developed for both carbon steel and stainless steel pipes. Predictions of maximum loads were made using the net-section-collapse method, the IWB-3640 analysis procedures, and a newly developed finite-length surface- cracked pipe J-estimation method. The net-section-collapse method gave good maximum-load predictions for certain types of pipe. However, for pipes with large radius to thickness (R(m)/t) ratios and/or low toughness, this analysis method tended to overpredict the experimental maximum load. A plastic-zone screening criterion was developed to show when this method was valid and when elastic-plastic fracture mechanics should be used. The limit-load procedures embodied in IWB-3640 provide the desired underprediction of the failure stress. The average failure stress for the nine stainless steel base metal experiments was 61 percent higher than predicted by Table IWB-3641-1 and 23 percent higher than predicted by the Source Equations. For the three stainless steel flux weld experiments the predicted failure stresses were adjusted by a stress multiplier to account for the lower toughness of the flux welds. The average failure stress for the flux weld experiments was 78 percent higher than predicted by Table IWB-3641-5 and 39 percent higher than predicted by the Source Equations. Predictions from two versions of the new finite-length surface-cracked pipe J-estimation method were compared to experimental results. One version is for pipes with large R(m)/t ratios (SC.TNP) while the other is a more general approach (SC.TKP) where the large R(m)/t ratio restriction is relaxed. The results show that the SC.TNP method tends to overestimate the maximum loads by 15 percent on the average whereas the SC.TKP method tends to underpredict the maximum loads, as desired, by 32 percent.

NUREG/CR-4875: CHARACTERIZATION OF CRUSHED TUFF FOR THE EVALUATION OF THE FATE OF TRACERS IN TRANSPORT STUDIES IN THE UNSATURATED ZONE. POLZER,W.L.; FUENTES,H.R.; RAYMOND,R.; et al. Los Alamos National Laboratory. March 1987. 46pp. 8704280288. LA-10962-MS. 40693:324.

Results of field-scale (caisson) transport studies under unsaturated moisture and steady and nonsteady flow conditions indicate variability and a lack of conservation of mass in solute transport. The tuff materials used in that study were analyzed for the presence of tracers and of freshly precipitated material to help explain the variability and lack of conservation of mass. Selected tuff samples were characterized by neutron activation analysis for tracer identification, by x-ray diffraction for mineral identification, by petrographic analysis for identification of freshly precipitated material, and by x-ray fluorescence analysis for identification of major and trace elements. The results of these analyses indicate no obvious presence of freshly precipitated material that would retard tracer movement. The presence of the nonsorbing tracers (bromide and iodide) suggests the retention of these tracers in immobile water. The presence of sorbing and nonsorbing tracers on the tuff at some locations (even at the 415-cm depth) and not at others suggests variability in transport.

NUREG/CR-4877: ASSESSMENT OF DESIGN BASIS FOR LOAD-CARRYING CAPACITY OF WELD-OVERLAY REPAIRS. SCOTT,P.M. Battelle Memorial Institute, Columbus Laboratories. April 1987. 81pp. 8704280032. BMI-2150. 40709:171.

This study was conducted to assess the current load-carrying capacity design basis for weld-overlay repairs (WORs). Although not specifically addressed in it, the design of WORs is in the spirit of the ASME Boiler and Pressure Vessel Code, Section XI, Article IWB- 3640. NUREG-0313 Revision 2 provides guidance for the implementation of the procedures outlined in IWB-3640. However, neither of these documents specifies the values for diameter or thickness which are to be used in the WOR design analysis. Throughout this report we have used the combined thickness and diameter of the repaired cross section to calculate the membrane and bending stresses. The maximum stress from each of the four WOR pipe experiments conducted was significantly higher than that predicted by the IWB-3640 analysis for the design guidelines set forth in NUREG-0313 Revision 2 for a "Standard" overlay; the average failure stress was approximately 30 percent higher than the IWB-3640 predicted failure stresses. These values do not include the Code safety factors on stress. For a "Standard" overlay, the flaw size considered in the analysis is completely through the original pipe wall for the entire circumference of the pipe. Such an overlay would be suitable for long-term plant operation. If actual flaw dimensions were used in the design analysis, then in two of the four experiments the maximum stress was less than that predicted by the IWB-3640 Source Equations. The greatest difference was 8 percent. For the flaw sizes evaluated in this study, an overlay design based on actual flaw dimensions would be considered a "Limited Service" overlay, suitable only for short-term plant operation, not to exceed one fuel cycle.

NUREG/CR-4878: ANALYSIS OF EXPERIMENTS ON STAINLESS STEEL FLUX WELDS. Topical Report. WILKOWSKI,G.; AHMAD,J.; BRUST,F.; et al. Battelle Memorial Institute, Columbus Laboratories. April 1987. 187pp. 8705190355. BMI-2151. 40975:137.

This report describes experimental and analytical efforts to evaluate fracture of stainless steel flux-welded pipe. Seven pipe fracture experiments (four with through-wall circumferential cracks and three with circumferential internal surface cracks) were conducted at 550 degrees F (288 degrees C). Material characterization efforts involved laboratory specimen tests to assess specimen size effects, effects of solution-annealing, and crack-growth behavior in the HAZ, along the fusion line, and in the weld metal. Efforts involved assessing the net-section-collapse analysis, the plastic-zone screening criterion, inherent safety margins in the IWB-3640 flux weld analysis, through-wall-cracked pipe predictive J-estimation schemes for LBB analyses, n-factor J-R curves calculated from the pipe experiments for comparison to C(T) specimen results, and finite element analysis of C(T) specimens and one pipe experiment. This report also evaluates the technical significance of these results and their significance relative to licensing decisions.

NUREG/CR-4883: REVIEW OF RESEARCH ON UNCERTAINTIES IN ESTIMATES OF SOURCE TERMS FROM SEVERE ACCIDENTS IN NUCLEAR POWER PLANTS. KOUTS,H. Brookhaven National Laboratory. April 1987. 106pp. 8705190524. BNL-NUREG-52061. 40977:081.

A review has been undertaken by four panels of experts, of the sources of uncertainty in source terms from accidents to nuclear power plants as presented by the document NUREG-0956. These panels contained eminent scientists from the United States, the United Kingdom, and the Federal Republic of Germany. Separate reports by the panels provide detailed discussions and conclusions regarding the uncertainties and the NRC research programs for their resolution. An overall summary of the results of panel deliberation is also given.

NUREG/CR-4885: SEISMIC HAZARD CHARACTERIZATION OF THE EASTERN UNITED STATES.Comparative Evaluation Of The LLNL And EPRI Studies. BERNREUTER,D.L.; SAVY,J.B.; MENSING,R.W. Lawrence Livermore National Laboratory. May 1987. 289pp. 8706160074. UCID-20696. 41311:267.

In 1982, the Lawrence Livermore National Laboratory (LLNL) was funded by the U.S. Nuclear Regulatory Commission (NRC) to develop a methodology to characterize the seismic hazard for all sites of the eastern United States (EUS) east of the Rocky mountains. The utility- sponsored Electric Power Research Institute (EPRI) followed suit in late 1983 with a similar study. The LLNL methodology was applied at 10 test sites of the EUS and the results reported in 1985 (LLNL Report UCID-20421). The EPRI study was presented in a series of draft reports in 1985 under the project number P101-29. The purpose of this study was to help in understanding the reasons for differences in results between the LLNL and EPRI study. We first investigated possible differences in the theories and assumptions used to develop the hazard models and concluded that all inputs being equal, the two methods were essentially equivalent. We analyzed the various input parameters, their values and the way they were collected, and finally we performed sensitivity analysis. The three main differences were found to be (1) the lower bound magnitudes of integration, (2) the ground motion models, and (3) the fact that LLNL accounted for a local site correction and EPRI did not.

NUREG/CR-4889: ZIRCALOY-4 OXIDATION AT 1300 TO 2400 DEGREES C. PRATER,J.T.; COURTRIGHT,E.L. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1987. 41pp. 8705120087. PNL-6166. 40896:034.

The oxidation kinetics of Zircaloy-4 in steam have been extended to 2400 degrees C. The ZrO(2) and -Zr layers display parabolic growth behavior over the entire temperature range studied. A discontinuity in the oxidation kinetics at 1510 degrees C causes rates to increase above those previously established by the Baker-Just relationship. This increase coincides with the tetragonal-to-cubic phase transformation in ZrO(2-x). No additional discontinuity in the oxide growth rate was observed when the metal phase melted. The effects of temperature gradients were taken into account, and corrected values representative of near-isothermal conditions were computed. Oxide growth was also measured in various steam-hydrogen mixtures at 1565 degrees C and 1815 degrees C. Hydrogen concentrations up to 90 mol% had no effect on oxidation kinetics. The rate-controlling factor appears to be diffusion through the oxide layer. Finally, the oxidation kinetics of prereacted Zircaloy-15 mol% UO(2) were measured at 1400 to 2150 degrees C. The rates were comparable to those obtained for Zircaloy above 1500 degrees C.

NUREG/CR-4890: HEAT OF REACTION OF MOLTEN ZIRCONIUM WITH UO2. BRIMHALL,J.L.; PRATER,J.T. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1987. 32pp. 8705120091. PNL-6165. 40896:225.

The heat of reaction for the dissolution of UO(2) by molten zirconium has been measured at 1980 degrees C using differential thermal analysis. The dissolution of 25 wt% UO(2) by zirconium was determined to be an exothermic reaction with a heat release of approximately 20 kcal/mol UO(2). Experimental difficulties encountered at these high temperatures precluded a precise determination of the heat of reaction.

NUREG/CR-4894: A USER'S GUIDE TO THE NRC'S PIPING FRACTURE MECHANICS DATA BASE (PIFRAC). HISER,A.L.; CALLAHAN,G.M. Materials Engineering Associates, Inc. May 1987. 82pp. 8706030146. MEA-2210. 41164:278.

This Guide is the reference for use of the NRC's Piping Fracture Mechanics Data Base (PIFRAC), a computerized data base containing the material's property data for steels used in nuclear power plant piping. These data are for use by NRC regulators as required for structural integrity assessments of safety margins in nuclear piping. The data of primary utility in such assessments are fracture toughness (principally J-R curve) and tensile (stress-strain) data, but other characteristic data such as chemical composition and Charpy-V data are also provided where available. This Guide describes PIFRAC as presently configured. Revisions, updates, and corrections to PIFRAC will require revisions to this Guide.

NUREG/CR-4899: COMPONENT FRAGILITY RESEARCH PROGRAM.Phase I Component Prioritization. HOLMAN,G.S.; CHOU,C.K. Lawrence Livermore National Laboratory. June 1987. 177pp. 8706300007. UCID-21003. 41491:118.

Current probabilistic risk assessment (PRA) methods for nuclear power plants utilize seismic "fragilities" - probabilities of failure conditioned on the severity of seismic input motion - that are based largely on limited test data and on engineering judgment. Under the NRC Component Fragility Research Program (CFRP), the Lawrence Livermore National Laboratory (LLNL) has developed and demonstrated procedures for using test data to derive probabilistic fragility descriptions for mechanical and electrical components. As part of its CFRP activities, LLNL systematically identified and categorized components influencing plant safety in order to identify "candidate" components for future NRC testing. Plant systems relevant to safety were first identified; within each system components were then ranked according to their importance to overall system function and their anticipated seismic capacity. Highest priority for future testing was assigned to those "very important" components having "low" seismic capacity. This report describes the LLNL prioritization effort, which also included application of "high-level" qualification data as an alternate means of developing probabilistic fragility descriptions for PRA applications.

NUREG/CR-4901: EFFECTS FROM INFLUENT BOUNDARY CONDITIONS ON TRACER MIGRATION AND SPATIAL VARIABILITY FEATURES IN INTERMEDIATE-SCALE EXPERIMENTS. FUENTES,H.R.; POLZER,W.L.; SPRINGER,E.P. Los Alamos National Laboratory. April 1987. 131pp. 8706030186. LA-10981-MS. 41144:072.

In previous unsaturated transport studies at Los Alamos, dispersion coefficients were estimated to be higher close to the tracer source than at greater distances from the source. Injection of tracers through discrete influent outlets could have accounted for those higher dispersions. Also, a lack of conservation of mass of the tracers was observed and suspected to be due to spatial variability in transport. In the present study, experiments were performed under uniform influent (ponded) conditions in which breakthrough of tracers were monitored at four locations at each of four depths. All other conditions were similar to those of the unsaturated transport experiments. A comparison of results from these two sets of experiments indicates differences in the parameter estimates. Estimates were made for the dispersion coefficient and the retardation factor by the one-dimensional steady flow computer code, CFITIM. Estimates were also made for mass and for velocity and the dispersion coefficient by the method of moments. The dispersion coefficient decreased with depth under discrete influent application and increased with depth under ponded influent application. Differences in breakthroughs and in estimated parameters among locations at the same depth were observed under ponded influent application. Those differences indicate that there is a lack of conservation of mass as well as significant spatial variability across the experimental domain.

NUREG/CR-4903 V01: SELECTION OF EARTHQUAKE RESISTANT DESIGN CRITERIA FOR NUCLEAR POWER PLANTS - METHODOLOGY AND TECHNICAL CASES. Direct Empirical Scaling Of Response Spectral Amplitudes From Various Site And Earthquake Parameters. LEE,V.W.; TRIFUNAC,M.D. Structural & Earthquake Engineering Consultants. May 1987. 343pp. 8706120186. 41284:232.

New frequency dependent attenuation function of Fourier amplitude spectra of recorded strong earthquake ground acceleration has been developed. The iterative regression analyses assume simple functional forms to model the trends of the data and have sufficient flexibility to detect dependence of attenuation on source dimensions, depth and frequency of wave motion. It has been found that for distances less than about 100 km there is clear frequency dependent variation of attenuation functions, with high frequency amplitudes attenuating faster with distance. Our previous empirical scaling model for Pseudo Relative Velocity (PSV) spectrum amplitudes has been refined by introducing this new frequency dependent attenuation of amplitudes with distance. The new model also considers the depth of earthquake focus and the approximate characterization of the source size to compute the "representative" source to station distances in addition to all other scaling parameters used previously.

NUREG/CR-4903 V02: SELECTION OF EARTHQUAKE RESISTANT DESIGN CRITERIA FOR NUCLEAR POWER PLANTS - METHODOLOGY AND TECHNICAL CASES. Methods For Introduction Of Geological Data Into Characterization Of Active Faults And Seismicity And.... ANDERSON,J.G.; LEE,V.W.; TRIFUNAC,M.D. Structural & Earthquake Engineering Consultants. May 1987. 197pp. 8706120163. 41289:020.

This report reviews the physical and experimental bases for a quantitative relationship between earthquake occurrence rates and geological deformation rates. These relationships are well founded on mathematical statements of the elastic rebound theory, and well supported by observations. The concept of uniform risk spectra of Anderson and Trifunac (1977) is generalized to include (1) more refined description of earthquake source zones, (2) the uncertainties in estimating seismicity parameters a and b in log(10)N = a-bM, (3) the uncertainties in estimation of maximum earthquake size in each source zone, and (4) the most recent results on empirical scaling of strong motion amplitudes at a site. Examples of using the new NEQRISK program are presented and compared with the corresponding case studies of Anderson and Trifunac (1977). The organization of the computer program NEQRISK is also briefly described.

NUREG/CR-4903 V03: SELECTION OF EARTHQUAKE RESISTANT DESIGN CRITERIA FOR NUCLEAR POWER PLANTS - METHODOLOGY AND TECHNICAL CASES. Dislocation Models Of Near-Source Earthquake Ground Motion. A Review. * Structural & Earthquake Engineering Consultants. LUCO,J.E. California, Univ. of, San Diego, CA. May 1987. 177pp. 8706120158. 41288:203.

The solutions available for a number of theoretical fault models are examined in an attempt at establishing some of the expected characteristics of earthquake ground motion in the near-source region. In particular, solutions for two-dimensional anti-plane shear and plane-strain models as well as for three-dimensional fault models in full space, uniform half-space and layered half-space media are reviewed.

NUREG/CR-4912: DATING GROUND WATER AND THE EVALUATION OF REPOSITORIES FOR RADIOACTIVE WASTE. DAVIS,S.N.; MURPHY,E. Arizona, Univ. of, Tucson, AZ. April 1987. 197pp. 8705190537. 40977:315.

The age of ground water is the length of time that the water has been isolated from the atmospheric portion of the hydrologic cycle. It is a theoretical concept only, because all ground water to some extent is a mixture of waters of different ages. In simple systems, however, relative ages of ground water from different portions of an aquifer can be determined by different methods, and the dates obtained are commonly in accord with each other and reflect systematic increases of water ages in downgradient directions. At least nine independent methods can be used to approximate ground-water ages. The methods vary widely in precision but all give useful information. In complex ground-water systems, as many dating methods as possible should be used. Discordant "dates" will result which, when properly interpreted, will not give a single water age but will give valuable information concerning the hydrodynamics of the ground-water system. Dating methods which use isotopic and other geochemical techniques will read the actual history of the water and will give direct information on average ground-water conditions over long periods of time. If these periods exceed several hundred years, geochemical methods which use past conditions to predict the future are superior to hydrodynamic methods which use an extrapolation of short-term data to predict long-term hydrogeologic conditions.

NUREG/CR-4918 V01: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Annual Report,October 1985 - September 1986. SCHULZ,R.K.; RIDKY,R.W.; O'DONNELL,E.; et al. California, Univ. of, Berkeley, CA. April 1987. 30pp. 8705120092. 40896:196.

In the humid eastern part of the United States, trench covers have, in general, failed to prevent some of the incident precipitation from percolating downward to buried wastes. It is the purpose of the present work to investigate and demonstrate a procedure or technique that will control water infiltration to buried wastes regardless of above or below ground disposal. Results to date show the proposed procedure to be very promising and are applicable to shallow land burial as well as above ground disposal (e.g. Tumulus). In essence, the technique combines engineered or positive control of run-off, along with a vegetative cover, and is named "bioengineering management". To investigate control of infiltration, lysimeters are being used to make complete water balance measurements. The studies have been underway at the Maxey Flats, Kentucky, low-level waste disposal facility for the past three seasonal years. When the original Maxey Flats site closure procedure is followed, it is necessary to pump large amounts of water out of the lysimeters to prevent the water table from rising closer than 2 meters from the surface. Using the bioengineering management procedure, no pumping is required. As a result of the encouraging initial findings in the rather small-scale lysimeters at Maxey Flats, a large-scale facility for demonstration of the bioengineering management technique has been constructed at Beltsville, Maryland.

This facility is now operational with the demonstration and data collection underway.

NUREG/CR-4921: ENGINEERING AND QUALITY ASSURANCE COST FACTORS ASSOCIATED WITH NUCLEAR PLANT MODIFICATION. SMITH,M.H.; ZIEGLER,E.J. United Engineers & Constructors, Inc. (subs. of Raytheon Co.). April 1987. 59pp. 8706240192. 41452:001.

This study provides generic estimates of engineering and Quality Assurance (QA) costs based upon the development and analysis of new and existing data. The estimates are cost factors, not absolute dollar values, expressed as a percentage of the direct cost of implementing the plant modification. These factors vary significantly depending upon the work environment at the time of the modification. Generally the work environment refers to two groupings of plants: the first relates to requirements affecting structures and systems already in place, while the second relates to new construction requirements which may be applicable to future plants, plants under construction, and/or operating plants. The types of modifications this study addresses are the physical modifications to the structures/systems of nuclear power plants as opposed to analytical or procedural changes. The derived estimates, when multiplied by the direct cost (i.e., equipment, material, and installation labor) or installing specific structures, systems, and pieces of equipment at a nuclear power plant will generate a reasonable order-of-magnitude estimate of the engineering Q/A costs associated with a physical modification.

NUREG/CR-4922: STEAM SEPARATOR MODELING FOR VARIOUS NUCLEAR REACTOR TRANSIENTS. PAIK,C.Y.; MULLEN,G.; KNOESS,C.; et al. Massachusetts Institute of Technology, Cambridge, MA. June 1987. 227pp. 8706240352. EPRI NP-5272. 41439:289.

Experiments were performed using air and water on three different types of centrifugal separators: a cyclone as a generic separator, a Combustion Engineering type stationary swirl vane separator, and a Westinghouse type separator. The cyclone separator system has three stages of separation: first the cyclone, then a gravity separator, and finally a chevron plate separator. The other systems have only a centrifugal separator to isolate the effect of the primary separator. Experiments were also done in MIT blowdown rig, with and without a separator, using steam and water. The separators appear to perform well at flow rates well above the design values as long as the downcomer water level is not high. High downcomer water level rather than high flow rates appear to be the primary cause of degraded performance. Appreciable carry-over from the separator section of a steam generator occurs when the drain lines from three stage of separation are unable to carry off the liquid flow.

NUREG/CR-4928: DEGRADATION OF NUCLEAR PLANT TEMPERATURE SENSORS. HASHEMIAN,H.M.; PETERSEN,K.M.; KERLIN,T.W.; et al. Analysis & Measurement Services Corp. June 1987. 85pp. 8707020108. 41568:281.

A program was established and initial tests were performed to evaluate long term performance of resistance temperature detectors (RTDs) of the type used in U.S. nuclear power plants. The effort addressed the effect of aging on RTD calibration accuracy and response time. This Phase I effort included exposure of thirteen nuclear safety system grade RTD elements to simulated LWR temperatures. Full calibrations were performed initially and monthly, sensors were monitored and cross checked continuously during exposure, and response time tests were performed before and after exposure. Short term calibration drifts of as much as 1.8 degrees F (1 degree C) were observed. Response times were essentially unaffected by this testing. This program shows that there is a sound reason for concern about the accuracy of temperature measurements in nuclear power plants. These limited tests should be expanded in a Phase II program to involve more sensors and longer exposures to simulated LWR conditions in order to obtain statistically significant data. This data is needed to establish meaningful testing or replacement intervals for safety system RTDs. An important corollary benefit from this expanded program will be a better definition of achievable accuracies in RTD calibration. This report concludes a six-month Phase I project performed for the Nuclear Regulatory Commission under the SBIR program.

NUREG/CR-4936: AN INTEGRATED GEOLOGICAL,GEOPHYSICAL,AND GEOCHEMICAL INVESTIGATION OF THE MAJOR FRACTURES ON THE EAST SIDE OF THE NEW MADRID EARTHQUAKE ZONE. STEARNS,R.G.; REESMAN,A.L. Vanderbilt Univ., Nashville, TN. May 1987. 36pp. 8705290297. 41115:261.

The eastern edge of the Mississippi Valley graben (Reelfoot rift) is a series of offset segments marked by offset "ridges" of gravity anomaly. The most prominent offsetting fault is the Dyersburg line that is at least 60 miles long and cuts completely through the graben. The Dyersburg line is traced by earthquakes, the pattern of the geothermal gradient, offset gravity anomalies, and surface linears. Composition of ground water from the Cretaceous McNairy Formation, the Eocene Wilcox and Claiborne formations, and Holocene alluvium are all significant for the structure of the rift. Trends for some chemical species are parallel to the rift or the Dyersburg line (and probably the pre-Cretaceous Pascola arch). Barium and lithium show parallelism to both; strontium isotope ratios trend parallel to the rift whereas carbon isotope ratios trend parallel to the Pascola arch. Upward leakage along faults of mineralized and high pressure water from the McNairy is the likely explanation for localized occurrence of mineralized water in younger aquifers. The best example is chloride water in Holocene alluvium on the Dyersburg line.

NUREG/CR-4938: OCCUPATIONAL RADIATION EXPOSURES ASSOCIATED WITH ALTERNATIVE METHODS OF LOW-LEVEL WASTE DISPOSAL. HERRINGTON,W.N.; HARTY,R.; MERWIN,S.E. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1987. 110pp. 8706240313. PNL-6217. 41450:176.

The Low-Level Radioactive Waste Policy Amendments (LLRWPA) Act of 1985 mandates that the U.S. Nuclear Regulatory Commission (NRC), in consultation with states and other interested parties, identify disposal methods other than shallow land burial (SLB), the method currently used at the three low-level waste (LLW) disposal sites operating in the United States. We compared projected occupational exposures associated with the SLB method and five alternative disposal methods, including below ground vaults (BGV), above ground vaults (AGV), earth mounded concrete bunkers (EMCB), augured holes (AH), and mined cavities (MC). MC facilities were studied in less detail because this disposal method is not being actively considered. Reference facility designs and a list of probable tasks required at each site for receiving and disposing of the low-level waste were developed. Each task was analyzed by worker requirements, time requirements, distances between workers and the waste, and exposure rates at those distances. This information was used to estimate the dose received by workers during disposal of four types of waste packaging: drums, wood boxes, resin liners, and dumpsters. The results of this study suggest that, of the methods studied in detail, occupational dose equivalents would be highest for the EMCB method (1.81 person-mrem/m(3) of waste disposed). The lowest occupational dose equivalents would occur for the AH method (1.29 person-mrem/m(3)). Projected occupational dose equivalents for SLB, BGV, and AGV disposal methods are 1.38, 1.47, and 1.61 person-mrem/m(3),respectively.

NUREG/CR-4950 V01: THE SHORELINE ENVIRONMENT ATMOSPHERIC DISPERSION EXPERIMENT (SEADEX).Experiment Description. CANTRELL,B.K.; JOHNSON,W.B.; MORLEY,B.M.; et al. SRI International. June 1987. 40pp. 8707130162. 41684:136.

The SEADEX atmospheric dispersion field study was conducted during the period May-June 8, 1982, in northeastern Wisconsin, in the vicinity of the Kewaunee Power Plant on the western shore of Lake Michigan. The specific objectives of SEADEX were to characterize (1) the atmospheric dispersion and (2) the meteorological conditions influencing this dispersion as completely as possible during the test period. This field study included a series of controlled tracer tests utilizing state-of-the-art tracer measurement technology to determine horizontal and vertical dispersion over both land and water. Extensive meteorological measurements were obtained to thoroughly characterize the three-dimensional structure of the atmospheric boundary layer controlling the dispersion process. This volume describes the experimental design for, and conduct of, the study.

NUREG/CR-4959: PERFORMANCE TESTING OF EXTREMITY DOSIMETERS. HARTY,R.; REECE,W.D.; HOOKER,C.D. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1987. 65pp. 8707020364. PNL-6218. 41566:202.

The Health Physics Society Standing Committee (HPSSC) Working Group on Performance Testing of Extremity Dosimeters has issued a draft of a proposed standard for extremity dosimeters. The draft standard proposes methods to be used for testing extremity dosimetry systems and the performance criterion used to determine compliance. This study evaluates the draft standard's proposed performance criterion (absolute value of B + S less than or equ-' 0.35, where B is the bias and S is the standard deviation) ac ' the performance of extremity dosimeter processors. Twe... one types of extremity dosimeters from 11 processors were irradiated by the Pacific Northwest Laboratory (PNL) to specific dose levels in one or more of seven categories. The processors evaluated the doses and returned the results to PNL for analysis. Approximately 60% of the dosimeters met the performance criterion. Two-thirds of the remaining dosimeters had large biases (ranging from 0.25 to 0.80) but small standard deviations (less than 0.15). Recommendations to improve the biases include providing calibrations (using appropriate irradiation standards and phantoms) followed by another set of performance tests, as well as visiting processors to identify other possible sources of error. It is further recommended that the draft standard be re-evaluated to ensure that it is appropriate for the performance testing of extremity dosimeters.

NUREG/CR-4964: UPDATE OF TABLE S-3 NONRADIOLOGICAL ENVIRONMENTAL PARAMETERS FOR A REFERENCE LIGHT-WATER REACTOR.Uranium Mining,Milling And Enrichment. HABEGGER,L.J.; CARSTEA,D.D.; OPELKA,J.H. Argonne National Laboratory. June 1987. 68pp. 8707010655. ANL/EES-TM-332. 41545:268.

In 1974, Table S-3 of the report "Environmental Survey of the Uranium Fuel Cycle" was published as a technical basis for consideration of the environmental effects of the uranium fuel cycle supporting operation of light-water reactors. A reference reactor cooled with light, or ordinary, water was established to reduce the burden on the Nuclear Regulatory Commission (NRC) staff, reactor license applicants, and other interested persons by removing the necessity to relitigate the environmental effects attributable to the fuel cycle, effects that are not within an applicant's control, in every individual reactor licensing proceeding. In a 1984 evaluation of a license application, it was demonstrated that the Table S-3 estimate of annual effluent of coal particulates is larger, possibly by as much as a factor of 100, than actual current values. Partially as a result of this evaluation, the NRC initiated a study to update all of the major nonradiological values in Table S-3. The results of the study are documented in this update. The report evaluates only the mining, milling, and isotopic-enrichment components of the fuel cycle's environmental parameters since these are the areas in which the greatest changes from the original study could be anticipated.

Secondary Report Number Index

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

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NUREG-1145 V03: U.S. NUCLEAR REGULATORY COMMISSION 1986 ANNUAL REPORT.
DIVISION OF PUBLICATION SERVICES (POST 870413)
NUREG-0304 V12 N01: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1987,January-March.
NUREG-0540 V09 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. February 1-28,1987.
NUREG-0540 V09 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. March 1-31,1987.
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NUREG-1100 V03 ADD: BUDGET ESTIMATES.Fiscal Years 1988-1989.
DIVISION OF COMPUTER & TELECOMMUNICATIONS SERVICES (POST 870413)
NUREG-0020 V10 N11: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of October 31,1986.(Gray Book I)
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OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
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NUREG-1300: ENVIRONMENTAL STANDARD REVIEW PLAN FOR THE REVIEW OF A LICENSE APPLICATION FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY.
DIVISION OF FUEL CYCLE, MEDICAL, ACADEMIC & COMMERCIAL USE SAFETY (POST
NUREG-0904 S01: DRAFT SUPPLEMENT TO THE FINAL ENVIRONMENTAL STATEMENT RELATED TO THE DECOMMISSIONING OF THE RARE EARTHS FACILITY, WEST CHICAGO,ILLINOIS.Docket No. 40-2061.(Kerr-McGee)
NUREG-1239: ENVIRONMENTAL ASSESSMENT FOR RENEWAL OF SOURCE MATERIAL LICENSE NO. STB-401.Docket No. 40-6563.(Columbium-Tantalum Division,Mallinckrodt,Inc.)

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NUREG-0386 D04 R05: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.July 1972 - September 1986.
NUREG-0980 R03: NUCLEAR REGULATORY LEGISLATION.

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NUREG-1265: UNCERTAINTY PAPERS ON SEVERE ACCIDENT SOURCE TERMS.
NUREG-1270 V01: INTERNATIONAL CODE ASSESSMENT AND APPLICATIONS PROGRAM. Annual Report.
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NUREG-1245 V01: RADIOACTIVE WASTE MANAGEMENT RESEARCH PROGRAM PLAN FOR HIGH-LEVEL WASTE - 1987.
DIVISION OF REACTOR & PLANT SYSTEMS (POST 870413)
NUREG-1230 DRFT FC: COMPENDIUM OF ECCS RESEARCH FOR REALISTIC LOCA ANALYSIS. Draft Report For Comment.
NUREG-1271: GUIDELINES AND PROCEDURES FOR THE INTERNATIONAL CODE ASSESSMENT AND APPLICATIONS PROGRAM.

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NUREG-0332: POTENTIAL HEALTH AND ENVIRONMENTAL IMPACTS ATTRIBUTABLE TO THE NUCLEAR AND COAL FUEL CYCLES.Final Report.
NUREG-0800 06.5.2 R2: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Proposed Revision 2 To Section 6.5.2, "Containment Spray As A Fission Product Cleanup System." For Comment.
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NUREG-1122 S01: KNOWLEDGES AND SKILLS CATALOG FOR NUCLEAR POWER PLANT OPERATIONS.Pressurized Water Reactors.
NUREG-1184 DRFT: INTEGRATED SAFETY ASSESSMENT REPORT,INTEGRATED SAFETY ASSESSMENT PROGRAM - MILLSTONE NUCLEAR POWER STATION,UNIT 1. Docket No. 50-245.(Northeast Nuclear Energy Co).Draft Report.
NUREG-1235: TECHNICAL SPECIFICATIONS FOR CLINTON POWER STATION,UNIT 1. Docket No. 50-461.(Illinois Power Company)

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NUREG-1057 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BEAVER VALLEY POWER STATION,UNIT 2.Docket No. 50-412.(Duquesne Light Company,et al)
NUREG-1259: TECHNICAL SPECIFICATIONS FOR BEAVER VALLEY POWER STATION, UNIT 2.Docket No. 50-412.(Duquesne Light Company)
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NUREG-1002 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BRAIDWOOD STATION,UNITS 1 AND 2.Docket Nos. 50-456 And 50-457.(Commonwealth Edison Company)
NUREG-1166: FINAL ENVIRONMENTAL STATEMENT FOR DECOMMISSIONING HUMBOLDT BAY POWER PLANT,UNIT 3.Docket No. 50-133.(Pacific Gas And Electric Company)
NUREG-1261: TECHNICAL SPECIFICATIONS FOR BRAIDWOOD STATION,UNITS 1 AND 2.Docket Nos. 50-456 And 50-457.(Commonwealth Edison Company)
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DIVISION OF REACTOR INSPECTION & SAFEGUARDS (POST 870411)
NUREG-0040 V11 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1987.(White Book)
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NUREG-1021 R04: OPERATOR LICENSING EXAMINER STANDARDS.
DIVISION OF HUMAN FACTORS TECHNOLOGY (851125-870411)
NUREG-1258 DRFT: EVALUATION PROCEDURE FOR SIMULATION FACILITIES CERTIFIED UNDER 10CFR55.Draft Report.

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There were no NUREG/IA reports for this quarter.

NRC Contract Sponsor Index (Contractor Reports)

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

OFFICE OF EXECUTIVE DIRECTOR FOR OPERATIONS (EDO)
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NUREG/CR-4651: DEVELOPMENT OF RIPRAP DESIGN CRITERIA BY RIPRAP TESTING IN FLUMES.Phase I.

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA
OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA, DIRECTOR
NUREG/CR-2000 V06 N2: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of February 1987.
NUREG/CR-2000 V06 N3: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of March 1987.
NUREG/CR-2000 V06 N4: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of April 1987.
NUREG/CR-2000 V06 N5: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of May 1987.
NUREG/CR-4674 V03: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1984,A STATUS REPORT.Main Report And Appendixes A And B.
NUREG/CR-4674 V04: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1984,A STATUS REPORT.Appendixes C,D And E.

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NUREG/CR-4664: CLOSEOUT OF IE BULLETIN 83-04:FAILURE OF THE UNDERVOLTAGE TRIP FUNCTION OF REACTOR TRIP BREAKERS.

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OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS,DIRECTOR
NUREG/CR-4615 V02: MODELING STUDY OF SOLUTE TRANSPORT IN THE UNSATURATED ZONE. Workshop Proceedings.
NUREG/CR-4651: DEVELOPMENT OF RIPRAP DESIGN CRITERIA BY RIPRAP TESTING IN FLUMES.Phase I.
NUREG/CR-4773: DESIGN FEATURES TO FACILITATE INTERNATIONAL SAFEGUARDS AT MIXED-OXIDE CONVERSION FACILITIES.
NUREG/CR-4901: EFFECTS FROM INFLUENT BOUNDARY CONDITIONS ON TRACER MIGRATION AND SPATIAL VARIABILITY FEATURES IN INTERMEDIATE-SCALE EXPERIMENTS.
DIVISION OF FUEL CYCLE, MEDICAL, ACADEMIC & COMMERCIAL USE SAFETY (POST
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NUREG/CR-4875: CHARACTERIZATION OF CRUSHED TUFF FOR THE EVALUATION OF THE FATE OF TRACERS IN TRANSPORT STUDIES IN THE UNSATURATED ZONE.

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NUREG/CR-2331 V06 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, July-September 1986.
NUREG/CR-4082 V05: DEGRADED PIPING PROGRAM - PHASE II.Semiannual Report, April-September 1986.
NUREG/CR-4098: SEISMIC-FRAGILITY TESTS OF NEW AND ACCELERATED-AGED CLASS 1E BATTERY CELLS.
NUREG/CR-4219 V03 N2: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For April-September 1986.
NUREG/CR-4307 V03: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.1986 Annual Report,October 1985 - September 1986.
NUREG/CR-4617: ONSITE ASSESSMENTS OF THE EFFECTIVENESS AND IMPACTS OF UPGRADED EMERGENCY OPERATING PROCEDURES.
NUREG/CR-4623: IN-SITU STRESS MEASUREMENTS IN THE EARTH'S CRUST IN THE EASTERN UNITED STATES.
NUREG/CR-4679: QUANTITATIVE DATA ON THE FIRE BEHAVIOR OF COMBUSTIBLE MATERIALS FOUND IN NUCLEAR POWER PLANTS.A Literature Review.
NUREG/CR-4681: ENCLOSURE ENVIRONMENT CHARACTERIZATION TESTING FOR THE BASE LINE VALIDATION OF COMPUTER FIRE SIMULATION CODES.
NUREG/CR-4700 V4 DRF: CONTAINMENT EVENT ANALYSIS FOR POSTULATED SEVERE ACCIDENTS: GRAND GULF NUCLEAR STATION,UNIT 1.Draft For Comment.
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NUREG/CR-4765: MXS CROSS-SECTION PREPROCESSOR USER'S MANUAL.
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NUREG/CR-4819 V01: AGING AND SERVICE WEAR OF SOLENOID-OPERATED VALVES USED IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS.Volume 1.Operating Experience And Failure Identification.
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NUREG/CR-4845: AN ANALYSIS OF THE SEMISCALE MOD-2C S-NH-3 TEST USING THE TRAC-PF1 COMPUTER PROGRAM.
NUREG/CR-4848: STEAM GENERATOR GROUP PROJECT.Annual Report - 1985.
NUREG/CR-4872: EXPERIMENTAL AND ANALYTICAL ASSESSMENT OF CIRCUMFERENTIALLY SURFACE-CRACKED PIPES UNDER BENDING.
NUREG/CR-4877: ASSESSMENT OF DESIGN BASIS FOR LOAD-CARRYING CAPACITY OF WELD-OVERLAY REPAIRS.

NUREG/CR-4878: ANALYSIS OF EXPERIMENTS ON STAINLESS STEEL FLUX WELDS. Topical Report.
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NUREG/CR-4161 V02: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 2:Tuff.
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NUREG/CR-4469 V06: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report, October 1986 - March 1987.
NUREG/CR-4527 V01: AN EXPERIMENTAL INVESTIGATION OF INTERNALLY IGNITED FIRES IN NUCLEAR POWER PLANT CONTROL CABINETS.Part 1:Cabinet Effects Tests.
NUREG/CR-4583 V02: DEVELOPMENT AND VALIDATION OF A REAL-TIME SAFT-UT SYSTEM FOR THE INSPECTION OF LIGHT WATER REACTOR COMPONENTS.Annual Report,October 1984 - September 1985.
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NUREG/CR-4912: DATING GROUND WATER AND THE EVALUATION OF REPOSITORIES FOR RADIOACTIVE WASTE.
NUREG/CR-4918 V01: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Annual Report,October 1985 - September 1986.
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NUREG/CR-4936: AN INTEGRATED GEOLOGICAL,GEOPHYSICAL,AND GEOCHEMICAL INVESTIGATION OF THE MAJOR FRACTURES ON THE EAST SIDE OF THE NEW MADRID EARTHQUAKE ZONE.
NUREG/CR-4950 V01: THE SHORELINE ENVIRONMENT ATMOSPHERIC DISPERSION EXPERIMENT (SEADEX).Experiment Description.
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NUREG/CR-4551 V3 PT1: EVALUATION OF SEVERE ACCIDENT RISKS AND THE POTENTIAL FOR RISK REDUCTION:PEACH BOTTOM,UNIT 2.Main Report.Draft For Comment.
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NUREG/CR-4551 V2 DRF: EVALUATION OF SEVERE ACCIDENT RISKS AND THE POTENTIAL FOR RISK REDUCTION:SEQUOYAH POWER STATION,UNIT 1.Draft For Comment.
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NUREG/CR-4821: REACTOR COOLANT PUMP SHAFT SEAL STABILITY DURING STATION BLACKOUT.

AMERICAN CHEMICAL SOCIETY

NUREG/CP-0078: PROCEEDINGS OF THE SYMPOSIUM ON CHEMICAL PHENOMENA ASSOCIATED WITH RADIOACTIVITY RELEASES DURING SEVERE NUCLEAR PLANT ACCIDENTS.

ANALYSIS & MEASUREMENT SERVICES CORP.

NUREG/CR-4928: DEGRADATION OF NUCLEAR PLANT TEMPERATURE SENSORS.

ARGONNE NATIONAL LABORATORY

NUREG/CR-4842: A STUDY OF NATURAL GLASS ANALOGUES AS APPLIED TO ALTERATION OF NUCLEAR WASTE GLASS.

NUREG/CR-4964: UPDATE OF TABLE S-3 NONRADIOLOGICAL ENVIRONMENTAL PARAMETERS FOR A REFERENCE LIGHT-WATER REACTOR.Uranium Mining,Milling And Enrichment.

ARIZONA, UNIV. OF, TUCSON, AZ

NUREG/CR-4655: UNSATURATED FLOW AND TRANSPORT THROUGH FRACTURED ROCK RELATED TO HIGH-LEVEL WASTE REPOSITORIES.Final Report - Phase II.

NUREG/CR-4912: DATING GROUND WATER AND THE EVALUATION OF REPOSITORIES FOR RADIOACTIVE WASTE.

BATTELLE MEMORIAL INSTITUTE, COLUMBUS LABORATORIES

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NUREG/CR-4877: ASSESSMENT OF DESIGN BASIS FOR LOAD-CARRYING CAPACITY OF WELD-OVERLAY REPAIRS.

NUREG/CR-4878: ANALYSIS OF EXPERIMENTS ON STAINLESS STEEL FLUX WELDS. Topical Report.

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NUREG/CR-2850 V05: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1983.

NUREG/CR-3231: PIPE-TO-PIPE IMPACT PROGRAM.

NUREG/CR-4300 V04 N1: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Progress Report, October 1986 - March 1987.

NUREG/CR-4330 V03: REVIEW OF LIGHT WATER REACTOR REGULATORY REQUIREMENTS. Assessment Of Selected Regulatory Requirements That May Have Marginal Importance To Risk:Postaccident Sampling System, Turbine Missiles,Combustible Gas Control,Charcoal Filters.

NUREG/CR-4469 V05: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report, April-September 1986.

NUREG/CR-4469 V06: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report, October 1986 - March 1987.

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