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Vol. 12, No. 1

Regulatory and Technical Reports (Abstract Index Journal)

Compilation for
First Quarter 1987
January - March

U.S. Nuclear Regulatory Commission

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:

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

Secondary Report Number Index
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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 N09: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of August 31,1986.(Gray Book I) ROSS,P.A.; BEEBE,M.R. Division of Computer & Telecommunications Services (Pre 870413). March 1987. 463pp. 8703170242. 40052:208.

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 N10: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of September 30,1986.(Gray Book I) ROSS,P.A.; BEEBE,M.R. Division of Computer & Telecommunications Services (Pre 870413). March 1987. 447pp. 8704090032. 40468:027.

See NUREG-0020,V10,N09 abstract.

NUREG-0040 V10 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1986.(White Book) * Division of QA, Vendor & Technical Training Center Programs (850212-870411). February 1987. 274pp. 8703170192. 40047:139.

This periodical covers the results of inspections performed by the NRC's Vendor Program Branch that have been distributed to the inspected organizations during the period from October 1986 thru December 1986. Also, included in this issue are the results of certain inspections performed prior to October 1986 that were not included in previous issues of NUREG-0040.

NUREG-0090 V09 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1986. * Office for Analysis & Evaluation of Operational Data, Director. January 1987. 71pp. 8703030835. 39866:152.

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 April 1 through June 30, 1986. During the report period, there were two abnormal occurrences at the nuclear power plants licensed to operate. One involved an out of sequence control rod withdrawal and the other involved a boiling water reactor emergency core cooling system design deficiency. There were five abnormal occurrences at the other NRC licensees. Two involved willful failure to report diagnostic medical misadministrations to the NRC; one involved a therapeutic medical misadministration; and two involved diagnostic medical misadministrations. There were two abnormal occurrences reported by Agreement States. One involved an uncontrolled release of krypton-85 to an unrestricted area; the other involved a contaminated radiopharmaceutical used in diagnostic administrations. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0304 V11 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1986. * Office of Administration (Pre 870413). March 1987. 200pp. 8704010175. 40324:254.

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-0325 R10: U.S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS. * Office of Administration (Pre 870413). February 1987. 57pp. 8702240496. 39725:023.

Functional organization charts for the NRC Commission Offices, Divisions, and Branches are presented.

NUREG-0386 D04 R04: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.July 1972 - June 1986. * Office of the General Counsel. February 1987. 661pp. 8703170245. 40050:267.

This Revision 4 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 June 30, 1986, interpreting the NRC Rules of Practice in 10 CFR Part 2. This Revision 4 replaces in part earlier editions and supplements and includes appropriate changes reflecting the amendment to the Rules of Practice effective June 30, 1986.

NUREG-0430 V07 N01: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.January-June 1986.(Gray Book II) * Office of Inspection & Enforcement, Director (820201-870411). February 1987. 15pp. 8703120210. 39984:016.

NRC is committed to the periodic publication of licensed fuel facilities inventory difference data, following agency review of the information and completion of any related NRC investigations. Information in this report includes inventory difference data for active fuel fabrication facilities possessing more than one effective kilogram of high enriched uranium, low enriched uranium, plutonium, or uranium-233.

NUREG-0525 R12: SAFEGUARDS SUMMARY EVENT LIST (SSEL). GRAMANN,R.H. Division of Safeguards (Pre 870413). February 1987. 47pp. 8703130205. 40005:289.

The Safeguards Summary Event List provides brief summaries of hundreds of safeguards-related events involving nuclear material of facilities regulated by the U.S. Nuclear Regulatory

Commission. Events are described under the categories: bomb-related, intrusion, missing/allegedly stolen, transportation-related, tampering/vandalism, arson, firearms-related, radiological sabotage, nonradiological sabotage, and miscellaneous. Information in the event descriptions was obtained from official NRC reports.

NUREG-0540 V08 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. October 1-31,1986. * Division of Technical Information & Document Control (Pre 870120). December 1986. 479pp. 8701200421. 39353:106.

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 V08 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30,1986. * Office of Administration (Pre 870413). January 1987. 377pp. 8701200372. 39319:208.

See NUREG-0540,V08,N10 abstract.

NUREG-0540 V08 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. December 1-31,1986. * Office of Administration (Pre 870413). February 1987. 439pp. 8703120259. 39984:032.

See NUREG-0540,V08,N10 abstract.

NUREG-0540 V09 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31,1987. * Office of Administration (Pre 870413). March 1987. 350pp. 8704060097. 40398:060.

See NUREG-0540,V08,N10 abstract.

NUREG-0750 V24 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1986.Pages 1-195. * Office of Administration (Pre 870413). January 1987. 204pp. 8702170032. 39659:214.

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 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1986.Pages 197-396. * Office of Administration (Pre 870413). February 1987. 208pp. 8703090229. 39930:021.

See NUREG-0750,V24,N01 abstract.

NUREG-0750 V24 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1986.Pages 397-488. * Office of Administration (Pre 870413). March 1987. 102pp. 8704010165. 40319:061.

See NUREG-0750,V24,N01 abstract.

NUREG-0781 S02: 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 Pressurized Water Reactor Licensing - A (851125-870411). January 1987. 76pp. 8702190398. 39690:058.

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 addresses and resolves some of the outstanding issues remaining after issuance of the Safety Evaluation Report and Supplement No. 1.

NUREG-0837 V06 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, July-September 1986. JANG,J.; RABATIN,K.; COHEN,L. Region 1, Office of Director. February 1987. 224pp. 8703120254. 39985:111.

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 third quarter of 1986.

NUREG-0853 S08: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF CLINTON POWER STATION,UNIT 1.Docket No. 50-461.(Illinois Power Company,et al) * Division of Boiling Water Reactor (BWR) Licensing (851125-870411). March 1987. 47pp. 8704090033. 40466:158.

Supplement No. 8 to the Safety Evaluation Report on the application filed by Illinois Power Company, Soyland Power Cooperative, Inc., and Western Illinois Power Cooperative, Inc., as applicants and owners, for a license to operate the Clinton Power Station, Unit No. 1, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Harp Township, DeWitt County, Illinois. This supplement reports the status of items that have been resolved by the staff since Supplement No. 7 was issued.

NUREG-0857 S11: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PALO VERDE NUCLEAR GENERATING STATION,UNITS 1,2 AND 3.Docket Nos. 50-528,50-529 And 50-530.(Arizona Public Service Company,et al) * Division of Pressurized Water Reactor Licensing - B (851125-870411). March 1987. 51pp. 8704080187. 40441:240.

Supplement No. 11 to the Safety Evaluation Report for the application filed by Arizona Public Service Company, et al, for licenses to operate the Palo Verde Nuclear Generating Station, Units 1, 2 and 3 (Docket Nos. STN 50-528/529/530) located in Maricopa County, Arizona, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation Report by providing an evaluation of (1) additional information submitted by the applicant since Supplement No. 10 was issued and (2) other matters requiring staff review since Supplement No. 10 was issued, specifically those issues that required resolution before Unit 3 low-power licensing.

NUREG-0876 S08: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BYRON STATION,UNITS 1 AND 2.Docket Nos. 50-454 And 50-455. (Commonwealth Edison Company) * Division of Pressurized Water Reactor Licensing - A (851125-870411). March 1987. 24pp. 8704010167. 40321:134.

Supplement No. 8 to the Safety Evaluation Report related to Commonwealth Edison Company's application for licenses to operate the Byron Station, Units 1 and 2, located in Rockvale Township, Igle County, Illinois, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement provides recent information regarding resolution of the license conditions identified in the SER. Because of the favorable resolution of the items discussed in this report, the staff concludes that Byron Station, Unit 2 can be operated by the licensee at power levels greater than 5% without endangering the health and safety of the public.

NUREG-0933 S06: A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R.; VANDERMOLEN,H.; PITTMAN,J.; et al. Division of Safety Review & Oversight (851125-870411). March 1987. 325p. 8704090020. 40467:073.

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

NUREG-0936 V05 N03: NRC REGULATORY AGENDA.Quarterly Report,July-September 1986. * Division of Rules & Records (Pre 870413). January 1987. 160pp. 8702060261. 39527:089.

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. The Agendas for April and October are published in their entirety in the Federal Register while a notice of availability is published in the Federal Register for the January and July Agendas.

NUREG-0940 V05 N04: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,October-December 1986. * Office of Inspection & Enforcement, Director (820201-870411). February 1987. 579pp. 8703120167. 39993:261.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (October-December 1986) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions and the licensee's responses. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, in the interest of promoting public health and safety as well as common defense and security.

NUREG-0975 V05: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS BRANCH, DIVISION OF ENGINEERING SAFETY.Annual Rept For FY 1986. * Division of Engineering Safety (860720-870413). March 1987. 411pp. 8704030183. 40377:267.

This report presents summaries of the research work performed during Fiscal Year 1986 by laboratories and organizations under contracts administered by the NRC's Materials Branch, Office of Nuclear Regulatory Research. Each contractor has written a more complete and detailed annual report of their work which can be obtained by writing to NRC; however, we believe it is useful to have a summary of each contractor's efforts for the year combined into one volume.

NUREG-1030: SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING NUCLEAR POWER PLANTS.Unresolved Safety Issue A-46. CHANG,T.Y. Division of Safety Review & Oversight (851125-870411). February 1987. 183pp. 8703130028. 40007:196.

The margin of safety provided in existing nuclear power plant equipment to resist seismically induced loads and perform their intended safety functions may vary considerably, because of significant changes in design criteria and methods for the seismic qualification of equipment over the years. Therefore, the seismic qualification of equipment in operating plants must be reassessed to determine whether requalification is necessary. The objective of USI A-46 is to establish an explicit set of guidelines and acceptance criteria to judge the seismic adequacy of equipment at all operating plants, in lieu of requiring qualification to the current criteria that are applied to new plants. This report summarizes the work accomplished on USI A-46 by the Nuclear Regulatory Commission staff and its contractors. In addition, the collection and review of seismic experience data and existing seismic test data by the SQUG and EPRI respectively, and the review and recommendations of the SSRAP are presented. The principal technical finding of USI A-46 is that seismic experience data, supplemented by existing seismic test data, applied in accordance with the guidelines developed, provides the most reasonable alternative to current qualification criteria to verify the seismic adequacy of equipment in operating nuclear plants. Explicit seismic qualification should be required only if seismic experience data or existing test data on similar components cannot be shown to apply.

NUREG-1057 S04: 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 Pressurized Water Reactor Licensing - A (851125-870411). March 1987. 91pp. 8704010184. 40324:162.

Supplement No. 4 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. 3 was issued, and (2) matters that the staff had under review when Supplement No. 3 was issued.

NUREG-1100 V03: BUDGET ESTIMATES.Fiscal Years 1988-1989. * Division of Budget & Analysis (Pre 870413). January 1987. 72pp. 8702060083. 39526:087.

This report contains the fiscal year budget justifications to Congress. The budget provides estimates for salaries and expenses for fiscal years 1988-1989.

NUREG-1101 V02: ONSITE DISPOSAL OF RADIOACTIVE WASTE.Methodology For The Radiological Assessment Of Disposal By Subsurface Burial. NEUDER,S.M.; KENNEDY,W.E. Division of Waste Management (Pre 870413). February 1987. 50pp. 8704010173. 40340:309.

Volume 1 of this NUREG provides guidance for academic, medical, and industrial licensees seeking authorization to dispose of small quantities of radioactive material by onsite subsurface disposal. Licensee requests for such authorizations are made pursuant to Section 20.302 of 10 CFR Part 20 "Standards for Protection Against Radiation." This volume (Volume 2) describes the criteria and technical methodology used by NRC staff to evaluate requests by licensees for approval of onsite disposal by burial in soil. The technical methodology includes the ONSITE/MAXI1 code for calculating radiological exposure from various pathways, the MOMOD84 code, and analytical methods for calculating contaminant transport and concentration of radionuclides in flowing groundwater. Radiological exposure analyses include the following pathways: (1) exposure to direct gamma from any surface contamination or buried waste, (2) drinking water from a well contaminated by migration of radionuclides, (3) ingesting agricultural products derived from radionuclide- contaminated soil, and (4) inhaling radionuclides resuspended at the burial site. Licensee-proposed disposal activities are evaluated in terms of radiological impact on public health and safety and the environment. The estimated committed effective dose equivalent resulting from the technical evaluation will usually be the determining factor in the authorization of the proposed disposal.

NUREG-1137 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF VOGTLE ELECTRIC GENERATING PLANT,UNITS 1 AND 2.Docket Nos. 50-424 And 50-425.(Georgia Power Company,et al) * Division of Pressurized Water Reactor Licensing - A (851125-870411). January 1987. 121pp. 8702060100. 39526:288.

In June 1985, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1137) regarding the application of Georgia Power Company, Municipal Electric Authority of Georgia, Oglethorpe Power Corporation, and the City of Dalton, Georgia, for licenses to operate the Vogtle Electric Generating Plant, Units 1 and 2 (Docket Nos. 50-424 and 50-425). Supplement 1 to NUREG-1137 was issued by the staff in October 1985, Supplement 2 was issued in May 1986, Supplement 3 was issued in August 1986, and Supplement 4 was issued in December 1986. The facility is located in Burke County, Georgia, approximately 26 miles south-southeast of Augusta, Georgia, and on the Savannah River. This fifth supplement to NUREG-1137 provides recent information regarding resolution of some of the open and confirmatory items that remained unresolved at the time the Safety Evaluation Report was issued.

NUREG-1137 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF VOGTLE ELECTRIC GENERATING PLANT,UNITS 1 AND 2.Docket Nos. 50-424 And 50-425.(Georgia Power Company,et al) * Division of Pressurized Water Reactor Licensing - A (851125-870411). March 1987. 125pp. 8704010183. 40341:294.

In June 1985, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1137) regarding the application of Georgia Power Company, Municipal Electric Authority of Georgia, Oglethorpe Power Corporation, and the City of Dalton, Georgia, for licenses to operate the Vogtle Electric Generating Plant, Units 1 and 2 (Docket Nos. 50-424 and 50-425). Supplement 1 to NUREG- 1137 was issued by the staff in October 1985, Supplement 2 was issued in May 1986, Supplement 3 was issued in August 1986, Supplement 4 was issued in December 1986, and Supplement 5 was issued in January 1987. The facility is located in Burke County, Georgia, approximately 26 miles south-southeast of Augusta, Georgia, and on the Savannah River. This sixth supplement to NUREG-1137 provides recent information regarding resolution of some of the open and confirmatory items that remained unresolved at the time the Safety Evaluation Report was issued.

NUREG-1150 DRF V1 FC: REACTOR RISK REFERENCE DOCUMENT.Main Report.Draft For Comment. ERNST,M.L.; MURPHY,J.A.; CUNNINGHAM,M.A.; et al. Office of Nuclear Regulatory Research, Director (Post 860720). February 1987. 364pp. 8703100040. 39934:011.

This document discusses the risks of severe accidents in a set of commercial nuclear power plants. This risk is characterized by the types and frequencies of accidents leading to severe core damage, the performance of containment structures under severe accident loadings, possible radioactive releases into the environment if the containment were to fail, and the offsite consequences of such releases. Volume 1 of this document summarizes the principal results of the risk analyses, and displays these results in the context of specific regulatory issues (e.g., safety goals).

NUREG-1150 DRF V2 FC: REACTOR RISK REFERENCE DOCUMENT.Appendices A-I.Draft For Comment. DENNING,R.S.; LEONARD,M.; WREATHALL,J. Office of Nuclear Regulatory Research, Director (Post 860720). February 1987. 320pp. 8703170207. 40053:342.

This document discusses the risks of severe accidents in a set of commercial nuclear power plants. This risk is characterized by the types and frequencies of accidents leading to severe core damage, the performance of containment structures under severe accident loadings, possible radioactive releases into the environment if the containment were to fail, and the offsite consequences of such releases. Volume 2 of this document provides a discussion of the methods used to calculate risk, and summarizes the principal results of the analyses of the studied plants.

NUREG-1150 DRF V3 FC: REACTOR RISK REFERENCE DOCUMENT.Appendices J-O.Draft For Comment. * Office of Nuclear Regulatory Research, Director (Post 860720). February 1987. 517pp. 8703180080. 40072:142.

This document discusses the risks of severe accidents in a set of commercial nuclear power plants. This risk is characterized by the types and frequencies of accidents leading to severe core damage, the performance of containment structures under severe accident loadings, possible radioactive releases into the environment if the containment were to fail, and the offsite consequences of such releases. Volume 3 of this document provides discussions of NRC staff analyses of specific technical and regulatory issues, compares present risk results with those of other studies, and describes computer codes used in the risk analyses.

NUREG-1163: COORDINATION OF SAFETY RESEARCH FOR THE BABCOCK AND WILCOX INTEGRAL SYSTEM TEST PROGRAM. YOUNG,M.W.; SURSOCK,J.P. Division of Reactor System Safety (860720-870413). March 1987. 250pp. 8704010410. 40328:068.

This report describes the MIST facility and all the Integral System Test (IST) support projects sponsored by the USNRC and by EPRI. These support projects have been deemed to play an essential role in helping resolve issues raised by MIST scaling compromises. Each support project is described in detail and application of the expected data to resolution of issues is discussed. The combined effort of MIST and seven other support projects will resolve virtually all questions addressed by the IST program.

NUREG-1199: STANDARD FORMAT AND CONTENT OF A LICENSE APPLICATION FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY. * Division of Waste Management (Pre 870413). January 1987. 108pp. 8702170341. 39668:281.

The Standard Format and Content of a License Application for a Low-Level Radioactive Waste Disposal Facility, NUREG-1199, discusses the information to be provided in the Safety Analysis Report and establishes a uniform format for presenting the information required to meet the licensing requirements for land disposal of radioactive waste as required by 10 CFR 61. The use of the Standard Format will (1) help ensure that the Safety Analysis Report (SAR) contains the information required by 10 CFR 61, (2) aid the applicant in ensuring that the information is complete, (3) help persons reading the SAR to locate information, and (4) contribute to shortening the time required for the review process. The Standard Format and Content (NUREG-1199) ensures that the information required to perform the review is provided, and in a usable format while the Standard Review Plan, NUREG-1200, defines the technical review process. These documents provide assurance that NRC can review and process a license application within 15 months and meet the requirements of Section 9(1) and (2) of P.L. 99-240, the Low-Level Radioactive Waste Policy Amendments Act (LLRWPAA) of 1985.

NUREG-1200: STANDARD REVIEW PLAN FOR THE REVIEW OF A LICENSE APPLICATION FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY. * Division of Waste Management (Pre 870413). January 1987. 473pp. 8702170380. 39669:243.

The Standard Review Plan (SRP) is prepared for the guidance of staff reviewers in the Office of Nuclear Material Safety and Safeguards in performing safety reviews of applications to construct and operate a low-level waste disposal facility. The principal purpose of the SRP is to assure the quality and uniformity of staff reviews and to present a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. It is also a purpose of the SRP to make information about regulatory matters widely available and to improve communication and understanding of the staff's review process by interested members of the public and the nuclear industry. NUREG-1200 consists of 11 Chapters containing approximately 60 individual SRP sections. Each section identifies who performs the review, the matters that are reviewed, the basis for review, how the review is performed, and the conclusions that are sought.

NUREG-1210 V01: PILOT PROGRAM:NRC SEVERE REACTOR ACCIDENT INCIDENT RESPONSE TRAINING MANUAL.Overview And Summary Of Major Points. MCKENNA,T.J.; MARTIN,J.A.; MILLER,C.W.; et al. Division of Emergency Preparedness & Engineering Response (850212-870411). February 1987. 111pp. 8703090078. 39931:107.

This is one in a series of volumes that collectively provide for the U.S. Nuclear Regulatory Commission (NRC) emergency response personnel the necessary background information for an adequate response to severe reactor accidents. The volumes in the series are: Volume 1, "Overview and Summary of Major Points," Volume 2, "Severe Reactor Overview," Volume 3, "Response of Licensee and State and Local Officials," Volume 4, "Public Protective Actions - Predetermined Criteria and Initial Actions," and Volume 5, "U.S. Nuclear Regulatory Commission Response." Each volume serves, respectively, as the text for a course of instruction in a series of courses for NRC response personnel. These materials do not provide guidance or license requirements for NRC licensees or state or local response organizations. Each volume is accompanied by an appendix of slides that can be used to present this material. The slides are called out in the text.

NUREG-1210 V02: PILOT PROGRAM:NRC SEVERE REACTOR ACCIDENT INCIDENT RESPONSE TRAINING MANUAL.Severe Reactor Accident Overview. MCKENNA,T.J.; MARTIN,J.A.; MILLER,C.W.; et al. Division of Emergency Preparedness & Engineering Response (850212-870411). February 1987. 132pp. 8703090131. 39931:218.

See NUREG-1210,V01 abstract.

NUREG-1210 V03: PILOT PROGRAM:NRC SEVERE REACTOR ACCIDENT INCIDENT RESPONSE TRAINING MANUAL.Response Of Licensee And State And Local Officials. SAKENAS,C.A.; MCKENNA,T.J.; MILLER,C.W.; et al. Division of Emergency Preparedness & Engineering Response (850212-870411). February 1987. 101pp. 8703090090. 39931:006.

See NUREG-1210,V01 abstract.

NUREG-1210 V04: PILOT PROGRAM:NRC SEVERE REACTOR ACCIDENT INCIDENT RESPONSE TRAINING MANUAL.Public Protective Actions - Predetermined Criteria And Initial Actions. MARTIN,J.A.; MCKENNA,T.J.; MILLER,C.W.; et al. Division of Emergency Preparedness & Engineering Response (850212-870411). February 1987. 117pp. 8703090147. 39922:001.

See NUREG-1210,V01 abstract.

NUREG-1210 V05: PILOT PROGRAM:NRC SEVERE REACTOR ACCIDENT INCIDENT RESPONSE TRAINING MANUAL.U.S. Nuclear Regulatory Commission Response. SAKENAS,C.A.; MCKENNA,T.J.; PERKINS,K.; et al. Division of Emergency Preparedness & Engineering Response (850212-870411). February 1987. 105pp. 8703090067. 39921:258.

See NUREG-1210,V01 abstract.

NUREG-1211: REGULATORY ANALYSIS FOR RESOLUTION OF UNRESOLVED SAFETY ISSUE A-46,SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS. CHANG,T.Y.; ANDERSON,N.R. Division of Safety Review & Oversight (851125-870411). February 1987. 73pp. 8703120330. 39981:321.

The margin of safety provided in existing nuclear power plant equipment to resist seismically induced loads and perform required safety functions may vary considerably, because of significant changes in design criteria and methods for the seismic qualification of equipment over the years. Therefore, the seismic qualification of equipment in operating plants must be reassessed to determine whether requalification is necessary. The objective of USI A-46 is to establish an explicit set of guidelines and acceptance criteria to judge the seismic adequacy of equipment at all operating plants, in lieu of requiring these plants to meet the criteria that are applied to new plants. This report presents the regulatory analysis for Unresolved Safety Issue (USI) A-46. It includes (1) Statement of the Problem, (2) the Objective of USI A-46, (3) a Summary of A-46 Tasks, (4) a Proposed Implementation Procedure, (5) a Value-Impact Analysis, (6) Implementation, (7) a Summary of A-46 Risk Analyses and (8) Operating Plants To Be Reviewed to USI A-46 Requirements.

NUREG-1224: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE UNIVERSITY OF NEW MEXICO RESEARCH REACTOR.Docket No. 50-252. (University Of New Mexico) * Division of Pressurized Water Reactor Licensing - B (851125-870411). March 1987. 48pp. 8704090018. 40466:110.

This Safety Evaluation Report for the application filed by the University of New Mexico for renewal of operating license number R-102 to continue to operate a research reactor has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is owned and operated by the University of New Mexico and is located on the University's campus in Albuquerque, New Mexico. The staff concludes that the AGN-201M type reactor facility can continue to be operated by the University of New Mexico without endangering the health and safety of the public.

NUREG-1237: TECHNICAL SPECIFICATIONS FOR VOGTLE ELECTRIC GENERATING PLANT,UNIT 1.Docket No. 50-424.(Georgia Power Company) * Division of Pressurized Water Reactor Licensing - A (851125-870411). January 1987. 460pp. 8702060231. 39538:194.

The Vogtle Electric Generating Plant, Unit No. 1, 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 facility as set forth in Section 50.36 of 10 CFR 50 for the protection of the health and safety of the public.

NUREG-1240: TECHNICAL SPECIFICATIONS FOR SHEARON HARRIS NUCLEAR POWER PLANT UNIT 1.Docket No. 50-400.(Carolina Power & Light Company) * Division of Pressurized Water Reactor Licensing - A (851125-870411). January 1987. 473pp. 8702040174. 39508:110.

The Shearon Harris Nuclear Power Plant, Unit 1 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 facility as set forth in Section 50.36 of 10 CFR 50 for the protection of the health and safety of the public.

NUREG-1243: GROUND-WATER PROTECTION ACTIVITIES OF THE U.S. NUCLEAR REGULATORY COMMISSION. * Division of Waste Management (Pre 870413). February 1987. 66pp. 8703130105. 40007:134.

The U.S. Nuclear Regulatory Commission (NRC) provides for ground- water protection through regulations and licensing conditions that require prevention, detection, and correction of ground-water contamination. Prepared by the interoffice Ground-Water Protection Group, this report evaluates the internal consistency of NRC's ground- water protection programs. These programs have evolved consistently with growing public concerns about the significance of ground-water contamination and environmental impacts. Early NRC programs provided for the protection of public health and safety by minimizing releases of radionuclides. More recent programs have included provisions for minimizing releases of non-radiological constituents, mitigating environmental impacts, and correcting ground-water contamination. NRC's ground-water protection programs are categorized according to program areas, including nuclear materials and waste management (NMSS), nuclear reactor operations (NRR), confirmatory research and standards development (RES), inspection and enforcement (IE), and agreement state programs (SP).

NUREG-1247: TECHNICAL SPECIFICATIONS FOR VOGTLE ELECTRIC GENERATING PLANT,UNIT 1.Docket No. 50-424.(Georgia Power Company) * Division of Pressurized Water Reactor Licensing - A (851125-870411). March 1987. 450pp. 8704020068. 40344:250.

The Vogtle Electric Generating Plant, Unit No. 1, 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 facility as set forth in Section 50.36 of 10 CFR 50 for the protection of the health and safety of the public.

NUREG-1248: TECHNICAL SPECIFICATIONS FOR PALO VERDE NUCLEAR GENERATING STATION,UNIT 3.Docket No. 50-530.(Arizona Public Service Company) * Division of Pressurized Water Reactor Licensing - B (851125-870411). March 1987. 300pp. 8704090024. 40462:114.

The Palo Verde, Unit 3 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-1250: REPORT ON THE ACCIDENT AT THE CHERNOBYL NUCLEAR POWER STATION. * NRC - No Detailed Affiliation Given. * Energy, Dept. of. *; et al. Environmental Protection Agency. January 1987. 214pp. 8702170005. 39657:142.

This report presents the compilation of information obtained by various organizations regarding the accident (and the consequences of the accident) that occurred at Unit 4 of the nuclear power station at Chernobyl in the USSR on April 26, 1986. Each organization has independently accepted responsibility for one or more chapters. The various authors are identified in a footnote to each chapter. Chapter 1 provides an overview of the report. Very briefly the other chapters cover: Chapter 2, the design of the Chernobyl nuclear station Unit 4; Chapter 3, safety analyses for Unit 4; Chapter 4, the accident scenario; Chapter 5, the role of the operator; Chapter 6, an assessment of the radioactive release, dispersion, and transport; Chapter 7, the activities associated with emergency actions; and Chapter 8, information on the health and environmental consequences from the accident. These subjects cover the major aspects of the accident that have the potential to present new information and lessons for the nuclear industry in general. The task of evaluating the information obtained in these various areas and the assessment of the potential implications has been left to each organization to pursue according to the relevance of the subject to their organizations. Those findings will be issued separately by the cognizant organizations. The basic purpose of this report is to provide the information upon which such assessments can be made.

NUREG-1260 V01: A REPORT TO CONGRESS ON NUCLEAR REGULATORY RESEARCH.Project Descriptions For FY87. * Office of Nuclear Regulatory Research, Director (Post 860720). February 1987. 574pp. 8703260021. 40244:001.

The report presents project descriptions of NRC research projects funded in Fiscal Year 1987. The individual project descriptions presented in this report are divided into six major groups of related projects. These groups, called issues, are as follows: Severe Accident, Risk and Reliability, Thermal Hydraulic Transients, Plant Aging and Life Extension, Seismic Research, and Waste Management. Within each issue, the project descriptions are further divided into subgroups, called subissues. An overview is provided prior to each issue and subissue giving a statement of the problem being addressed and the research objectives.

NUREG-1280: STANDARD FORMAT AND CONTENT ACCEPTANCE CRITERIA FOR THE MATERIAL CONTROL AND ACCOUNTING (MC&A) REFORM AMENDMENT. 10 CFR Part 74 Subpart E. EMEIGH,C.W. Division of Safeguards (Pre 870413). March 1987. 105pp. 8704080097. 40446:284.

This report documents a standard format suggested by the NRC for use in preparing fundamental nuclear material control plans as required by the Material Control and Accounting Reform Amendment (portions of 10 CFR Part 74). The report also describes the necessary contents of a comprehensive plan and provides example acceptance criteria which are intended to communicate acceptable means of achieving the performance capabilities of the Reform Amendment. By using the suggested format, the license applicant will minimize administrative problems associated with the submittal review and approval of the FNMC plan. Preparation of the plan in accordance with this format will assist the NRC in evaluating the plan and in standardizing the review and licensing process. However, conformance with this guidance is not required by the NRC. A license applicant who employs a format that provides an equivalent level of completeness and detail may use their own format.

NUREG/CP-0054: PROCEEDINGS OF THE WORKSHOP ON SOIL-STRUCTURE INTERACTION. GRAVES,H.L.; PHILIPPACOPOULO Brookhaven National Laboratory. December 1986. 423pp. 8703090227. BNL-NUREG-52011. 39932:006.

The Workshop on Soil-Structure Interaction provided an exchange of information between regulators, practitioners and researchers for the purpose of examining SSI licensing criteria in the light of recent analytical and experimental development. These proceedings contain the papers presented by panelists and summaries of the sessions along with recommendations of the panel members for each session. Technical areas covered by the panels were (1) definition of free-field motion, (2) ground motion input needed for site specific SSI analysis, (3) SSI methodology, and (4) experience and experimental observation. The summaries were derived to identify areas in the licensing criteria which could be changed to improve the licensing process.

NUREG/CP-0082 V01: PROCEEDINGS OF THE FOURTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. * Office of Nuclear Regulatory Research, Director (Post 860720). February 1987. 521pp. 8702270064. 39764:109.

This six-volume report contains 156 papers out of the 175 that were presented at the Fourteenth Water Reactor Safety Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland, during the week of October 27-31, 1986. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included thirty-four different papers presented by researchers from Canada, Czechoslovakia, Finland, Germany, Italy, Japan, Mexico, Spain, Sweden, Switzerland and the United Kingdom. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

NUREG/CP-0082 V02: PROCEEDINGS OF THE FOURTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. * Office of Nuclear Regulatory Research, Director (Post 860720). February 1987. 444pp. 8703030039. 39837:163.

See NUREG/CP-0082,V01 abstract.

NUREG/CP-0082 V03: PROCEEDINGS OF THE FOURTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. * Office of Nuclear Regulatory Research, Director (Post 860720). February 1987. 433pp. 8703030810. 39867:011.

See NUREG/CP-0082,V01 abstract.

NUREG/CP-0082 V04: PROCEEDINGS OF THE FOURTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. * Office of Nuclear Regulatory Research, Director (Post 860720). February 1987. 528pp. 8702270057. 39760:303.

See NUREG/CP-0082,V01 abstract.

NUREG/CP-0082 V05: PROCEEDINGS OF THE FOURTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. * Office of Nuclear Regulatory Research, Director (Post 860720). February 1987. 594pp. 8702270056. 39765:270.

See NUREG/CP-0082,V01 abstract.

NUREG/CP-0082 V06: PROCEEDINGS OF THE FOURTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Brookhaven National Laboratory. * Office of Nuclear Regulatory Research, Director (Post 860720). February 1987. 424pp. 8703030035. 39836:099.

See NUREG/CP-0082,V01 abstract.

NUREG/CP-0084: PROCEEDINGS OF THE WORKSHOP ON A CONTAINMENT PERFORMANCE DESIGN OBJECTIVE,MAY 12-13,1986,HARPERS FERRY,WEST VIRGINIA. * Brookhaven National Laboratory. November 1986. 77pp. 8704080102. BNL-NUREG-52044. 40447:026.

The "Containment Performance Design Objective Workshop" was designed to obtain a broad range of knowledgeable views concerning the issues in the development and implementation of a containment performance design objective (CPDO). It was a discussion workshop, involving invited experts representing a broad range of viewpoints, and drawn from utilities, reactor vendors, architect engineers, universities, national laboratories, and public interest groups. The participants were requested to review background information concerning the safety goals and their status, a description of CPDO options selected for evaluation, an outline of an implementation approach and recognized issues of CPDO structure and implementation. The general objective of the workshop was to generate information that could be used in the NRC's study and decision process concerning the formulation of a containment performance design objective. The participants' views were obtained on specific CPDO options and issues that were presented to the Workshop and new ones that emerged during the discussion. An attempt was also made to identify areas of consensus emerging from the discussion.

NUREG/CP-0085: MEETING WITH STATES ON THE LOW-LEVEL RADIOACTIVE WASTE POLICY AMENDMENTS ACT (LLRWPAA) OF 1985. MAUPIN,C.; SCHNEIDER,K. Assistant Director for State Agreements Programs. February 1987. 168pp. 8703260292. 40238:111.

The purpose of this meeting was to discuss with selected State officials NRC responsibilities under the Low-Level Radioactive Waste Policy Amendments Act, including the approach being taken and progress being made in fulfilling NRC responsibilities. The NRC staff objective was to obtain State views on technical and institutional issues associated with NRC and State implementation of the Act and to determine any additional areas in which NRC can be of assistance in the development of disposal facilities. We believe this objective was accomplished. The transcript of the meeting is being published at this time to make available information discussed at the meeting to those individuals and groups that have responsibilities under the LLRWPAA for developing disposal capacity and for regulating low-level waste disposal sites.

NUREG/CR-2000 V05N12: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of December 1986. * Oak Ridge National Laboratory. January 1987. 133pp. 8702170056. ORNL/NSIC-200. 39657:009.

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 N1: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of January 1987. * Oak Ridge National Laboratory. March 1987. 117pp. 8703250557. ORNL/NSIC-200. 40234:330.

See NUREG/CR-2000,V05,N12 abstract.

NUREG/CR-2331 V06 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April-June 1986. WEISS,A.J. Brookhaven National Laboratory. November 1986. 107pp. 8704080236. BNL-NUREG-51454. 40445:157.

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-2478 V03: A STUDY OF TRENCH COVERS TO MINIMIZE INFILTRATION AT WASTE DISPOSAL SITES.Final Rept. CARTWRIGHT,K.; LARSON,T.H.; HERZOG,B.L.; et al. Illinois, State of. February 1987. 139pp. 8703120215. 39987:171.

We have investigated methods to limit infiltration through trench covers by reviewing current practices, testing selected geologic materials, simulating selected cover designs, and designing, constructing, and monitoring four field-scale experimental covers. Of the many designs considered, we conclude that multilayered soil covers are superior to single-layered covers. Laboratory and computer simulations indicate that a wick effect can be established by placing a fine-grained layer over a coarse-grained layer, thereby delaying and possibly reducing moisture infiltration through the entire cover. Field experiments indicate that the wick effect does occur under certain circumstances; but a potentially more important feature of the layered cover design is the ability of the coarse-grained layer to remove moisture from the system through drain tiles.

NUREG/CR-3232: DETAILED STUDIES OF SELECTED,WELL EXPOSED FRACTURE ZONES IN THE ADIRONDACK MOUNTAINS DOME,NEW YORK. WIENER,R.W.; ISACHSEN,Y.W. New York, State Univ. of, Albany, NY. January 1987. 94pp. 8702260644. 39759:123.

The Adirondack Mountains constitute a relatively young (Mesozoic? Cenozoic?) dome on the craton. The dome is undergoing contemporary uplift, based on geodetic releveling, and is seismically active. The breached dome provides a very large window through Paleozoic cover and thus permits ground study of the fracture systems that characterize the seismogenic basement and influence the patterns of brittle deformation that are found in overlying Paleozoic rocks of the platform. The predominant fracture zones are linear valleys that trend NNE to NE, parallel to the long axis of the dome. The 36 field studies of the lineament segments discussed in this report suggest that the prominent NE to NNE fracture systems in the eastern Adirondacks are dominantly high angle faults down-stepped to the east, whereas those in the central Adirondacks are dominantly zero-displacement crackle zones. The origin of these features is related to the rapid uplift of the Adirondack dome. Similar features can be expected to be found in other areas of domal uplift or rapid regional uplift.

NUREG/CR-3412 V02: CONTAINMENT INTEGRITY PROGRAM.Progress Report,April 1983 -December 1984. BLEJWAS,T.E.; HORSCHEL,D.S. Sandia National Laboratories. December 1986. 58pp. 8702060139. SAND83-1482. 39527:249.

This report contains a description of work performed between April 1983 and December 1984 under the Containment Integrity Program. The program is one of three at Sandia National Laboratories in the general area of containment integrity. The overall objective of the three programs is the qualification of methods for reliably predicting the capability of containment structures for light water reactor nuclear power plants to function under loadings caused by severe accidents and extreme environments. In the subject program, models of entire containments are tested for loadings beyond the design basis. Experiments completed during the reporting period include a series of internal-pressure tests of 1/32 and 1/8 size models of hybrid steel containments. Comparisons with pretest analyses are described.

NUREG/CR-3444 V04: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL,AND ASSOCIATED OCCUPATIONAL EXPOSURE.Annual Report, Fiscal Year 1986. PICIULO,P.L.; ADAMS,J.W. Brookhaven National Laboratory. October 1986. 48pp. 8704130283. BNL-NUREG-51699. 40495:328.

Leach tests were initiated in order to determine if organic reagents released from different size waste forms can be represented by a diffusion controlled mechanism. Data for the release of EDTA from cement forms showed that the CFR from the 5 cm diameter 10 cm long forms were smaller than that from the 15 cm diameter x 15 cm diameter forms. This would not be expected if the predominant mechanism of release was diffusion. Specimens containing Co-60 spiked cation exchange resins solidified in cement were leached with deionized water and leachants containing either formate or picolinate. Data from the first 60 days of leaching indicate that the releases of Co-60 were similar with deionized water and formate as leachants. Picolinic acid present in the leachant caused an acceleration of the release of Co-60.

NUREG/CR-3468: HYDROGEN:AIR:STEAM FLAMMABILITY LIMITS AND COMBUSTION CHARACTERISTICS IN THE FITS VESSEL. MARSHALL,B.W. Sandia National Laboratories. December 1986. 149pp. 8704080363. SAND84-0383. 40445:008.

Experimentally observed flammability limits of hydrogen:air:steam mixtures in both turbulent and quiescent environments were measured and a correlation developed that describes the three-component flammability limit. The combustion pressure data measured for the hydrogen:air:steam tests indicate that the addition of steam reduces the normalized peak combustion pressure (Pmax/Po) as compared to equivalent hydrogen:air burns. Turbulence was found to affect the extent of combustion and other combustion characteristics of the lean hydrogen burns (i.e., less than or equal to 10% hydrogen by volume) where buoyancy governs flame propagation. The experimentally measured pressure decays were used to infer the "global" heat transfer characteristics during the postcombustion cooling phase. Convection was found to dominate the time-integrated heat transfer of the leaner (less than or equal to 10%) hydrogen:air burns, accounting for 50 to 70% of the postcombustion heat transfer. Radiation was slightly more prevalent than convection for the hydrogen:air burns near stoichiometry. When moderate quantities of steam were added to the environment, radiation became the dominant postcombustion cooling mechanism due to the increase in bulk gas emittance. If richer steam concentrations were added to the environment, radiation and convection appear to be equally important heat transfer mechanisms.

NUREG/CR-3469 V03: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS. Annotated Bibliography Of Selected Readings In Radiation Protection And ALARA. BAUM,J.W.; KHAN,T.A. Brookhaven National Laboratory. November 1986. 124pp. 8703100021. BNL-NUREG-51708. 39933:246.

This report is the third in a series of bibliographies supporting the efforts at Brookhaven National Laboratory on dose reduction at nuclear power plants. Abstracts for this report were selected from papers presented at recent technical meetings, journals and research reports reviewed at the BNL ALARA Center, and searches of the DOE/RECON data base on energy-related publications. The references selected for inclusion in the bibliography relate not only to operational health physics topics but also to plant chemistry, stress corrosion cracking, and other aspects of plant operation which have important impacts on occupational exposure. Also included are references to improved design, planning, materials selection and other topics related to what might be called ALARA engineering. Thus, an attempt has been made to cover a broad spectrum of topics related directly or indirectly to occupational exposure reduction. This report contains 252 abstracts and both author and subject indices.

NUREG/CR-3620 S02: INTRUDER DOSE PATHWAY ANALYSIS FOR THE ONSITE DISPOSAL OF RADIOACTIVE WASTES.The ONSITE/MAXI1 Computer Program. KENNEDY,W.E.; PELOQUIN,R.A.; NAPIER,B.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1987. 453pp. 8703120220. PNL-4054. 39986:078.

The document entitled "Intruder Dose Pathway Analysis of the Onsite Disposal of Radioactive Wastes: The ONSITE/MAXI1 Computer Program" (1984) by Napier et al. summarizes our initial efforts to develop human-intrusion scenarios and a modified version of the MAXI computer program for potential use by the NRC in reviewing applications for onsite radioactive waste disposal. Supplement 1 of NUREG/CR-3620 (1986) by Kennedy et al. summarized modifications and improvements to the ONSITE/MAXI1 software package. This document summarizes a modified version of the ONSITE/MAXI1 computer program. This modified version of the computer program operates on a personal computer and permits the user to optionally select radiation dose conversion factors published by the International Commission on Radiological Protection (ICRP) in their Publication No. 30 (ICRP 1979- 1982) in place of those published by the ICRP in their Publication No. 2 (ICRP 1959) (as implemented in the previous versions of the ONSITE/MAXI1 computer program). The pathway-to-human models used in the computer program have not been changed from those described previously (Napier et al. 1984; Kennedy et al. 1986). Computer listings of the ONSITE/MAXI1 computer program and supporting data bases are included in the appendices of this document.

NUREG/CR-3861: STRESS-CORROSION CRACKING OF LOW-STRENGTH CARBON STEELS IN CANDIDATE HIGH-LEVEL WASTE REPOSITORY ENVIRONMENTS. BEAVERS,J.A.; THOMPSON,N.G.; PARKINS,R.N. Battelle Memorial Institute, Columbus Laboratories. February 1987. 88pp. 8703120328. BMI-2147. 39981:234.

A survey of the literature was performed to identify potential stress-corrosion cracking agents for low-strength carbon and low alloy steels in repository environments. It was found that a number of potent cracking agents are present, but stress-corrosion cracking is relatively unlikely in the bulk repository environment because of their low concentration. On the other hand, concentration of these species may occur by a number of mechanisms, and thus it is conceivable that the waste package could fail prematurely by stress corrosion. Accordingly, it is recommended that the lower concentration limits for potential cracking agents be identified under typical repository environments, in conjunction with modeling studies to assess the likelihood that the concentrating mechanisms will operate and to bound the upper limits of concentration for each mechanism.

NUREG/CR-3950 V03: FUEL PERFORMANCE ANNUAL REPORT FOR 1985. BAILEY,W.J. Battelle Memorial Institute, Pacific Northwest Laboratories. WU,S. NRC - No Detailed Affiliation Given. February 1987. 103pp. 8703120248. PNL-5210. 39985:335.

This annual report, the eighth in a series, provides a brief description of fuel performance during 1985 in commercial nuclear power plants. Brief summaries of fuel design changes, fuel surveillance programs, fuel operating experience and trends, fuel problems, high-burnup fuel experience, and items of general significance are provided. References to additional, more detailed information and related NRC evaluations are included.

NUREG/CR-3968: STUDY OF OPERATING PROCEDURES IN NUCLEAR POWER PLANTS. Practices And Problems. MORGENSTERN,M.; BARNES,V.E.; MCGUIRE,M.V.; et al. Battelle Human Affairs Research Centers. February 1987. 156pp. 8703090132. PNL-5648. 39923:340.

This report describes the project activities, findings, and recommendations of a project entitled "Program Plan for Assessing and Upgrading Operating Procedures for Nuclear Power Plants." The project was performed by the Pacific Northwest Laboratory and Battelle Human Affairs Research Centers for the Division of Human Factors Technology, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission (NRC). The project team analyzed and evaluated samples of normal and abnormal operating procedures from 31 commercial nuclear power plants operating in the United States. The project team also visited nine nuclear power plants in the United States to obtain information on the development, use, and control of operating procedures. A peer review group was convened to advise the project team on the conduct of the project and to review and comment on the project report. The report contains findings on the useability of operating procedures and on practices concerning the development, use, and control of operating procedures in nuclear power plants. The report includes recommendations to the NRC on the need to upgrade the quality of operating procedures. The report also discusses an approach to a program plan to assess and upgrade operating procedures.

NUREG/CR-4012 V02: REPLACEMENT ENERGY COSTS FOR NUCLEAR ELECTRICITY-GENERATING UNITS IN THE UNITED STATES:1987-1991. VANKUIKEN,J.C.; GUZIEL,K.A.; BUEHRING,W.A.; et al. Argonne National Laboratory. January 1987. 253pp. 8702190449. ANL-AA-30. 39690:235.

Seasonal replacement energy costs are estimated for potential short-term shutdowns of 116 nuclear electricity-generating units. These estimates were developed to help the U.S. Nuclear Regulatory Commission (NRC) establish regulatory policies, particularly those requiring safety modifications that might necessitate temporary reactor shutdowns. Cost estimates were derived from probabilistic production-cost simulations of pooled utility-system operations. Factors affecting replacement energy costs, such as random unit failures, maintenance and refueling requirements, and load variations, are treated in the analysis. Seasonal costs are presented for the five-year period beginning with 1987 and ending with 1991. This information updates cost estimates that were developed previously for the NRC and published in NRC Report NUREG/CR-4012, Vol. 1. The updates were undertaken to extend the time frame of cost estimates and to account for recent changes in utility system conditions, such as fluctuations in fuel prices, changes in construction and retirement schedules, and adjustments to system demand projections.

NUREG/CR-4016 V02: APPLICATION OF SLIM-MAUD:A TEST OF AN INTERACTIVE COMPUTER-BASED METHOD FOR ORGANIZING EXPERT ASSESSMENT OF HUMAN PERFORMANCE AND RELIABILITY.Volume II:Appendices. SPETTELL,C.M.; ROSA,E.A.; HUMPHREYS,P.C.; et al. Brookhaven National Laboratory. October 1986. 219pp. 8702060128. BNL-NUREG-51828. 39537:248.

The U.S. Nuclear Regulatory Commission (NRC) has been conducting a multi-year research program to investigate different methods for using expert judgments to estimate human error probabilities (HEPs) in nuclear power plants. One of the methods investigated, derived from multi-attribute utility theory, is the Success Likelihood Index Methodology implemented through Multi-Attribute Utility Decomposition (SLIM-MAUD). This report describes a systematic test application of the SLIM-MAUD methodology. The test application is evaluated on the basis of three criteria: practicality, acceptability, and usefulness. Volume I of this report presents an overview of SLIM-MAUD, describes the procedures followed in the test application, and provides a summary of the results obtained. Volume II consists of technical appendices to support in detail the materials contained in Volume I, and the users' package of explicit procedures to be followed in implementing SLIM-MAUD. The results obtained in the test application provide support for the application of SLIM-MAUD to a wide variety of applications requiring estimates of human errors.

NUREG/CR-4300 V03 N2: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR INSERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Progress Rept,April-September 1986. HUTTON,P.H. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1987. 16pp. 8702260636. PNL-5511. 39759:217.

This report discusses technical progress for the period April 1986 - September 1986 for the NRC-sponsored research program concerned with "Acoustic Emission/Flaw Relationships for Inservice Monitoring of Nuclear Reactor Pressure Boundaries." Included in the discussion are the topics of AE monitoring of primary piping during reactor operation, substantiation of the AE signal identification method, development of AE/IGSCC relationships, and progress in establishing an ASTM AE standard and an ASME appendix for on-line AE monitoring.

NUREG/CR-4301: STATUS REPORT ON EQUIPMENT QUALIFICATION ISSUES RESEARCH AND RESOLUTION. BONZON,L.L.; WYANT,F.J.; BUSTARD,L.D.; et al. Sandia National Laboratories. January 1987. 582pp. 8703090128. SAND85-1309. 39922:118.

Since its inception in 1975, the Qualification Testing Evaluation (QTE) Program has produced numerous results pertinent to equipment qualification issues. Many have been incorporated into Regulatory Guides, Rules, and industry practices and standards. This report summarizes the numerous reports and findings to date. Thirty separate issues are discussed encompassing three generic areas: accident simulation methods; aging simulation methods; and, special topics related to equipment qualification. Each issue-specific section contains: (1) a brief description of the issue; (2) a summary of the applicable research effort; and (3) a summary of the findings to date.

NUREG/CR-4320: THE RELATIONSHIP AND INFLUENCES OF FUEL AND COOLANT SYSTEM PROCESSES DURING LWR SEVERE ACCIDENTS. RIVARD,J.B. Sandia National Laboratories. December 1986. 59pp. 8704080124. SAND85-1449. 40446:182.

This report places the processes of fuel and core damage, reactor coolant system (RCS) flow and heat transfer, and pressure vessel breach - and their respective fission product considerations - in the context of typical risk-dominant accident sequences. Thus the enhanced perception of the relationship (and thus importance) of these processes to the potential consequences of a severe accident is provided. This is accomplished using both generic and plant-specific relational methods. It is found that the vessel and RCS processes pervasively influence consequences. The experiment programs designed to provide the in-vessel and RCS fuel damage data base are examined in light of this conclusion, and some suggestions are offered.

NUREG/CR-4409 V02: DATA BASE ON NUCLEAR POWER PLANT DOSE REDUCTION RESEARCH PROJECTS. KHAN,T.A.; BAUM,J.W. Brookhaven National Laboratory. November 1986. 232pp. 8704080217. BNL-NUREG-51934. 40444:134.

This report describes 142 international projects on dose reduction research. It is the second report on a data base maintained by Brookhaven National Laboratory as part of an NRC sponsored project on occupational dose reduction. The first report described 180 similar projects. A wide area of research is covered, including plant chemistry, stress corrosion cracking, steam generator repair and replacement, robotics, and decontamination. Analysis indicates that dose reduction research is beginning to affect occupational radiation exposure. There is a general diminution in exposures in countries with dose reduction research programs, such as Japan, The Federal Republic of Germany, Canada, Sweden, France, and the United States. Most of the present research, however, is directed towards engineering approaches to dose reduction. More attention in the non-engineering areas is called for.

NUREG/CR-4448: SHUTDOWN DECAY HEAT REMOVAL ANALYSIS OF A GENERAL ELECTRIC BWR3/MARK I.Case Study. HATCH,S.W.; ERICSON,D.M.; SANDERS,G.A. Sandia National Laboratories. March 1987. 739pp. 8704060465. SAND85-2373. 40405:123.

A General Electric Boiling Water Reactor (BWR3) with a Mark I containment has been evaluated as part of Task Action Plan A-45, "Decay Heat Removal Requirements." Probabilistic risk assessment models were constructed to determine the dominant internal, randomly initiated accident sequences and special emergency sequences (e.g., earthquakes). The dominant sequences were reviewed to determine what modifications might be made to enhance the plant's ability to remove decay heat. Modifications which held promise went through a preliminary cost and design analysis. Additionally, the impact on the probability of core melt accidents was estimated given implementation of modifications. In the final step, these results were combined in a value-impact format according to NRC guidelines. The results indicate that feasible modifications to enhance decay heat removal do exist at the subject plant. The central estimates of the value-impact results tended, however, to show marginal cost effectiveness under current guidelines for most of the modifications. Alternate assumptions involving source term magnitude and interdiction criteria were found to significantly affect the results. The insights gained from this study will become part of an information base which will be used to develop generic recommendations regarding the adequacy of decay heat removal systems in light water reactors.

NUREG/CR-4458: SHUTDOWN DECAY HEAT REMOVAL ANALYSIS OF A WESTINGHOUSE 2-LOOP PRESSURIZED WATER REACTOR.Case Study. CRAMOND,W.R.; ERICSON,D.M.; SANDERS,G.A. Sandia National Laboratories. March 1987. 900pp. 8704060155. SAND86-2496. 40395:017.

This is one of six case studies for USI A-45 Decay Heat Removal (DHR) Requirements. The purpose of this study is to identify any potential vulnerabilities in the DHR systems of a typical Westinghouse 2-loop PWR, to suggest possible modifications to improve the DHR capability, and to assess the value & impact of the most promising alternatives to the existing DHR systems. The systems analysis considered small LOCAs and transient internal initiating events, and seismic, fire, extreme wind, internal and external flood, and lightning external events. A full-scale systems analysis was performed with detailed fault trees and event trees including support system dependencies. The system analysis results were extrapolated into release categories using applicable past PRA phenomenological results and improved containment failure mode probabilities. Public consequences were estimated using site specific CRAC2 calculations. The Value-Impact (VI) analysis of possible alternatives considered both onsite and offsite impacts arriving at several risk measures such as averted population dose out to a 50-mile radius and dollars per person rem averted. Uncertainties in the VI analysis are discussed and the issues of feed and bleed and secondary blowdown are analyzed.

NUREG/CR-4469 V04: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report,October 1985 - March 1986. DOCTOR,S.R.; BATES,D.J.; DEFFENBAUGH,J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1987. 91pp. 8704060414. PNL-5711. 40410:169.

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 1985 through March 1986.

NUREG/CR-4491: DEVELOPMENT OF MODELS FOR WARM PRESTRESSING. STONESIFER,R.B. Computational Mechanics. RYBICKI,E.F. PROSIG, Inc. * Materials Engineering Associates, Inc. January 1987. 78pp. 8702060276. MEA-2122. 39538:107.

The objective of this project is to evaluate available mathematical models and associated fracture criteria for predicting warm prestress (WPS) effects. A verified model of the WPS phenomenon is required before credit for improved low temperature toughness can be taken in analysis of postulated accident scenarios such as pressurized thermal shock. The primary basis of evaluation is finite- element analysis using a highly refined mesh and work hardening, modeled by a piece-wise linear fit of stress-strain data. The criteria being evaluated are J(e), (Chell, et al.), critical stress (Curry), T*(p) (Atluri) and a criterion introduced herein which is related to differential CTOD and denoted dCTOD*FLOW. The finite element model is used to simulate a load-unload-cool-fracture (LUCF) type of WPS cycle for which experimental results are available. The various models and criteria are evaluated in terms of their agreement with the finite-element results such as crack opening displacements, stresses and plastic-zone sizes, and in terms of their ability to predict fracture load. The nonfinite-element-based models of Chell and Curry are used to simulate 32 additional WPS experiments so as to further assess the relative merits of the models and the

J(e) critical stress, and dCTOD*FLOW fracture criteria. While K(Ic) scatter band behavior allows significant latitude for manipulation of model predictions, which impedes critical evaluation of the models and criteria, both models and all three fracture criteria are found to predict WPS behavior which is qualitatively consistent with experimental data.

NUREG/CR-4524: CLOSEOUT OF IE BULLETIN 80-24:PREVENTION OF DAMAGE DUE TO WATER LEAKAGE INSIDE CONTAINMENT (OCTOBER 17,1980 INDIAN POINT 2 EVENT). FOLEY,W.J.; DEAN,R.S.; HENNICK,A. Parameter, Inc. March 1987. 51pp. 8704090025. IEB-80-24. 40464:058.

On October 24, 1980, IE Information Notice 80-37 was issued by the NRC to describe reactor vessel pit flooding which had been discovered a week earlier at Indian Point 2. The lower nine feet of the reactor vessel had been wetted while at operating temperature, and a thermal stress condition of potential safety significance had been caused. IE Bulletin 80-24 was issued by the NRC on November 21, 1980 because of concern about this event. Licensees of operating power reactors were required to take short term actions to ensure continued interim operation without containment flooding. The long term purpose of the bulletin was to obtain operating data on which to base future NRC requirements for generic corrective actions. The bulletin was issued to holders of construction permits for information. Inspection requirements for reviewing licensee actions were clarified by issuing Temporary Instruction 2515/47 on December 18, 1980, and a special memorandum on February 19, 1981. Evaluation of utility responses, licensee event reports, an NRC/IE memorandum, NRC/IE inspection reports and an NRC/IE letter shows that the bulletin can be closed per specific criteria for all of the 69 facilities to which it was issued for action, and that no further action is necessary.

NUREG/CR-4531: AN INVESTIGATION OF INTEGRAL FACILITY SCALING AND DATA RELATION METHODS (INTEGRAL SYSTEM TEST PROGRAM). LARSON,T.K. EG&G Idaho, Inc. (subs. of EG&G, Inc.). February 1987. 129pp. 8704080137. EGG-2440. 40440:178.

The Integral Systems Test Program was initiated in 1982 by government and industry in response to the Three Mile Island accident. Three different integral test facilities, each scaled to a Babcock and Wilcox design nuclear steam supply system, will ultimately contribute data to the program. Each of the facilities was designed using different scaling methodologies, and each has different operating capabilities. The overall scaling of each facility is examined in this report, and local scaling is analyzed to demonstrate potential similarities and dissimilarities in facility response relative to expected plant responses. The scaling relationships are used to show how local thermal-hydraulic phenomena in each facility can be compared to each other or to expected plant behavior. The concept of an equilibrium plot is used to show how the global response of each facility can be related for a specific small break loss-of-coolant transient. Potential complications that may arise as a consequence of the facility scaling or facility limitations are enumerated. The potential use of dimensionless groupings for relating and specifying experiments is discussed. Finally, some specific experiments and conditions are proposed for the purpose of simplifying interfacility comparison of test results.

NUREG/CR-4541: EXPERIMENTAL ASSESSMENT OF THE SEALING EFFECTIVENESS OF ROCK FRACTURE GROUTING. SCHAFFER,A.; DAEMEN,J.J. Arizona, Univ. of, Tucson, AZ. March 1987. 193pp. 8704010149. 40340:050.

The objective of this investigation is to determine the effectiveness of cement grouts as sealants of fractures in rock. Laboratory experiments have been conducted on seven 15-cm granite cubes containing saw cuts, three 23-cm diameter andesite cores containing induced tension cracks, and one 15-cm diameter marble core containing a natural fracture. Prior to grouting, the hydraulic conductivity of the fractures is determined under a range of normal stresses, applied in the loading and unloading cycles, from 0 to 14 MPa (2000 psi). Grout is injected through an axial borehole, at a pressure of 1.2 to 8.3 MPa (180 to 1200 psi), pressure selected to provide a likely groutable fracture aperture, while the fracture is stressed at a constant normal stress. The fracture permeability is measured after grouting. Flow tests on the ungrouted samples confirm the inverse relation between normal stress and fracture permeability. The equivalent aperture determined by these tests is a reliable indicator of groutability. Post-grouting permeability measurements as performed here, and frequently in practice, can be misleading, since incomplete grouting of fractures can result in major apparent reductions in permeability. The apparent permeability reduction is caused by grouting of a small area of a highly preferential flowpath directly adjacent to the hole used for grouting and for permeability testing. Experimental results confirm claims in the literature that ordinary portland cement inadequately penetrates fine fractures.

NUREG/CR-4550 V03: ANALYSIS OF CORE DAMAGE FREQUENCY FROM INTERNAL EVENTS:SURRY UNIT 1. BERTUCIO,R.C.; QUILICI,M.D.; YOUNG,J.; et al. Sandia National Laboratories. November 1986. 450pp. 8704130170. SAND86-2084. 40512:157.

This document contains the accident sequence for Surry, 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 Surry, Unit 1, and the accompanying plant damage state frequencies. Sensitivity and uncertainty analyses provide additional insights regarding the dominant contributors to the Surry core damage frequency estimate. The mean core damage frequency at Surry was calculated to be 2.6E-5 per year. Station blackout type accidents (loss of all AC power) were the largest contributors to core damage frequency, accounting for approximately 38% of the total. The next type of dominant contributors were transient induced LOCAs caused by loss of electrical bus initiators. These sequences account for 19% of core damage frequency. No other type of sequence accounts for more than 10% of core damage frequency. The numerical results are driven to some degree by modeling assumptions and data selection for issues such as reactor coolant pump seal LOCAs, common cause failure probabilities, and plant response to station blackout and loss of electrical bus initiators. 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 V04: ANALYSIS OF CORE DAMAGE FREQUENCY FROM INTERNAL EVENTS:PEACH BOTTOM UNIT 2. KOLACZKOWSKI,A.; LAMBRIGHT,J.A.; FERRELL,W.L.; et al. Sandia National Laboratories. October 1986. 663pp. 8703090175. SAND86-2084. 39919:315.

This document contains the internal event initiated accident sequence analyses for Peach Bottom, Unit 2; 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 Peach Bottom, Unit 2, and the accompanying plant damage state frequencies. Sensitivity and uncertainty analyses provide additional insights regarding the dominant contributors to the Peach Bottom core damage frequency estimate. The mean core damage frequency at Peach Bottom was calculated to be 8.2E-6. Station Blackout type accidents (loss of all AC power) were found to dominate the overall results. Anticipated Transients Without Scram accidents were also found to be non- negligible contributors. The numerical results are largely driven by common mode failure probability estimates and to some extent, human error. Because of significant data and analysis uncertainties in these two areas (important, for instance, to the most dominant scenario in this study), it is recommended that the results of the uncertainty and sensitivity analyses be considered before any actions are taken based on this analysis.

NUREG/CR-4551 V1 DRF: EVALUATION OF SEVERE ACCIDENT RISKS AND THE POTENTIAL FOR RISK REDUCTION:SURRY POWER STATION,UNIT 1.Draft For Comment. BENJAMIN,A.S.; BOYD,G.J.; KUNSMAN,D.M.; et al. Sandia National Laboratories. February 1987. 720pp. 8703170262. SAND86-1309. 40044:071.

The Severe Accident Risk Reduction Program (SARRP) has completed a rebaselining of the risks to the public from a particular pressurized water reactor with a subatmospheric containment (Surry, 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 lower than previously evaluated in the Reactor Safety Study (RSS). However, certain unresolved issues (such as direct containment heating) caused the top of the uncertainty band to appear at a level that is comparable with the RSS point estimate. None of the postulated safety options appears to be cost effective for the Surry power plant. This work supports the Nuclear Regulatory Commission's assessment of severe accidents in NUREG-1150.

NUREG/CR-4552: A REVIEW OF THE SEABROOK STATION PROBABILISTIC SAFETY ASSESSMENT.Containment Failure Modes And Radiological Source Terms. KHATIB-RAHBAR; AGRAWAL,A.K.; LUDEWIG,H.; et al. Brookhaven National Laboratory. March 1987. 75pp. 8704090039. BNL-NUREG-51961. 40463:244.

A technical review and evaluation of the Seabrook Station Probabilistic Safety Assessment has been performed. It is determined that (1) containment response to severe core melt accidents is judged to be an important factor in mitigating the consequences, (2) failure during the first few hours after core melt is also unlikely and the timing of overpressure failure is very long compared to WASH-1400, (3) the point-estimate radiological releases are comparable in magnitude to those used in WASH-1400, and (4) the energy of release is somewhat higher than for the previously reviewed studies.

NUREG/CR-4610: EFFECTS OF LATERAL SEPARATION OF OXIDIC AND METALLIC CORE DEBRIS ON THE BWR MK I CONTAINMENT DRYWELL FLOOR. HYMAN,C.R.; WEBER,C.F. Oak Ridge National Laboratory. January 1987. 118pp. 8704080065. ORNL/TM-10057. 40444:007.

In evaluating core debris/concrete for a BWR MK I containment design, it has been common practice to assume that at reactor vessel breach, the core debris is homogeneous and of low viscosity so that it is uniformly distributed radially on the drywell floor. In a recent study performed by the NRC-sponsored BWR Severe Accident Technology (BWRSAT) program at Oak Ridge National Laboratory, calculations indicate that at reactor vessel bottom head failure, the debris is such that the metallic components (Zr, Fe, Ni, Cr) are completely molten while the oxidic components (UO_2, ZrO_2, FeO) are completely frozen. Thus, the frozen oxides are expected to remain within the reactor pedestal while the molten metallic species radially separate from the frozen oxidic species, flow through the opening in the reactor pedestal, and spread over the annular region of the drywell floor between the pedestal and the containment shell. This report assesses the impact on calculated containment response and the production and release of fission product-laden aerosols for two different cases of debris distribution: Uniform distribution and the laterally separated case of 95% oxides-5% metals inside the pedestal and 5% oxides-95% metals outside the pedestal. The computer codes used in this assessment are CORCON-MOD2, MARCON 2.1B, and VANESA.

NUREG/CR-4613: EVALUATION OF NUCLEAR POWER PLANT OPERATING PROCEDURES CLASSIFICATIONS AND INTERFACES.Problems And Techniques For Improvement. BARNES,V.E.; RADFORD,L.R. Battelle Human Affairs Research Centers. * Battelle Memorial Institute, Pacific Northwest Laboratories. February 1987. 120pp. 8703120284. PNL-5852. 39983:256.

This report presents activities and findings of a project designed to evaluate current practices and problems related to procedure classification schemes and procedure interfaces in commercial nuclear power plants. The phrase "procedure classification scheme" refers to how plant operating procedures are categorized and indexed (e.g., normal, abnormal, emergency operating procedures). The term "procedure interface" refers to how reactor operators are instructed to transition within and between procedures. The project consisted of four key tasks, including (1) a survey of literature regarding problems associated with procedure classifications and interfaces within and between procedures, as well as techniques for overcoming them; (2) interviews with experts in the nuclear industry to discuss the appropriate scope of different classes of operating procedures and techniques for managing interfaces between them; (3) a reanalysis of data gathered about nuclear power plant normal operating and off-normal operating procedures in a related project, "Program Plan for Assessing and Upgrading Operating Procedures for Nuclear Power Plants"; and (4) solicitation of the comments and expert opinions of a peer review group on the draft project report and on proposed techniques for resolving classification and interface issues. In addition to describing these activities and their results, recommendations for the NRC and utility actions to address procedure classification and interface problems are offered.

NUREG/CR-4616: ROOT CAUSES OF COMPONENT FAILURES PROGRAM.Methods And Applications. SATTERWHITE,D.; CADWALLADER,L.; MEALE,B.M.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). December 1986. 60pp. 8702060177. EGG-2455. 39526:227.

This report contains information pertaining to definitions, methodologies, and applications of root cause analysis. Of specific interest, and highlighted throughout the discussion, are applications pertaining to current and future Nuclear Regulatory Commission (NRC) light water safety programs. These applications are discussed in view of addressing specific program issues under NRC consideration and reflect current root cause analysis capabilities.

NUREG/CR-4626 V02: IMPROVING THE RELIABILITY OF OPEN-CYCLE WATER SYSTEMS. Application Of Biofouling Surveillance And Control Techniques To Sediment And Corrosion Fouling At Nuclear Power Plants. JOHNSON,K.I.; NEITZEL,D.A. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1987. 55pp. 8703300146. PNL-5876. 40280:224.

Biofouling surveillance and control techniques are evaluated for their applicability to sediment and corrosion fouling and suggestions are given to improve their effectiveness. Alternative techniques to better detect and control sedimentation and corrosion are also evaluated. Environmental conditions that allow biofouling, sedimentation, and corrosion to occur are summarized. A correlation between sediment and corrosion is identified and the causes are described. Environmental regulations, especially those in the Clean Water Act of 1977, are reviewed to identify those that may limit or prevent the use of surveillance and control techniques described in this report. Flow velocity is the major design factor that determines whether or not biofouling, sedimentation, and corrosion will occur. Monitoring flow conditions can provide early warning of conditions that will allow fouling to occur. Visual inspection is the most common and most effective technique for identifying the cause and extent of fouling in the open-cycle water system. Most biofouling control techniques in current use are not effective against sediment and corrosion. Frequent, high-velocity flushing of cooling loops may effectively remove sediment and reduce under-sediment corrosion. Alternative biocide treatments such as targeted chlorination or the use of ozone or 2, 2-dibromo-3-nitrilo propionamide (DBNPA) may also be effective in reducing under-sediment corrosion.

NUREG/CR-4672: ANALYSIS OF INSTRUMENT TUBE RUPTURES IN WESTINGHOUSE 4-LOOP PRESSURIZED WATER REACTORS. FLETCHER,C.D.; BOLANDER,M.A. EG&G Idaho, Inc. (subs. of EG&G, Inc.). December 1986. 51pp. 8702060201. EGG-2461. 39527:310.

A recent safety concern for Westinghouse 4-loop pressurized water reactors (PWRs) is that, because of a seismic event, instrument tubes may be broken at the flux mapping seal table, resulting in an uncovering and heatup of the reactor core. This study's purpose was to determine the effects upon findings of a similiar 1980 study if certain test variables changed. A 1980 U.S. Nuclear Regulatory Commission (USNRC) analysis of PWR behavior used the RELAP4/MOD7 computer code to determine the effects of breaking instrument tubes at the reactor vessel lower plenum wall. The 1986 study discussed here was performed using RELAP5/MOD2, an advanced best-estimate computer code. Separate effects analyses investigated instrument tube pressure loss, heat loss, and tube nodalization effects on break flow. Systems effects analyses: (a) investigated the effects of changing the break location from the reactor vessel to the seal table, (b) compared RELAP4/MOD7 and RELAP5/MOD2 results for an identical transient, (c) verified a key finding from the 1980 analysis, and (d) investigated instrument tube ruptures in th Zion-1 PWR using best-estimate boundary and initial conditions. The outcome of these analyses permits adjustment of the 1980 analysis findings for instrument tube ruptures at seal table and indicates the best-estimate response of a Westinghouse PWR to the rupture of 25 small instrument tubes at the seal table.

NUREG/CR-4685: POST-PLIOCENE DISPLACEMENT ON FAULTS WITHIN THE KENTUCKY RIVER FAULT SYSTEM OF EAST-CENTRAL KENTUCKY. VANARSDALE,R.B.; SERGEANT,R.E. Kentucky, Univ. of, Lexington, KY. February 1987. 47pp. 8703130193. 40007:088.

The Kentucky River Fault System forms the northern boundary of the Rome Trough (a Paleozoic aulacogen) in east-central Kentucky. Paleozoic recurrent movement along this fault system has been documented by a number of previous workers; however, recognition of Mesozoic and early Tertiary displacement has not been possible due to the absence of preserved post-Paleozoic strata. Numerous faults of the Kentucky River Fault System are partially overlain by Pliocene- Pleistocene terrace deposits along the Kentucky River. Results from preliminary drilling and electrical-resistivity surveys indicate that a number of these faults may have been active since deposition of terrace materials. Based on indications from the preliminary survey, four sites were selected and nine trenches were excavated. Of these nine trenches, four revealed faulted or folded terrace sediments. Comparison of the nine trenches suggests that the folding and faulting of the terrace deposits is tectonic in origin and that the Kentucky River Fault System has been active within the last 5 million years and probably within the last 1 million years.

NUREG/CR-4689: THERMAL-HYDRAULIC AND CHARACTERISTIC MODELS FOR PACKED DEBRIS BEDS. MUELLER,G.E.; SOZER,A. Oak Ridge National Laboratory. January 1987. 108pp. 8702060116. ORNL/TM-10117. 39537:131.

APRIL is a mechanistic core-wide meltdown and debris relocation computer code for Boiling Water Reactor (BWR) severe accident analyses. The capabilities of the code continue to be increased by the improvement of existing models. This report contains information on theory and models for degraded core packed debris beds. The models, when incorporated into APRIL, will provide new and improved capabilities in predicting BWR debris bed coolability characteristics. These models will allow for a more mechanistic treatment in calculating temperatures in the fluid and solid phases in the debris bed, in determining debris bed dryout, debris bed quenching from either top-flooding or bottom-flooding, single and two-phase pressure drops across the debris bed, debris bed porosity, and in finding the minimum fluidization mass velocity. The inclusion of these models in a debris bed computer module will permit a more accurate prediction of the coolability characteristics of the debris bed and therefore reduce some of the uncertainties in assessing the severe accident characteristics for BWR application. Some of the debris bed theoretical models have been used to develop a FORTRAN 77 subroutine module called DEBRIS. DEBRIS is a driver program that calls other subroutines to analyze the thermal characteristics of a packed debris bed. FORTRAN 77 listings of each subroutine are provided in the appendix.

NUREG/CR-4695: MATERIAL CONTROL AND ACCOUNTING (MC&A) LOSS DETECTION DURING TRANSITION PERIODS AND PROCESS UPSET CONDITIONS. GRIFFIN,E.A.; YOUNG,J.K.; SMITH,B.W.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1987. 62pp. 8704130187. PNL-5890. 40494:263.

The Nuclear Regulatory Commission has implemented regulations that require licensees to perform tests to detect significant losses of strategic special nuclear material on a timely basis and to resolve anomalies resulting from such tests. These capabilities have been demonstrated for processes operating at equilibrium; however, the conditions that exist during transition periods, i.e., at startup and shutdown, and during process upsets will impact a licensee's ability to achieve the specified levels of loss detection and alarm resolution. This report discusses the types of data available, potentially useful loss tests, and techniques that can be used in developing models for the abnormal conditions.

NUREG/CR-4696: CONTAINMENT VENTING ANALYSIS FOR THE PEACH BOTTOM ATOMIC POWER STATION. HANSON,D.J.; BLACKMAN,H.S.; NELSON,W.R.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). February 1987. 82pp. 8704070476. EGG-2464. 40438:227.

The extent to which containment venting is an effective means of preventing or mitigating the consequences of overpressurization during severe accidents was evaluated for the Peach Bottom Atomic Power Station Units 2 and 3 (boiling water reactors with Mark I containments). Detailed analyses were conducted on operator performance, equipment performance, and the physical phenomenology for three severe accident sequences currently identified as being important contributors to risk. The results indicate that containment venting can be effective in reducing risk for several classes of severe accidents but, based on procedures in draft form and equipment in place at the time of the analyses, has limited potential for further reducing the risk for severe accidents currently identified as being important contributors to the risk for Peach Bottom.

NUREG/CR-4700 V1 DRF: CONTAINMENT EVENT ANALYSIS FOR POSTULATED SEVERE ACCIDENTS. Surry Power Station,Unit 1.Draft For Comment. BENJAMIN,A.S.; BEHR,V.L.; KUNSMAN,D.M.; et al. Sandia National Laboratories. February 1987. 195pp. 8703170196. SAND86-1135. 40054:302.

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 a subatmospheric containment (Surry Unit 1) to postulated severe accidents. A detailed containment event tree for the Surry 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 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 at the time of reactor vessel breach) was found to be very low for most accident sequence initiators. However, uncertainties surrounding the issues of direct containment pressure capacity could lead to much higher early failure likelihoods. This work supports NRC's assessment of severe accident risks to be published in NUREG-1150.

NUREG/CR-4708 V01 N1: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Semiannual Report For October 1985 - March 1986. MEYER,R.E.; ARNOLD,W.D.; BLENCOE,J.G.; et al. Oak Ridge National Laboratory. January 1987. 63pp. 8702170050. ORNL/TM-10147. 39655:245.

Information that is being developed by projects within the Department of Energy (DOE) pertinent to the potential geochemical behavior of radionuclides at candidate sites for a high-level radioactive waste repository is being evaluated by Oak Ridge National Laboratory (ORNL) for the Nuclear Regulatory Commission (NRC). During this report period emphasis was placed upon the Yucca Mountain, Nevada, site. Several samples of tuff were analyzed and characterized by X-ray diffraction and petrographic analysis. Initial sorption experiments with cesium and strontium demonstrated the necessity of preequilibration and control of the CO(2) partial pressure over the experiment. A small particle size effect was observed for sorption of N(p) on various size fractions of tuff. Difficulties were experienced in preparing standard solutions of europium. Preliminary comparison of our data for sorption of cesium and strontium with those of NNWSI showed that sorption ratios were similar for approximately the same conditions. One of our principal concerns with the NNWSI data is that much of their data was taken without control of CO(2) partial pressure.

NUREG/CR-4711: LOW UPPER-SHELF TOUGHNESS,HIGH-TRANSITION TEMPERATURE TEST INSERT IN HSST PTSE-2 VESSEL AND WIDE-PLATE TEST SPECIMENS. Final Report. DOMIAN,H.A. Babcock & Wilcox Co. * Oak Ridge National Laboratory. February 1987. 85pp. 8704130637. 40495:015.

A piece of A387, Grade 22 Class 2 (2-1/4 Cr - 1 Mo) steel plate specially heat treated to produce low upper-shelf (LUS) toughness and high-transition temperature was installed in the side wall of Heavy Section Steel Technology (HSST) vessel V-8. This vessel is to be tested by the Oak Ridge National Laboratory (ORNL) in the Pressurized- Thermal-Shock Experiment-2 (PTSE-2) project of the HSST program. Comparable pieces of the plate were made into six wide-plate specimens and other samples for characterization testing to be performed by ORNL.

NUREG/CR-4712: REGULATORY ANALYSIS OF REGULATORY GUIDE 1.35 (REVISION 3, DRAFT 2) - IN-SERVICE INSPECTION OF UNGROUTED TENDONS IN PRESTRESSED CONCRETE CONTAINMENTS. NAUS,D.J. Oak Ridge National Laboratory. February 1987. 123pp. 8704130262. ORNL/TM-10163. 40495:206.

The objectives of this study were to review all the changes in the latest version (Rev. 3, Draft 2) of "Regulatory Guide 1.35" and to provide a regulatory analysis for all positions in the guide. Three tasks were undertaken to meet these objectives: (1) review of the evolution of prestressed concrete containments, their prestressing systems and the development of "Regulatory Guide 1.35"; (2) development of a comparative regulatory analysis between Rev. 3 (Draft 2) and the current version, which is in effect (Rev. 2); and (3) conduction of a backfit analysis in conformance with the requirements of the Backfitting Rule (Section 50.109). Results of the study indicate that there are certain areas where additional costs may be incurred by industry due to implementation of Rev. 3; however, a reassessment of requirements in other areas can produce an estimated net cost savings to industry in excess of $4,500,000 in terms of net discounted future cost impact when a discount rate of 5% is used. Also, although changes in the guide were determined to produce an unquantifiable change in risk, it is anticipated that the changes will have a positive impact on safety and thus will lower the risk and should enhance containment availability.

NUREG/CR-4713: SHUTDOWN DECAY HEAT REMOVAL ANALYSIS OF A BABCOCK AND WILCOX PRESSURIZED WATER REACTOR.Case Study. CRAMOND,W.R.; ERICSON,D.M.; SANDERS,G.A. Sandia National Laboratories. March 1987. 800pp. 8704010504. SAND86-1832. 40325:245.

This is one of six case studies for USI A-45 Decay Heat Removal (DHR) Requirements. The purpose of this study is to identify any potential vulnerabilities in the DHR systems of a typical Babcock and Wilcox PWR, to suggest possible modifications to improve DHR capability, and to assess the value and impact of the most promising alternatives to the existing DHR systems. The systems analysis considered small LOCAs and transient internal initiating events, and seismic, fire, extreme wind, internal and external flood, and lightning external events. A full-scale systems analysis was performed with detailed fault trees and event trees including support system dependencies. The system analysis results were extrapolated into release categories using applicable past PRA phenomenological results and improved containment failure mode probabilities. Public consequences were estimated using site-specific CRAC2 calculations. The value-impact (VI) analysis of possible alternatives considered both onsite and offsite impacts arriving at several risk measures such as averted population dose out to a 50-mile radius and dollars per person rem averted. Uncertainties in the VI analysis are discussed and the issues of feed and bleed and secondary blowdown are analyzed.

NUREG/CR-4718: EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES PROGRAM.Summary Report. LANNING,D.D. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1987. 53pp. 8702170067. PNL-5972. 39656:051.

This report summarizes the activities and results of the 11-year Experimental Support and Development of Single-Rod Codes Program, sponsored at Pacific Northwest Laboratory by the U.S. Nuclear Regulatory Commission. The program included irradiation of extensively instrumented test fuel assemblies at the Halden Reactor in Norway and postirradiation examination at Harwell Laboratories in the United Kingdom; ex-reactor studies on gap conductance and fuel rod mechanics; and model development for the FRAPCON-2 fuel performance computer code. Significant results included long-term in-reactor data on fuel temperature, fission gas release, and rod elongation; quantification of the inherent scatter in observed fuel temperatures in replicate rods; and definition and modeling of the thermal and mechanical consequences of fuel pellet cracking and fragment outward relocation. The data obtained are used as benchmark data for fuel performance codes and models throughout the world.

NUREG/CR-4724: FATIGUE CRACK GROWTH RATES IN PRESSURE VESSEL AND PIPING STEELS IN LWR ENVIRONMENTS.Final Report. CULLEN,W.H. Materials Engineering Associates, Inc. March 1987. 68pp. 8704010118. MEA-2175. 40321:347.

The measurement of fatigue crack growth rates for pressure and piping steels in high-temperature, pressurized water has been carried out using compact fracture specimens. Over the last ten years, the programs sponsored by the NRC and carried out at the Naval Research Laboratory and Materials Engineering Associates have provided data for over three hundred tests of these specimens, which have been published in a series of NUREG topical reports and annual reports. This is the final report in this series and describes briefly the significant findings of the program, reports on the most recent data which have been acquired, and indicates some directions for future research in this area. Further testing of compact specimens have been nearly phased out, and the program has turned more towards applications- oriented tasks, such as variable cyclic amplitude testing, part- through crack geometry tests, and environmental effects on the stress- life curves.

NUREG/CR-4730: EVALUATION OF POTENTIAL MIXED WASTES CONTAINING LEAD, CHROMIUM,USED OIL,OR ORGANIC LIQUIDS. SISKIND,B.; MACKENZIE,D.R.; BOWERMAN,B.S.; et al. Brookhaven National Laboratory. January 1987. 175pp. 8702270062. BNL-NUREG-52019. 39759:249.

This report presents the results of follow-on studies conducted by Brookhaven National Laboratory (BNL) for the Nuclear Regulatory Commission (NRC) on certain kinds of low-level waste (LLW) which could also be classified as hazardous waste subject to regulation by the Environmental Protection Agency (EPA). Such LLW is termed "mixed waste". Additional data have been collected and evaluated on two categories of potential mixed waste, namely LLW containing metallic lead and LLW containing chromium. Additionally, LLW with organic liquids, especially liquid scintillation wastes, are reviewed. In light of a proposed EPA rule to list used oil as hazardous waste, the potential mixed waste hazard of used oil contaminated with radionuclides is discussed.

NUREG/CR-4734: SEISMIC TESTING OF TYPICAL CONTAINMENT PIPING PENETRATION SYSTEMS. CLOSE,J.A.; HILL,R.C.; STEELE,R. Idaho National Engineering Laboratory. December 1986. 41pp. 8702060293. EGG-2470. 39527:049.

This report provides the results of seismic tests of three typical light water reactor containment penetration systems to provide a technical basis for the support and development of equipment qualification procedures. The three systems tested: (a) An eight-inch gate valve system modeling a containment spray system; (b) An eight- inch butterfly valve system modeling a purge and vent system, and (c) A two-inch globe valve system modeling the numerous small bore piping systems that are often characterized by high valve to pipe size ratio. The valve types, sizes, piping configurations, penetrations and supports used for the tests are typical of those found in commercial U.S. nuclear power plants for containment isolation applications. The three systems tested were mounted in a fixture and excited with simulated seismic loads. The loads imposed during the tests were equal or greater than those expected during U.S. operating-basis and safe-shutdown earthquakes. The test results indicate that adverse valve, penetration, or piping system behavior during typical seismic events is very unlikely.

NUREG/CR-4735 V01: EVALUATION AND COMPILATION OF DOE WASTE PACKAGE TEST DATA. Biannual Report:December 1985 - July 1986. INTERRANTE,C.; ESCALANTE,E.; FRAKER,A.; et al. Commerce, Dept. of, National Bureau of Standards. March 1987. 120pp. 8704010532. 40325:106.

This report summarizes results to date of NBS evaluations of Department of Energy (DOE) activities in waste packages designed for containment of radioactive high-level nuclear waste (HLW). The waste package is a proposed engineering barrier that is part of a permanent repository for HLW. Candidate repository sites include three different media: tuff, basalt, and salt. Metal alloys are the principal barriers for the proposed canisters and overpacks. In addition, borosilicate glass and various packing materials have been proposed as components of this engineering system. Thus, the associated technical problems involve corrosion, leaching, dissolution and transport within the waste packages. This report gives status reports on waste package activities related to each of the three host media. Appended to the report are NBS reviews of selected DOE technical reports and NBS trip reports of pertinent meetings, seminars, and workshops attended. Also presented in the report is background information on the Materials Characterization Center (MCC) as well as discussion on statistical considerations in fitting leaching and corrosion models to measurements. The MCC was established to assess and characterize waste package materials for reliable performance for DOE's nuclear waste needs.

NUREG/CR-4736: COMBUSTION AEROSOLS FORMED DURING BURNING OF RADIOACTIVELY CONTAMINATED MATERIALS - EXPERIMENTAL RESULTS. HALVERSON,M.A.; BALLINGER,M.Y.; DENNIS,G.W. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1987. 61pp. 8703250500. PNL-5999. 40234:279.

Safety assessments and environmental impact statements for nuclear fuel cycle facilities require an estimate of potential airborne releases. Radioactive aerosols generated by fires were investigated in experiments in which combustible solids and liquids were contaminated with radioactive materials and burned. Uranium in powder and liquid form was used to contaminate five fuel types: polychloroprene, polystyrene, polymethylmethacrylate, cellulose, and a mixture of 30% tributylphosphate (TBP) in kerosene. Heat flux, oxygen concentration, air flow, contaminant concentration, and type of ignition were varied in the experiments. The highest release (7.1 wt%) came from burning TBP/kerosene over contaminated nitric acid. Burning cellulose contaminated with uranyl nitrate hexahydrate liquid gave the lowest release (0.01 wt%). Rate of release and particle size distribution of airborne radioactive particles were highly dependent on the type of fuel burned.

NUREG/CR-4737: INTERPRETATIVE ANALYSIS OF DATA FOR SOLUTE TRANSPORT IN THE UNSATURATED ZONE. FUENTES,H.R.; POLZER,W.L. Los Alamos Scientific Laboratory. January 1987. 242pp. 8702170076. LA-10817-MS. 39655:003.

In this report, the movement of iodide, bromide, and lithium under unsaturated flow conditions is modeled using the computer code CFITIM. This code is a solution of the one-dimensional convective- dispersive equation when steady-state flow exists and when interactions between the solute and Bandelier tuff can be described by the linear isotherm. The model predicts well the transport of the solutes iodide, bromide, and lithium when flow conditions are near steady-state. When assuming average steady-state flow conditions, the model predicts dispersion factors for unsteady flow within one to two orders of magnitude of the predictions at steady-state flow; retardation factors, on the other hand, are predicted much better than the dispersion factors. Differences in the estimated dispersion coefficients for solutes of two steady-state pulses indicate that the intended replication of those steady-state flow pulses was not achieved during experimentation. A comparison of breakthrough curves of solutes from one depth to another in the 3-m x 6-m field experimental caisson indicates poor conservation of solute mass during transport.

NUREG/CR-4741: FEEDWATER TRANSIENT AND SMALL BREAK LOSS OF COOLANT ACCIDENT ANALYSES FOR THE BELLEFONTE NUCLEAR PLANT. BAYLESS,P.D.; DOBBE,C.A.; CHAMBERS,R. EG&G Idaho, Inc. (subs. of EG&G, Inc.). March 1987. 110pp. 8704130238. EGG-2471. 40496:015.

Specific sequences that may lead to core damage were analyzed for the Bellefonte nuclear plant as part of the U.S. Nuclear Regulatory Commission's Severe Accident Sequence Analysis Program. The RELAP5, SCDAP, and SCDAP/RELAP5 computer codes were used in the analyses. The two main initiating events investigated were a loss of all feedwater to the steam generators and a small cold leg break loss of coolant accident. The transients of primary interest within these categories were the TMLB' and S(2)D sequences. Variations on systems availability were also investigated. Possible operator actions that could prevent or delay core damage were identified, and two were investigated for a small break transient. All of the transients were analyzed until either core damage began or long-term decay heat removal was established. The analyses showed that for the sequences considered the injection flow from one high-pressure injection pump was necessary and sufficient to prevent core damage in the absence of operator actions. Operator actions were able to prevent core damage in the S(2)D sequence; no operator actions were available to prevent core damage in the TMLB' sequence.

NUREG/CR-4742: MELPROG-PWR/MOD1 ANALYSIS OF A TMLB' ACCIDENT SEQUENCE. KELLY,J.E.; HENNINGER,R.J.; DEARING,J.F. Sandia National Laboratories. January 1987. 83pp. 8704070501. SAND86-2175. 40437:114.

The first complete, coupled, and mechanistic analysis of a TMLB' (station blackout) core meltdown accident for the Surry plant has been made with MELPROG-PWR/MOD1. This analysis has provided the timing of the major events occurring in the accident, the amount and timing of hydrogen produced by oxidation of the cladding, and the condition and composition of the disrupted material at the time of vessel failure. Due to the preliminary nature of this first calculation, a limited number of auxiliary calculations have been performed. Comparison of these results with previous calculations have provided further insights into this accident. In particular, it is shown that natural convection reduces the rate of core heating, but increases the rate of heating of upper plenum structures. This increased heating can inhibit fission product deposition and increase the amount of molten structural steel in the melt at vessel failure. It is also shown that coupling between vessel flow and primary system flow may lead to early heating and failure of the primary system. Hence, natural circulation within the vessel with coupling to the primary system can completely change the course and timing of a meltdown sequence. This underlines the importance of a multi-dimensional vessel flow capability as provided by MELPROG. In addition, the effect of the modeling of the initial fuel rod melting and relocation has been studied. Variations in the assumptions were found to strongly affect hydrogen production and the subsequent course and timing of the accident.

NUREG/CR-4744 V01 N1: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,October 1985 - March 1986. CHOPRA,O.K.; CHUNG,H.M. Argonne National Laboratory. January 1987. 47pp. 8704080064. ANL-86-54. 40441:290.

This progress report summarizes work performed by Argonne National Laboratory during the six months from October 1985 to March 1986 on long-term embrittlement of cast duplex stainless steels used in light-water reactors.

NUREG/CR-4752: COINCIDENT STEAM GENERATOR TUBE RUPTURE AND STUCK-OPEN SAFETY RELIEF VALVE CARRYOVER TESTS.MB-2 Steam Generator Transient Response Test Program. GARBELT,K.; MENDLER,O.J.; GARDNER,G.C.; et al. Westinghouse Electric Corp. March 1987. 630pp. 8704010141. EPRI NP-4787. 40319:168.

In PWR steam generator tube rupture (SGTR) faults, a direct pathway for the release of radioactive fission products can exist if there is a coincident stuck-open safety relief valve (SORV) or if the safety relief valve is cycled. The test program consisted of sixteen separate tests designed to cover a range of steady-state and transient fault conditions. The main conclusions from these tests were that moisture carryover was very low in the absence of an SGTR, that there was no significant increase in moisture carryover during an SGTR/SORV fault and that very little or no primary coolant passed through the steam generator without having first completely mixed with the bulk secondary liquid (primary coolant bypassing). Short-term perturbations to steady-state conditions were found to produce transient releases, which could be mainly due to primary coolant bypassing or these releases were the equivalent of steady-state releases over tens of hours, and could be important factors in determining the overall activity release in these types of fault. At very low water levels, when recirculation within the boiler could not be maintained, conditions typical of early stages in an SGTR/SORV fault produced large transient releases.

NUREG/CR-4762: SHUTDOWN DECAY HEAT REMOVAL ANALYSIS OF A WESTINGHOUSE 3-LOOP PRESSURIZED WATER REACTOR.Case Study. SANDERS,G.A.; ERICSON,D.M.; CRAMOND,W.R. Sandia National Laboratories. March 1987. 350pp. 8704010158. SAND86-2377. 40322:055.

This is one of six case studies for USI A-45 Decay Heat Removal (DHR) Requirements. The purpose of this study is to identify any potential vulnerabilities in the DHR systems of a typical Westinghouse 3-loop PWR, to suggest possible modifications to improve the DHR capability, and to assess the value and impact of the most promising alternatives to the existing DHR systems. The systems analysis considered small LOCAs and transient internal initiating events, and seismic, fire, extreme wind, internal and external flood, and lightning external events. A full-scale systems analysis was performed with detailed fault trees and event trees including support system dependencies. The system analysis results were extrapolated into release categories using applicable past PRA phenomenological results and improved containment failure mode probabilities. Public consequences were estimated using site specific CRAC2 calculations. The Value-Impact (VI) analysis of possible alternatives considered both onsite & offsite impacts arriving at several risk measures such as averted population dose out to a 50-mile radius and dollars per person rem averted. Uncertainties in the VI analysis are discussed and the issues of feed and bleed and secondary blowdown are analyzed.

NUREG/CR-4776: RESPONSE OF SEISMIC CATEGORY I TANKS TO EARTHQUAKE EXCITATION. BUTLER,T.A.; BENNETT,J.G.; BABCOCK,C.D.; et al. Los Alamos Scientific Laboratory. February 1987. 69pp. 8703260013. LA-10871-MS. 40238:043.

The response of vertical, above-ground, fluid-filled tanks to seismic loads is reviewed and licensing criteria are recommended for use by the U.S. Nuclear Regulatory Commission in assessing the safety of seismic Category I tanks. Analysis methods and relevant experiments are first reviewed to provide a basis for recommending analytical techniques that are useful for tank safety evaluation. Next, field damage that has occurred during several earthquakes, starting with the 1964 Great Alaska Earthquake, are reviewed and the damage is categorized. This information is then used, along with experimental evidence, to assess the adequacy of current formal design codes. Finally, a procedure for Category I tank evaluation is recommended and topics that need additional research are identified.

NUREG/CR-4787: CONFERENCE OF RADIATION CONTROL DIRECTOR'S INFORMATION FOR LICENSING LOW-LEVEL RADIOACTIVE WASTE INCINERATORS AND COMPACTORS. * Conference of Radiation Control Program Directors, Inc. January 1987. 152pp. 8703030822. 39866:219.

This guidance was written to assist Agreement States and applicants addressing low-level waste processing as regulators or as licensees. The Low-Level Radioactive Waste Management Committee (E-5) of the Conference of Radiation Control Program Directors prepared this guidance document after evaluating current waste compaction and incineration practices with consideration of present applicable regulatory requirements for licensing. Incineration and compaction processes to reduce low-level radioactive waste volume have been licensed by Agreement States and by the U.S. Nuclear Regulatory Commission for over 20 years. Incineration volume reduction factors in the range from 10 to 100 have been achieved and compaction can reduce the waste volume by factors from 2 to 10 or more with accompanying reduction in the costs for waste disposal. In preparation of this guidance, the focus has been on keeping radiation exposure "as low as reasonably achievable." Compaction and incineration in particular to produce a more stable waste form with enhanced performance characteristics after disposal is a positive step toward that end. This document does not specifically address incinerator or compactor installations at nuclear power plants.

NUREG/CR-4788: AN ANALYTICAL AND EXPERIMENTAL INVESTIGATION OF NATURAL CIRCULATION TRANSIENTS IN A MODEL PRESSURIZED WATER REACTOR. MASSOUD,M. Maryland, Univ. of, College Park, MD. January 1987. 271pp. 8702170061. 39656:098.

The University of Maryland-College Park "2x4 Loop" scaled model facility was used to study natural circulation. This facility simulates a B&W lowered loop type PWR. The experimental investigation included determination of system characteristics as well as system response to imposed transients under symmetric and asymmetric conditions. Asymmetric transients were imposed to study flow oscillation and possible instability. The analytical investigation encompassed development of mathematical model for single phase, steady state, and transient natural circulation as well as modification of existing model for two-phase flow analysis of phenomena such as small break LOCA, high pressure coolant injection, and pump coast down. The modification included addition of models for once-through steam generator and electric heater rods. The development included coding of a computer program entitled "Symmetric and Asymmetric Analysis of Single-Phase Flow." Flow instability resulting in cessation of circulation was not observed. Primary system average temperature rose during a symmetric to asymmetric transient while the total secondary- side flow rate was maintained.

NUREG/CR-4789: THE SIMULATION OF THERMOHYDRAULIC PHENOMENA IN A PRESSURIZED WATER REACTOR PRIMARY LOOP. POPP,M. Maryland, Univ. of, College Park, MD. January 1987. 280pp. 8702170045. 39658:001.

Several fluid flow and heat transfer phenomena were investigated. Scaling and modeling laws for PWRs are reviewed and a new scaling approach focusing on the overall loop behavior is presented. Scaling criteria for one- and two-phase natural circulation are developed. Reactor vessel vent valve effects are included in the analysis. Two new dimensionless numbers, which uniquely describe one-phase flow in natural circulation loops, were deduced and are discussed. A scaled model of the primary loop of a typical Babock and Wilcox reactor was designed, built, and tested. The model operates at a maximum pressure of 300 psig and has a maximum heat input of 188 kW. It is about 4 times smaller in height than the real reactor, with a nominal volume scale of 1:500. Experiment measurements included primary side temperatures, hot leg velocities, and other primary and secondary loop performance data. All test data is compared to the theoretically derived performance predictions and scaling laws. The capability of the model to simulate reactor vent valve effects and small break loss of coolant accidents is discussed. Suggested changes to the model and its instrumentation are recommended. General system scaling with complete and simultaneous two-phase flow and heat transfer simulation in all components is not possible with a low pressure loop using water.

NUREG/CR-4793: RESULTS OF SEMISCALE MOD-2C SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITHOUT HPI (S-NH) EXPERIMENT SERIES. STREIT,J.E. EG&G Idaho, Inc. (subs. of EG&G, Inc.). January 1987. 61pp. 8704080273. EGG-2482. 40441:180.

Four experiments simulating small-break, loss-of-coolant accidents of 0.5% and 2.1% without high-pressure injection were performed in the Semiscale Mod-2C facility. These experiments differed in break size and recovery procedures. Three of the experiments had the same break size (0.5%), but recovery procedures were varied to determine what influence various procedures had on transient severity. The recovery procedures included secondary steam-and-feed initiation when a condition of inadequate core cooling was detected, secondary steam-and-feed initiation when the vessel level reached the top of the core (before a condition of inadequate core cooling was detected), and the restart of a primary coolant pump when a condition of inadequate core cooling was detected. The influence of the break size on the transient severity was also studied with the inclusion of the 2.1% break transient with secondary steam-and-feed initiation when condition of inadequate core cooling was detected. All four of the experiments were performed at high pressure and temperature [15.6 MPa (2262 psia) primary system pressure; 37.5 K (67.5 degrees F) core differential temperature; 587 K (597 degrees F) hot leg fluid temperature] and all four experiments had an initial bypass flow rate near 3%. Comparisons are made between the four experiments and conclusions drawn regarding what recovery procedure to use to prevent heater rod temperature excursions.

NUREG/CR-4797: PROGRESS REVIEWS OF SIX SAFETY PARAMETER DISPLAY SYSTEMS. LINER,R.T.; DEBOR,J. Science Applications International Corp. (formerly Science Applications, Inc.). March 1987. 78pp. 8704010129. SAIC-86/3066. 40340:241.

A pilot program of progress reviews of Safety Parameter Display Systems (SPDSs) was carried out through information-gathering visits to six plants in the period June-November 1985. The purpose was to sample industry progress toward the SPDS requirements stated in NUREG- 0737 Supplement 1 and thereby to determine the need for a post- implementation audit program. While three plants had, to varying degrees, demonstrated the viability of the SPDS concept through effective implementation, three of the six plants, some having been declared operational for as long as two years, had encountered major problems to the extent that their SPDSs could confuse or mislead operators in an emergency. The problems observed had not been apparent from prior reviews of Safety Analysis Reports on the systems. Major conclusions were that (1) a significant number of plants may be having problems with their SPDSs, and (2) assurance that any given SPDS meets the requirements cannot be determined with reasonable confidence without an on-site audit and discussions with personnel responsible for developing, operating, and maintaining the system. This report summarizes observations from the six plant visits and presents a plan, with procedures and guidance, for conducting post- implementation audits of additional SPDSs if they are undertaken.

NUREG/CR-4798: IRON OXIDE AEROSOL EXPERIMENTS IN STEAM-AIR ATMOSPHERES:NSPP TESTS 501-505 AND 511,DATA RECORD REPORT. ADAMS,R.E.; TOBIAS,M.L. Oak Ridge National Laboratory. January 1987. 89pp. 8704130219. ORNL/TM-10301. 40497:056.

This data record report summarizes the results from five tests involving Fe(2)O(3) test aerosol in a steam-air environment and one test in a dry air environment. This research sponsored by the U.S. Nuclear Regulatory Commission was conducted in the Nuclear Safety Pilot Plant at the Oak Ridge National Laboratory. The purpose of this project is to provide a data base on the behavior of aerosols in containment under conditions assumed to occur in postulated LWR accident sequences; this data base will provide experimental validation of aerosol behavioral codes under development. In the report a brief description is given of each test together with the results in the form of tables and graphs. Included are data on aerosol mass concentration, aerosol fallout and plateout rates, total mass fallout and plateout, aerosol particle size, vessel atmosphere pressure, vessel atmosphere temperatures, temperature gradients near the vessel wall, and steam condensation rates on the vessel wall.

NUREG/CR-4801: CLIMATOLOGY OF EXTREME WINDS IN SOUTHERN CALIFORNIA. RAMSDELL,J.V.; HUBBE,J.M.; ELLIOT,D.L.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1987. 104pp. 8702170052. PNL-6085. 39655:307.

A climatology of annual extreme winds in southern California has been prepared. The climatology includes a description of extreme wind regions, defined on the basis of observed winds and topography. Extreme wind distribution parameters have been estimated for 46 locations using data obtained from the National Climatic Data Center. Probabilities associated with extreme winds have been estimated for these locations. The results of the analysis are generally consistent with previous estimates of extreme winds in southern California. Although, in several instances the current estimates are significantly higher than previous estimates. The data examined do not indicate that there has been a significant change in the extreme wind climate of southern California.

NUREG/CR-4803: THE POSSIBILITY OF LOCAL DETONATIONS DURING DEGRADED-CORE ACCIDENTS IN THE BELLEFONTE NUCLEAR POWER PLANT. SHERMAN,M.P.; BERMAN,M. Sandia National Laboratories. January 1987. 45pp. 8704080052. SAND86-1180. 40446:240.

It is possible to objectively determine whether a detonation can propagate in a given geometry (volume, shape and size, obstacle configuration, degree of confinement) for a given mixture composition (concentrations of hydrogen, air and steam); this is done by conservatively equating the detonation propagation criteria with the criteria for transition from deflagration to detonation. This paper attempts to reduce the degree of conservatism in this procedure by constructing estimates of the probability of transition to detonation based on subjective extrapolations of empirical data. A methodology is introduced which qualitatively ranks mixtures and geometries according to the degree to which they are conducive to transition to detonation. The methodology is then applied to analyzing the potential for local detonations in the Bellefonte reactor containment for a variety of accident scenarios. Based on code-calculated rates and quantities of hydrogen generation and calculated rates for transport and mixing, this methodology indicated a low potential for detonation except for one volume in a few cases.

NUREG/CR-4813: ASSESSMENT OF LEAK DETECTION SYSTEMS FOR LWRs.October 1985 - September 1986. KUPPERMAN,D.S. Argonne National Laboratory. January 1987. 46pp. 8704130658. ANL-86-52. 40494:324.

It has become apparent that no currently available single lead- detection method for light-water reactors combines optimal leakage detection sensitivity, leak-locating ability, and the desired level of accuracy in leakage measurement. In this paper, NRC guidelines for leak detection will be reviewed, current practices described, potential safety-related problems discussed, and potential improvements in leak detection technology (with emphasis on acoustic methods) evaluated. Although information presented here is believed to be valid for most plants, additional data are needed to identify exceptions. For example, although quantitative leakage determination is possible with condensate flow monitors, sump monitors, and primary coolant inventory balance, these methods do not provide adequate location information, and are not necessarily sensitive enough to meet U.S. Nuclear Regulatory Commission Regulatory Guide 1.45 goals. Leak detection capability can be improved at specified sites by use of acoustic monitoring or moisture-sensitive tape (MST). However, current acoustic monitoring techniques provide no source discrimination (e.g., to distinguish between leaks from pipe cracks and valves) and no leak-rate information (a small leak may saturate the system). MST provides neither quantitative leak-rate information nor specific location information other than the location of the tape; moreover, its usefulness with "soft" insulation needs to be demonstrated.

NUREG/CR-4818: TRANSITION RANGE DROP TOWER J-R CURVE TESTING OF A106 STEEL. JOYCE,J.A. U.S. Naval Academy, Annapolis, MD. HACKETT,E.M. David W. Taylor Naval Research & Development Center. February 1987. 31pp. 8703250513. 40234:246.

Fracture toughness properties should be measured in the laboratory at loading rates and temperatures similar to those expected in the application of interest. This is not usually the case because of the experimental difficulties involved. This report describes a method being used to obtain J(Ic), J-R curves, and J at cleavage for three point bend tests conducted at drop tower rates through the ductile to brittle transition regime of the ferritic A106 steel being tested. The major conclusion is that these tests can now be accomplished, though a high degree of expertise and considerable practical experience is necessary to obtain good test results. The steel tested here is quite rate dependent as shown both by tensile tests and fracture toughness tests. A load elevation of 30 to 50% results in the drop tower 100 in/second tests on this material in comparison with static tests when both tests are conducted on the ductile upper shelf. Nonetheless, for this material J(Ic) and J-R curves are not elevated by the loading rate and this rather surprising result corresponds to a tendency for crack initiation to occur at a smaller bend angle for the high rate tests than for the static tests and a correspondingly greater amount of crack extension in the rapid specimen at a given bend angle beyond crack initiation than is present in the static test.

NUREG/CR-4820: COMPARISON OF THE 1982 SEADEX DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS. LEWELLEN,W.S.; SYKES,R.I.; CERASOLI,C.P.; et al. Aeronautical Research Associates of Princeton. February 1987. 163pp. 8703090089. ARAP 575. 39919:152.

The results from simulations by 12 dispersion models are compared with observations from an extensive field experiment conducted by the Nuclear Regulatory Commission in a shoreline environment during the early summer 1982. Ten of these models are the same as used in the earlier comparisons with the 1981 field tests at Idaho National Engineering Laboratory. All the models performed better on this SEADEX experiment. Little difference between the models is evident with hourly surface data, with the models able to predict the maximum value in the neighborhood of a sampler site within a factor of 2 approximately 25% of the time. This is raised to approximately 40% when the comparison is based on 12 hour integrated dose. For the longer time samples the more sophisticated models do show a distinct advantage when measured by correlation coefficient and root mean square error. If the data file is assumed incomplete, with higher data samples possible between the actual data samples, then the best results for the hourly samples show over 80% calculated within a factor of 2 when a 15 degree uncertainty in the plume position is permitted. This is raised to 90% for the 12 hour dose on the same comparison basis.

NUREG/CR-4824: EVALUATION OF INTEGRAL CONTINUING EXPERIMENTAL CAPABILITY (CEC) CONCEPTS FOR LIGHT WATER REACTOR RESEARCH - PWR SCALING CONCEPTS. CONDIE,K.G.; DAVIS,C.B.; LARSON,T.K.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). February 1987. 113pp. 8704130652. EGG-2494. 40494:046.

The United States Nuclear Regulatory Commission (USNRC) in assessing their future research needs for both separate effects and integral experiments has requested that EG&G Idaho,Inc., identify and technically evaluate potential concepts that will maintain the capability to conduct future integral, thermal-hydraulic facility experiments of interest to reactor safety. In this report reactor transients and thermal-hydraulic phenomena of importance (based on probabilistic risk assessments and the International Code Assessment Program) to reactor safety were examined and identified. Established scaling methodologies were used to develop potential concepts for integral thermal-hydraulic testing facilities. Advantages and disadvantages of each concept are evaluated. Analysis is conducted to examine the scaling of various phenomena in each of the selected concepts. Results generally suggest that a facility capable of operating at typical reactor operating conditions will scale most phenomena reasonably well. Although many phenomena in facilities using Freon or water at nontypical pressure will scale reasonably well, those phenomena that are heavily dependent on quality (heat transfer or critical flow for example) can be distorted. Further-more, relation of data produced in facilities operating with nontypical fluids or at nontypical pressure to large plants will be a difficult and time consuming process.

NUREG/CR-4825: A PRELIMINARY EVALUATION OF THE ECONOMIC RISK FOR CLEANUP OF NUCLEAR MATERIAL LICENSEE CONTAMINATION INCIDENTS. OSTMEYER,R.M.; SKINNER,D.J. Sandia National Laboratories. March 1987. 97pp. 8703260083. SAND86-2108. 40245:215.

This report documents an analysis of the economic risks from nuclear material licensee contamination incidents. The results of the analyses are intended to provide a technical basis for an NRC rulemaking that would require nuclear material licensees to demonstrate adequate financial means to cover the cleanup costs for accidental or inadvertent release of radioactive materials. The important products of this effort include (1) a method of categorizing licensees according to the potential cost and frequency of contamination incidents, (2) a model for ranking the categories of licensees according to potential incident costs, and (3) estimates of contamination risk for the licensee categories.

NUREG/CR-4826 V01: SEISMIC MARGIN REVIEW OF THE MAINE YANKEE ATOMIC POWER STATION.Volume 1.Summary Report. PRASSINOS,P.G.; MURRAY,R.C.; CUMMINGS,G.E. Lawrence Livermore National Laboratory. March 1987. 200pp. 8704090044. UCID-20948. 40464:128.

This Summary Report is the first of three volumes for the "Seismic Margin Review of the Maine Yankee Atomic Power Station." Volume 2 is the Systems Analysis of the first trial seismic margin review. Volume 3 documents the results of the fragility screening for the review. The three volumes demonstrate how the seismic margin review guidance (NUREG/CR-4482) of the Nuclear Regulatory Commission (NRC) Seismic Design Margins Program can be applied. The overall objectives of the trial review are to assess the seismic margins of a particular pressurized water reactor, and to test the adequacy of this review approach, quantification techniques, and guidelines for performing the review. Results from the trial review will be used to revise the seismic margin methodology and guidelines so that the NRC and industry can readily apply them to assess the inherent quantitative seismic capacity on nuclear power plants.

NUREG/CR-4826 V02: SEISMIC MARGIN REVIEW OF THE MAINE YANKEE ATOMIC POWER STATION.Volume 2.Systems Analysis. MOORE,D.L.; JONES,D.M.; QUILICI,M.D.; et al. Energy, Inc. March 1987. 201pp. 8704090075. UCID-20948. 40466:232.

This System Analysis is the second of three volumes for the "Seismic Margin Review of the Maine Yankee Atomic Power Station." Volume 1 is the Summary Report of the first trial seismic margin review. Volume 3, Fragility Analysis, documents the results of the fragility screening for the review. The three volumes demonstrate how the seismic margins review guidance (NUREG/CR-4482) of the NRC Seismic Design Margins Program can be applied. The overall objectives of the trial review are to assess the seismic margins of a particular pressurized water reactor, and to test the adequacy of this review approach, quantification techniques, and guidelines for performing the review. Results from the trial review will be used to revise the seismic margin methodology and guidelines so that the NRC and industry can readily apply them to assess the inherent quantitative seismic capacity of nuclear power plants.

NUREG/CR-4826 V03: SEISMIC MARGIN REVIEW OF THE MAINE YANKEE ATOMIC POWER STATION.Volume 3.Fragility Analysis. RAVINDRA,M.K.; HARDY,G.S.; HASHIMOTO,P.S.; et al. EQE, Inc. March 1987. 217pp. 8704090063. UCID-20948. 40461:139.

This Fragility Analysis is the third of three volumes for the "Seismic Margin Review of the Maine Yankee Atomic Power Station." Volume 1 is the Summary Report of the first trial seismic margin review. Volume 2, Systems Analysis, documents the results of the systems screening for the review. The three volumes demonstrate how the seismic margins review guidance (NUREG/CR-4482) of the NRC Seismic Design Margins Program can be applied. The overall objectives of the trial review are to assess the seismic margins of a particular pressurized water reactor, and to test the adequacy of this review approach, quantification techniques, and guidelines for performing the review. Results from the trial review will be used to revise the seismic margin methodology and guidelines so that the NRC and industry can readily apply them to assess the inherent quantitative seismic capacity of nuclear power plants.

NUREG/CR-4829 V01: SHIPPING CONTAINER RESPONSE TO SEVERE HIGHWAY AND RAILWAY ACCIDENT CONDITIONS.Main Report. FISCHER,L.E.; CHOU,C.K.; GERHARD,M.A.; et al. Lawrence Livermore National Laboratory. February 1987. 284pp. 8703120326. UCID-20733. 39982:032.

This report describes a study performed by the Lawrence Livermore National Laboratory to evaluate the level of safety provided under severe accident conditions during the shipment of spent fuel from nuclear power reactors. The evaluation is performed using data from real accident histories and using representative truck and rail cask models that likely meet 10 CFR 71 regulations. The responses of the representative casks are calculated for structural and thermal loads generated by severe highway and railway accident conditions. The cask responses are compared with those responses calculated for the 10 CFR 71 hypothetical accident conditions. By comparing the responses it is determined that most highway and railway accident conditions fall within the 10 CFR 71 hypothetical accident conditions. For those accidents that have higher responses, the probabilities and potential radiation exposures of the accidents are compared with those identified by the assessments made in the "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes," NUREG-0170. Based on this comparison, it is concluded that the radiological risks from spent fuel under severe highway and railway accident conditions as derived in this study are less than risks previously estimated in the NUREG-0170 document.

NUREG/CR-4829 V02: SHIPPING CONTAINER RESPONSE TO SEVERE HIGHWAY AND RAILWAY ACCIDENT CONDITIONS.Appendices. FISCHER,L.E.; CHOU,C.K.; GERHARD,M.A.; et al. Lawrence Livermore National Laboratory. February 1987. 300pp. 8703120299. UCID-20733. 39982:316.

See NUREG/CR-4829,V01 abstract.

NUREG/CR-4843 V01: UNIVERSITY OF MARYLAND AT COLLEGE PARK (UMCP) 2X4 LOOP TEST FACILITY.Annual Report For 1985. DIMARZO,M.; HSU,Y.Y.; LIN,W.K.; et al. Maryland, Univ. of, College Park, MD. March 1987. 275pp. 8704090054. 40465:020.

The efforts for the year 1985 of the investigators of the University of Maryland on the UMCP 2x4 Loop facility are presented. These efforts include: additional work on the facility, theoretical investigations, and experimental investigations. The 2x4 Loop facility is a low pressure scaled representation of a lowered loop B&W reactor and once through steam generator. The report is prepared in three chapters and seven appendices. A brief description of the facility including the final design details are presented in chapter one. Chapter two includes the theoretical basis for the experimental investigations. Chapter three contains the details of experiments, test results, and final conclusions. The appendices contain additional details about the topics discussed in the chapters.

NUREG/CR-4847: CASE HISTORIES OF WEST VALLEY SPENT FUEL SHIPMENTS.Final Report. * Aerospace Corp. January 1987. 239pp. 8702270048. WPR-86(6811)-1. 39760:064.

In 1983, NRC/FC initiated a study on institutional issues related to spent fuel shipments originating at the former spent fuel reprocessing facility in West Valley, New York. FC staff viewed the shipment campaigns as a one-time opportunity to document the institutional issues that may arise with a substantial increase in spent fuel shipping activity. NRC subsequently contracted with the Aerospace Corporation for the West Valley Study. This report contains a detailed description of the events which took place prior to and during the spent fuel shipments. The report also contains a discussion of the shipment issues that arose, and presents general findings. Most of the institutional issues discussed in the report do not fall under NRC's transportation authority. The case histories provide a reference to agencies and other institutions thay may be involved in future spent fuel shipping campaigns.

NUREG/CR-4852: THE MEERS FAULT:TECTONIC ACTIVITY IN SOUTHWESTERN OKLAHOMA. RAMELLI,A.R.; SLEMMONS,D.B.; BROCOUM,S.J. Nevada, Univ. of, Reno, NV. March 1987. 59pp. 8704090047. 40463:318.

The Meers Fault in Southwestern Oklahoma is capable of producing large, damaging earthquakes. By comparison to historical events, a minimum of M = 6 3/4 to 7 1/4 could be expected. The most recent surface rupturing event occurred in the late Holocene, and it appears that one or more pre-Holocene events preceded it. Surface rupture length is at least 37 km. Displacements comprising the present-day scarp have left-lateral and high-angle reverse components. Vertical separation of the ground surface reaches 5 m, while lateral separation exceeds the vertical by a ratio of about 3:1 to 5:1, reaching about 20 m. Individual events apparently had maximum displacements of several meters. The Meers Fault may be part of a larger active zone. Based on surface expressions, the Washita Valley, Oklahoma and Potter County, Texas Faults may also have ruptures during the late Quaternary, although not as recently as the Meers Fault. Low sun angle photography in Southwestern Oklahoma revealed no evidence of fault activity, other than that of the Meers Fault, although activity may be concealed by poor preservation or ductile surface deformation. This suggests that additional areas of activity may be sparse and rupture infrequently.

NUREG/CR-4853: APPROXIMATE METHODS FOR FRACTURE ANALYSES OF THROUGH-WALL CRACKED PIPES. BRUST,F.W. Battelle Memorial Institute, Columbus Laboratories. February 1987. 122pp. 8703250491. BMI-2145. 40234:124.

Current leak-before-break analyses involve assessing the load- carrying capacity of through-wall cracked pipe. Five prediction techniques were evaluated in this report. The technical basis for two analysis methods developed in the Degraded Piping Program, LBB.BCL1 and LBB.BCL2, are presented in this report. Other methods evaluated are the GE/EPRI, NUREG/CR-3464 and LBB.NRC analyses. These methods are all based on the J-integral/tearing modulus theory. As such, they all fall under the category of J-estimation schemes. These J- estimation schemes are all relatively simple to be compared to finite element analysis. Predicting the fracture performance can be achieved very quickly, and is done through the use of an IBM PC computer code called NRCPIPE. The assessments of the five methods involved comparing the experimental data to the predicted load versus load- point displacement curves. This was done for both carbon steel and stainless steel pipes with cracks in the base metal, as well as stainless steel pipes with cracks in TIG or submerged arc welds. In addition, both deformation and modified J-R curves were used in the assessments. Finally, a sensitivity study was made to show that the reference stress used in the Ramberg-Osgood relation could be based on either the yield strength or the flow stress if the coefficent is properly adjusted.

NUREG/CR-4856: FEASIBILITY STUDY ON A DATA-BASED SYSTEM FOR DECISIONS REGARDING OCCUPATIONAL RADIATION PROTECTION MEASURES. WATSON,E.C.; FISHER,D.R. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1987. 71pp. 8703250618. PNL-6137. 40235:087.

In a study commissioned by the U.S. Nuclear Regulatory Commission, Pacific Northwest Laboratory conducted a study to determine the feasibility of developing data-based protective measure decision levels for assuring uniformity of licensing requirements for radiation protection of workers. Eleven facilities licensed to work with unencapsulated radioactive material were visited to collect data for this purpose. Sufficient data were obtained from six facilities to estimate release fractions (i.e., fraction of material in process released to the workplace environment) for tritium tube filling operations (about 10(-5)), tritium compound preparation (about 10(-7)), I-125 radiopharmaceutical preparations (about 10(-7)), Am-241 source production (about 10(-11)), and forge filing and incineration of natural uranium (about 10(-7)). The fraction of material in process that may be taken into the bodies of workers involved in these processes was estimated to be about 4x10(-8), 4x10(-10), 2x10(-9), 6x10(-14), and (1-5)x10(-7), respectively. Five levels of radiation protection programs were developed. The most feasible approach to assuring uniformity of licensing requirements for these programs is a technical one based on mathematical relationships using decision levels developed by a panel of experts. This approach, however, requires a consensus among regulators and licensees on fundamental values used to determine the need for specific protective measures.

NUREG/CR-4859: SEISMIC FRAGILITY TEST OF A 6-INCH DIAMETER PIPE SYSTEM. CHEN,W.P.; ONESTO,A.T.; DEVITA,V. Energy Technology Engineering Center. February 1987. 174pp. 8703260100. 40237:229.

This report contains the test results and assessments of seismic fragility tests performed on a 6-inch diameter piping system. The test was funded by the U.S. Nuclear Regulatory Commission (NRC) and conducted by ETEC. The objective of the test was to investigate the ability of a representative nuclear piping system to withstand high level dynamic seismic and other loadings. Levels of loadings achieved during seismic testing were 20 to 30 times larger than normal elastic design evaluations to ASME Level D limits would permit. Based on failure data obtained during seismic and other dynamic testing, it was concluded that nuclear piping systems are inherently able to withstand much larger dynamic seismic loadings than permitted by current design practice criteria or predicted by the probabilistic risk assessment (PRA) methods and several proposed nonlinear methods of failure analysis.

NUREG/CR-4861: DEVELOPMENT OF SITE SPECIFIC RESPONSE SPECTRA. BERNREUTER,D.J.; CHEN,J.C.; SAVY,J.B. Lawrence Livermore National Laboratory. March 1987. 141pp. 8704010140. UCID-20980. 40321:189.

For a number of years the USNRC has employed site specific spectra (SSSP) in their evaluation of the adequacy of the Safe Shutdown Earthquake (SSE). As the data set has considerably increased for Eastern North America (ENA) and as more relevant data has become available from earthquakes occurring in other parts of the world (e.g., Italy), together with the fact that recent data indicated the importance of the vertical component, it became clear that an update of the SSSP's for ENA was desirable. This study used actual earthquake ground motion data with magnitudes within a certain range and recorded at distances and at sites similar to those that would be chosen for the definition of an SSE. An extension analysis of the origin and size of the uncertainty is an important part of this study. The results of this analysis of the uncertainties is used to develop criteria for selecting the earthquake records to be used in the derivation of the SSSP's. We concluded that the SSSPs were not very sensitive to the distribution of the source to site distance of the earthquake records used in the analysis. That is, the variability (uncertainty) introduced by the range of distances was relatively small compared to the variability introduced by other factors. We also concluded that the SSSP are somewhat sensitive to the distribution of the magnitudes of these earthquakes, particularly at rock sites and, by inference, at shallow soil sites. We found that one important criterion in selecting records to generate SSSP is the depth of soil at the site.

NUREG/CR-4868: METALLURGICAL EVALUATION OF AN 18-INCH FEEDWATER LINE FAILURE AT THE SURRY UNIT 2 POWER STATION. CZAJKOWSKI,C.J. Brookhaven National Laboratory. March 1987. 43pp. 8704090019. BNL-NUREG-52057. 40464:016.

A metallurgical failure analysis was performed on pieces from a catastrophically failed 18-inch diameter feedwater line from the Surry Unit 2 Nuclear Power Station. The failed pipe had been globally thinned and had a scalloped appearance on the inside surface. All fracture surfaces examined showed a ductile failure mode. The materials of construction met the appropriate specification requirements (both mechanical and chemical). The report has as its final conclusion that the pipe failed due to excessive thinning by an erosion-corrosion mechanism.

NUREG/IA-0004: THERMAL MIXING TESTS IN A SEMIANNULAR DOWNCOMER WITH INTERACTING FLOWS FROM COLD LEGS. TUOMISTO,H.; MUSTONEN,P. Finland, Govt. of. October 1986. 343pp. 8703090081. 39907:230.

This report describes the test facility and test program for studying thermal mixing of high-pressure injection (HPI) water in the two-fifths scale model of three cold legs, semiannular downcomer and lower plenum of a pressurized water reactor. This test series has been carried out by mutual agreement on the pressurized thermal shock (PTS) information exchange between the U.S. Nuclear Regulatory Commission and Imatran Voima Oy. The test facility was originally designed to model the Finish Loviisa plant but it was redesigned and modified for this test program. The facility can be operated at atmospheric pressure with loop and HPI flows from different cold legs in the area of interest to PTS. Transparent materials were used to allow flow visualization during the tests. The choice of transparent materials limit the upper temperature to 75 degrees C. The full buoyancy effect was induced by salt addition and the HPI temperature was used as a tracer. The test matrix consists of 20 tests. The varied parameters were flow rates and the number and configuration of cold legs with HPI and loop flows. Four tests were done with decreasing loop flow temperature to simulate primary flows during steam line breaks.

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-1211: REGULATORY ANALYSIS FOR RESOLUTION OF UNRESOLVED SAFETY ISSUE A-46,SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS.

Unsaturated Zone
NUREG/CR-4737: INTERPRETATIVE ANALYSIS OF DATA FOR SOLUTE TRANSPORT IN THE UNSATURATED ZONE.

Volume Reduction
NUREG/CR-4787: CONFERENCE OF RADIATION CONTROL DIRECTOR'S INFORMATION FOR LICENSING LOW-LEVEL RADIOACTIVE WASTE INCINERATORS AND COMPACTORS.

Warm Prestressing
NUREG/CR-4491: DEVELOPMENT OF MODELS FOR WARM PRESTRESSING.

Waste Disposal
NUREG/CR-3444 V04: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL,AND ASSOCIATED OCCUPATIONAL EXPOSURE.Annual Report, Fiscal Year 1986.

Waste Disposal Site
NUREG/CR-2478 V03: A STUDY OF TRENCH COVERS TO MINIMIZE INFILTRATION AT WASTE DISPOSAL SITES.Final Rept.

Waste Package Test Data
NUREG/CR-4735 V01: EVALUATION AND COMPILATION OF DOE WASTE PACKAGE TEST DATA. Biannual Report:December 1985 - July 1986.

Water Leakage
NUREG/CR-4524: CLOSEOUT OF IE BULLETIN 80-24:PREVENTION OF DAMAGE DUE TO WATER LEAKAGE INSIDE CONTAINMENT (OCTOBER 17,1980 INDIAN POINT 2 EVENT).

Water Quality
NUREG-1243: GROUND-WATER PROTECTION ACTIVITIES OF THE U.S. NUCLEAR REGULATORY COMMISSION.

West Valley
NUREG/CR-4847: CASE HISTORIES OF WEST VALLEY SPENT FUEL SHIPMENTS.Final Report.

Westinghouse
NUREG/CR-4672: ANALYSIS OF INSTRUMENT TUBE RUPTURES IN WESTINGHOUSE 4-LOOP PRESSURIZED WATER REACTORS.

Westinghouse 2-Loop
NUREG/CR-4458: SHUTDOWN DECAY HEAT REMOVAL ANALYSIS OF A WESTINGHOUSE 2-LOOP PRESSURIZED WATER REACTOR.Case Study.

Westinghouse 3-Loop
NUREG/CR-4762: SHUTDOWN DECAY HEAT REMOVAL ANALYSIS OF A WESTINGHOUSE 3-LOOP PRESSURIZED WATER REACTOR.Case Study.

Workshop
NUREG/CP-0054: PROCEEDINGS OF THE WORKSHOP ON SOIL-STRUCTURE INTERACTION.
NUREG/CP-0084: PROCEEDINGS OF THE WORKSHOP ON A CONTAINMENT PERFORMANCE DESIGN OBJECTIVE,MAY 12-13,1986,HARPERS FERRY,WEST VIRGINIA.

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NRC Originating Organization Index (Staff Reports)

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

OFFICE OF EXECUTIVE DIRECTOR FOR OPERATIONS (EDO)
REGION 1, OFFICE OF DIRECTOR
NUREG-0837 V06 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, July-September 1986.

EDO - OFFICE OF ADMINISTRATION
OFFICE OF ADMINISTRATION (PRE 870413)
NUREG-0304 V11 N04: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1986.
NUREG-0325 R10: U.S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS.
NUREG-0540 V08 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30,1986.
NUREG-0540 V08 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. December 1-31,1986.
NUREG-0540 V09 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31,1987.
NUREG-0750 V24 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1986.Pages 1-195.
NUREG-0750 V24 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1986.Pages 197-396.
NUREG-0750 V24 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1986.Pages 397-488.
DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL (PRE 870120)
NUREG-0540 V08 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. October 1-31,1986.
DIVISION OF RULES & RECORDS (PRE 870413)
NUREG-0936 V05 N03: NRC REGULATORY AGENDA.Quarterly Report,July-September 1986.

EDO - OFFICE OF STATE PROGRAMS
ASSISTANT DIRECTOR FOR STATE AGREEMENTS PROGRAMS
NUREG/CP-0085: MEETING WITH STATES ON THE LOW-LEVEL RADIOACTIVE WASTE POLICY AMENDMENTS ACT (LLRWPAA) OF 1985.

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA
OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA, DIRECTOR
NUREG-0090 V09 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1986.

OFFICE OF INSPECTION & ENFORCEMENT (POST 12/11/80)
OFFICE OF INSPECTION & ENFORCEMENT, DIRECTOR (820201-870411)
NUREG-0430 V07 N01: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.January-June 1986.(Gray Book II)
NUREG-0940 V05 N04: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,October-December 1986.
DIVISION OF EMERGENCY PREPAREDNESS & ENGINEERING RESPONSE (850212-87041
NUREG-1210 V01: PILOT PROGRAM:NRC SEVERE REACTOR ACCIDENT INCIDENT RESPONSE TRAINING MANUAL.Overview And Summary Of Major Points.
NUREG-1210 V02: PILOT PROGRAM:NRC SEVERE REACTOR ACCIDENT INCIDENT RESPONSE TRAINING MANUAL.Severe Reactor Accident Overview.
NUREG-1210 V03: PILOT PROGRAM:NRC SEVERE REACTOR ACCIDENT INCIDENT RESPONSE TRAINING MANUAL.Response Of Licensee And State And Local Officials.
NUREG-1210 V04: PILOT PROGRAM:NRC SEVERE REACTOR ACCIDENT INCIDENT RESPONSE TRAINING MANUAL.Public Protective Actions - Predetermined Criteria And Initial Actions.
NUREG-1210 V05: PILOT PROGRAM:NRC SEVERE REACTOR ACCIDENT INCIDENT RESPONSE TRAINING MANUAL.U.S. Nuclear Regulatory Commission Response.
DIVISION OF QA, VENDOR & TECHNICAL TRAINING CENTER PROGRAMS (850212-870
NUREG-0040 V10 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1986.(White Book)

OFFICE OF INFORMATION RESOURCES MANAGEMENT
DIVISION OF COMPUTER & TELECOMMUNICATIONS SERVICES (PRE 870413)
NUREG-0020 V10 N09: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of August 31,1986.(Gray Book I)
NUREG-0020 V10 N10: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of September 30,1986.(Gray Book I)

OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
DIVISION OF SAFEGUARDS (PRE 870413)
NUREG-0525 R12: SAFEGUARDS SUMMARY EVENT LIST (SSEL).
NUREG-1280: STANDARD FORMAT AND CONTENT ACCEPTANCE CRITERIA FOR THE MATERIAL CONTROL AND ACCOUNTING (MC&A) REFORM AMENDMENT. 10 CFR Part 74 Subpart E.
DIVISION OF WASTE MANAGEMENT (PRE 870413)
NUREG-1101 V02: ONSITE DISPOSAL OF RADIOACTIVE WASTE.Methodology For The Radiological Assessment Of Disposal By Subsurface Burial.
NUREG-1199: STANDARD FORMAT AND CONTENT OF A LICENSE APPLICATION FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY.
NUREG-1200: STANDARD REVIEW PLAN FOR THE REVIEW OF A LICENSE APPLICATION FOR A LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY.
NUREG-1243: GROUND-WATER PROTECTION ACTIVITIES OF THE U.S. NUCLEAR REGULATORY COMMISSION.

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF THE GENERAL COUNSEL
NUREG-0386 D04 R04: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.July 1972 - June 1986.
NRC - NO DETAILED AFFILIATION GIVEN
NUREG-1250: REPORT ON THE ACCIDENT AT THE CHERNOBYL NUCLEAR POWER STATION.
NUREG/CR-3620 S02: INTRUDER DOSE PATHWAY ANALYSIS FOR THE ONSITE DISPOSAL OF RADIOACTIVE WASTES.The ONSITE/MAXI1 Computer Program.
NUREG/CR-3950 V03: FUEL PERFORMANCE ANNUAL REPORT FOR 1985.

OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 4/05/81)
OFFICE OF NUCLEAR REGULATORY RESEARCH, DIRECTOR (POST 860720)
NUREG-1150 DRF V1 FC: REACTOR RISK REFERENCE DOCUMENT.Main Report.Draft For Comment.
NUREG-1150 DRF V2 FC: REACTOR RISK REFERENCE DOCUMENT.Appendices A-I.Draft For Comment.
NUREG-1150 DRF V3 FC: REACTOR RISK REFERENCE DOCUMENT.Appendices J-O.Draft For Comment.
NUREG-1260 V01: A REPORT TO CONGRESS ON NUCLEAR REGULATORY RESEARCH.Project Descriptions For FY87.
NUREG/CP-0082 V01: PROCEEDINGS OF THE FOURTEENTH WATER REACTOR SAFETY INFORMATION MEETING.
NUREG/CP-0082 V02: PROCEEDINGS OF THE FOURTEENTH WATER REACTOR SAFETY INFORMATION MEETING.
NUREG/CP-0082 V03: PROCEEDINGS OF THE FOURTEENTH WATER REACTOR SAFETY INFORMATION MEETING.
NUREG/CP-0082 V04: PROCEEDINGS OF THE FOURTEENTH WATER REACTOR SAFETY INFORMATION MEETING.
NUREG/CP-0082 V05: PROCEEDINGS OF THE FOURTEENTH WATER REACTOR SAFETY INFORMATION MEETING.

NUREG/CP-0082 V06: PROCEEDINGS OF THE FOURTEENTH WATER REACTOR SAFETY INFORMATION MEETING.

DIVISION OF ENGINEERING SAFETY (860720-870413)

NUREG-0975 V05: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS BRANCH, DIVISION OF ENGINEERING SAFETY.Annual Rept For FY 1986.

DIVISION OF REACTOR SYSTEM SAFETY (860720-870413)

NUREG-1163: COORDINATION OF SAFETY RESEARCH FOR THE BABCOCK AND WILCOX INTEGRAL SYSTEM TEST PROGRAM.

EDO-RESOURCE MANAGEMENT

DIVISION OF BUDGET & ANALYSIS (PRE 870413)

NUREG-1100 V03: BUDGET ESTIMATES.Fiscal Years 1988-1989.

OFFICE OF NUCLEAR REACTOR REGULATION (POST 4/28/80)

DIVISION OF PRESSURIZED WATER REACTOR LICENSING - A (851125-870411)

NUREG-0781 S02: 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)

NUREG-0876 S08: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BYRON STATION,UNITS 1 AND 2.Docket Nos. 50-454 And 50-455. (Commonwealth Edison Company)

NUREG-1057 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BEAVER VALLEY POWER STATION,UNIT 2.Docket No. 50-412.(Duquesne Light Company,et al)

NUREG-1137 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF VOGTLE ELECTRIC GENERATING PLANT,UNITS 1 AND 2.Docket Nos. 50-424 And 50-425.(Georgia Power Company,et al)

NUREG-1137 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF VOGTLE ELECTRIC GENERATING PLANT,UNITS 1 AND 2.Docket Nos. 50-424 And 50-425.(Georgia Power Company,et al)

NUREG-1237: TECHNICAL SPECIFICATIONS FOR VOGTLE ELECTRIC GENERATING PLANT,UNIT 1.Docket No. 50-424.(Georgia Power Company)

NUREG-1240: TECHNICAL SPECIFICATIONS FOR SHEARON HARRIS NUCLEAR POWER PLANT UNIT 1.Docket No. 50-400.(Carolina Power & Light Company)

NUREG-1247: TECHNICAL SPECIFICATIONS FOR VOGTLE ELECTRIC GENERATING PLANT,UNIT 1.Docket No. 50-424.(Georgia Power Company)

DIVISION OF PRESSURIZED WATER REACTOR LICENSING - B (851125-870411)

NUREG-0857 S11: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PALO VERDE NUCLEAR GENERATING STATION,UNITS 1,2 AND 3.Docket Nos. 50-528,50-529 And 50-530.(Arizona Public Service Company,et al)

NUREG-1224: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE UNIVERSITY OF NEW MEXICO RESEARCH REACTOR.Docket No. 50-252. (University Of New Mexico)

NUREG-1248: TECHNICAL SPECIFICATIONS FOR PALO VERDE NUCLEAR GENERATING STATION,UNIT 3.Docket No. 50-530.(Arizona Public Service Company)

DIVISION OF BOILING WATER REACTOR (BWR) LICENSING (851125-870411)

NUREG-0853 S08: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF CLINTON POWER STATION,UNIT 1.Docket No. 50-461.(Illinois Power Company,et al)

DIVISION OF SAFETY REVIEW & OVERSIGHT (851125-870411)

NUREG-0933 S06: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

NUREG-1030: SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING NUCLEAR POWER PLANTS.Unresolved Safety Issue A-46.

NUREG-1211: REGULATORY ANALYSIS FOR RESOLUTION OF UNRESOLVED SAFETY ISSUE A-46,SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS.

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NRC Originating Organization Index (International Agreements)

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

OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 4/05/81)
OFFICE OF NUCLEAR REGULATORY RESEARCH, DIRECTOR (POST 860720)
NUREG/IA-0004: THERMAL MIXING TESTS IN A SEMIANNULAR DOWNCOMER WITH INTERACTING FLOWS FROM COLD LEGS.

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)
REGION 2, OFFICE OF DIRECTOR
NUREG/CR-4868: METALLURGICAL EVALUATION OF AN 18-INCH FEEDWATER LINE FAILURE AT THE SURRY UNIT 2 POWER STATION.

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA
OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA, DIRECTOR
NUREG/CR-2000 V05N12: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of December 1986.
NUREG/CR-2000 V06 N1: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of January 1987.

OFFICE OF INSPECTION & ENFORCEMENT (POST 12/11/80)
DIVISION OF EMERGENCY PREPAREDNESS & ENGINEERING RESPONSE (850212-87041
NUREG/CR-4524: CLOSEOUT OF IE BULLETIN 80-24:PREVENTION OF DAMAGE DUE TO WATER LEAKAGE INSIDE CONTAINMENT (OCTOBER 17,1980 INDIAN POINT 2 EVENT).

OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
DIVISION OF FUEL CYCLE & MATERIAL SAFETY (PRE 870413)
NUREG/CR-4736: COMBUSTION AEROSOLS FORMED DURING BURNING OF RADIOACTIVELY CONTAMINATED MATERIALS - EXPERIMENTAL RESULTS.
NUREG/CR-4847: CASE HISTORIES OF WEST VALLEY SPENT FUEL SHIPMENTS.Final Report.
DIVISION OF SAFEGUARDS (PRE 870413)
NUREG/CR-4695: MATERIAL CONTROL AND ACCOUNTING (MC&A) LOSS DETECTION DURING TRANSITION PERIODS AND PROCESS UPSET CONDITIONS.
DIVISION OF WASTE MANAGEMENT (PRE 870413)
NUREG/CR-2478 V03: A STUDY OF TRENCH COVERS TO MINIMIZE INFILTRATION AT WASTE DISPOSAL SITES.Final Rept.
NUREG/CR-3620 S02: INTRUDER DOSE PATHWAY ANALYSIS FOR THE ONSITE DISPOSAL OF RADIOACTIVE WASTES.The ONSITE/MAXI1 Computer Program.
NUREG/CR-4708 V01 N1: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Semiannual Report For October 1985 - March 1986.
NUREG/CR-4730: EVALUATION OF POTENTIAL MIXED WASTES CONTAINING LEAD, CHROMIUM,USED OIL,OR ORGANIC LIQUIDS.
NUREG/CR-4735 V01: EVALUATION AND COMPILATION OF DOE WASTE PACKAGE TEST DATA. Biannual Report:December 1985 - July 1986.
NUREG/CR-4737: INTERPRETATIVE ANALYSIS OF DATA FOR SOLUTE TRANSPORT IN THE UNSATURATED ZONE.
NUREG/CR-4825: A PRELIMINARY EVALUATION OF THE ECONOMIC RISK FOR CLEANUP OF NUCLEAR MATERIAL LICENSEE CONTAMINATION INCIDENTS.

OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 4/05/81)
OFFICE OF NUCLEAR REGULATORY RESEARCH, DIRECTOR (POST 860720)
NUREG/CR-2331 V06 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,April-June 1986.
NUREG/CR-3469 V03: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS. Annotated Bibliography Of Selected Readings In Radiation Protection And ALARA.
NUREG/CR-4301: STATUS REPORT ON EQUIPMENT QUALIFICATION ISSUES RESEARCH AND RESOLUTION.
NUREG/CR-4320: THE RELATIONSHIP AND INFLUENCES OF FUEL AND COOLANT SYSTEM PROCESSES DURING LWR SEVERE ACCIDENTS.
NUREG/CR-4531: AN INVESTIGATION OF INTEGRAL FACILITY SCALING AND DATA RELATION METHODS (INTEGRAL SYSTEM TEST PROGRAM).
NUREG/CR-4741: FEEDWATER TRANSIENT AND SMALL BREAK LOSS OF COOLANT ACCIDENT ANALYSES FOR THE BELLEFONTE NUCLEAR PLANT.
NUREG/CR-4803: THE POSSIBILITY OF LOCAL DETONATIONS DURING DEGRADED-CORE ACCIDENTS IN THE BELLEFONTE NUCLEAR POWER PLANT.
DIVISION OF ENGINEERING SAFETY (860720-870413)
NUREG/CR-3232: DETAILED STUDIES OF SELECTED,WELL EXPOSED FRACTURE ZONES IN THE ADIRONDACK MOUNTAINS DOME,NEW YORK.
NUREG/CR-3412 V02: CONTAINMENT INTEGRITY PROGRAM.Progress Report,April 1983 -December 1984.
NUREG/CR-3444 V04: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION,WASTE DISPOSAL,AND ASSOCIATED OCCUPATIONAL EXPOSURE.Annual Report, Fiscal Year 1986.
NUREG/CR-3861: STRESS-CORROSION CRACKING OF LOW-STRENGTH CARBON STEELS IN CANDIDATE HIGH-LEVEL WASTE REPOSITORY ENVIRONMENTS.
NUREG/CR-4300 V03 N2: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR INSERVICE MONITORING OF NUCLEAR PRESSURE VESSELS.Progress Rept,April-September 1986.
NUREG/CR-4469 V04: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report,October 1985 - March 1986.
NUREG/CR-4491: DEVELOPMENT OF MODELS FOR WARM PRESTRESSING.
NUREG/CR-4541: EXPERIMENTAL ASSESSMENT OF THE SEALING EFFECTIVENESS OF ROCK FRACTURE GROUTING.
NUREG/CR-4685: POST-PLIOCENE DISPLACEMENT ON FAULTS WITHIN THE KENTUCKY RIVER FAULT SYSTEM OF EAST-CENTRAL KENTUCKY.
NUREG/CR-4711: LOW UPPER-SHELF TOUGHNESS,HIGH-TRANSITION TEMPERATURE TEST INSERT IN HSST PTSE-2 VESSEL AND WIDE-PLATE TEST SPECIMENS. Final Report.
NUREG/CR-4712: REGULATORY ANALYSIS OF REGULATORY GUIDE 1.35 (REVISION 3, DRAFT 2) - IN-SERVICE INSPECTION OF UNGROUTED TENDONS IN PRESTRESSED CONCRETE CONTAINMENTS.
NUREG/CR-4724: FATIGUE CRACK GROWTH RATES IN PRESSURE VESSEL AND PIPING STEELS IN LWR ENVIRONMENTS.Final Report.
NUREG/CR-4734: SEISMIC TESTING OF TYPICAL CONTAINMENT PIPING PENETRATION SYSTEMS.
NUREG/CR-4744 V01 N1: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,October 1985 - March 1986.
NUREG/CR-4787: CONFERENCE OF RADIATION CONTROL DIRECTOR'S INFORMATION FOR LICENSING LOW-LEVEL RADIOACTIVE WASTE INCINERATORS AND COMPACTORS.
NUREG/CR-4813: ASSESSMENT OF LEAK DETECTION SYSTEMS FOR LWRs.October 1985 - September 1986.
NUREG/CR-4818: TRANSITION RANGE DROP TOWER J-R CURVE TESTING OF A106 STEEL.
NUREG/CR-4820: COMPARISON OF THE 1982 SEADEX DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS.
NUREG/CR-4826 V01: SEISMIC MARGIN REVIEW OF THE MAINE YANKEE ATOMIC POWER STATION.Volume 1.Summary Report.
NUREG/CR-4826 V02: SEISMIC MARGIN REVIEW OF THE MAINE YANKEE ATOMIC POWER STATION.Volume 2.Systems Analysis.
NUREG/CR-4826 V03: SEISMIC MARGIN REVIEW OF THE MAINE YANKEE ATOMIC POWER STATION.Volume 3.Fragility Analysis.
NUREG/CR-4852: THE MEERS FAULT:TECTONIC ACTIVITY IN SOUTHWESTERN OKLAHOMA.

NUREG/CR-4853: APPROXIMATE METHODS FOR FRACTURE ANALYSES OF THROUGH-WALL CRACKED PIPES.
NUREG/CR-4859: SEISMIC FRAGILITY TEST OF A 6-INCH DIAMETER PIPE SYSTEM.
NUREG/CR-4861: DEVELOPMENT OF SITE SPECIFIC RESPONSE SPECTRA.
DIVISION OF REGULATORY APPLICATIONS (860720-870413)
NUREG/CR-4409 V02: DATA BASE ON NUCLEAR POWER PLANT DOSE REDUCTION RESEARCH PROJECTS.
NUREG/CR-4856: FEASIBILITY STUDY ON A DATA-BASED SYSTEM FOR DECISIONS REGARDING OCCUPATIONAL RADIATION PROTECTION MEASURES.
DIVISION OF REACTOR SYSTEM SAFETY (860720-870413)
NUREG/CR-3468: HYDROGEN:AIR:STEAM FLAMMABILITY LIMITS AND COMBUSTION CHARACTERISTICS IN THE FITS VESSEL.
NUREG/CR-4016 V02: APPLICATION OF SLIM-MAUD:A TEST OF AN INTERACTIVE COMPUTER-BASED METHOD FOR ORGANIZING EXPERT ASSESSMENT OF HUMAN PERFORMANCE AND RELIABILITY.Volume II:Appendices.
NUREG/CR-4550 V03: ANALYSIS OF CORE DAMAGE FREQUENCY FROM INTERNAL EVENTS:SURRY UNIT 1.
NUREG/CR-4550 V04: ANALYSIS OF CORE DAMAGE FREQUENCY FROM INTERNAL EVENTS:PEACH BOTTOM UNIT 2.
NUREG/CR-4551 V1 DRF: EVALUATION OF SEVERE ACCIDENT RISKS AND THE POTENTIAL FOR RISK REDUCTION:SURRY POWER STATION,UNIT 1.Draft For Comment.
NUREG/CR-4610: EFFECTS OF LATERAL SEPARATION OF OXIDIC AND METALLIC CORE DEBRIS ON THE BWR MK I CONTAINMENT DRYWELL FLOOR.
NUREG/CR-4616: ROOT CAUSES OF COMPONENT FAILURES PROGRAM.Methods And Applications.
NUREG/CR-4672: ANALYSIS OF INSTRUMENT TUBE RUPTURES IN WESTINGHOUSE 4-LOOP PRESSURIZED WATER REACTORS.
NUREG/CR-4689: THERMAL-HYDRAULIC AND CHARACTERISTIC MODELS FOR PACKED DEBRIS BEDS.
NUREG/CR-4696: CONTAINMENT VENTING ANALYSIS FOR THE PEACH BOTTOM ATOMIC POWER STATION.
NUREG/CR-4700 V1 DRF: CONTAINMENT EVENT ANALYSIS FOR POSTULATED SEVERE ACCIDENTS. Surry Power Station,Unit 1.Draft For Comment.
NUREG/CR-4718: EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES PROGRAM.Summary Report.
NUREG/CR-4742: MELPROG-PWR/MOD1 ANALYSIS OF A TMLB' ACCIDENT SEQUENCE.
NUREG/CR-4752: COINCIDENT STEAM GENERATOR TUBE RUPTURE AND STUCK-OPEN SAFETY RELIEF VALVE CARRYOVER TESTS.MB-2 Steam Generator Transient Response Test Program.
NUREG/CR-4788: AN ANALYTICAL AND EXPERIMENTAL INVESTIGATION OF NATURAL CIRCULATION TRANSIENTS IN A MODEL PRESSURIZED WATER REACTOR.
NUREG/CR-4789: THE SIMULATION OF THERMOHYDRAULIC PHENOMENA IN A PRESSURIZED WATER REACTOR PRIMARY LOOP.
NUREG/CR-4798: IRON OXIDE AEROSOL EXPERIMENTS IN STEAM-AIR ATMOSPHERES:NSPP TESTS 501-505 AND 511,DATA RECORD REPORT.
NUREG/CR-4824: EVALUATION OF INTEGRAL CONTINUING EXPERIMENTAL CAPABILITY (CEC) CONCEPTS FOR LIGHT WATER REACTOR RESEARCH - PWR SCALING CONCEPTS.
NUREG/CR-4829 V01: SHIPPING CONTAINER RESPONSE TO SEVERE HIGHWAY AND RAILWAY ACCIDENT CONDITIONS.Main Report.
NUREG/CR-4829 V02: SHIPPING CONTAINER RESPONSE TO SEVERE HIGHWAY AND RAILWAY ACCIDENT CONDITIONS.Appendices.
NUREG/CR-4843 V01: UNIVERSITY OF MARYLAND AT COLLEGE PARK (UMCP) 2X4 LOOP TEST FACILITY.Annual Report For 1985.
DIVISION OF ACCIDENT EVALUATION (POST 840101)
NUREG/CR-4793: RESULTS OF SEMISCALE MOD-2C SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITHOUT HPI (S-NH) EXPERIMENT SERIES.

EDO-RESOURCE MANAGEMENT
COST & STATISTICAL ANALYSIS STAFF (861123-870413)
NUREG/CR-4012 V02: REPLACEMENT ENERGY COSTS FOR NUCLEAR ELECTRICITY-GENERATING UNITS IN THE UNITED STATES:1987-1991.

OFFICE OF NUCLEAR REACTOR REGULATION (POST 4/28/80)
OFFICE OF NUCLEAR REACTOR REGULATION, DIRECTOR (851125-870411)
NUREG/CR-3469 V03: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS. Annotated Bibliography Of Selected Readings In Radiation Protection And ALARA.
DIVISION OF HUMAN FACTORS TECHNOLOGY (851125-870411)
NUREG/CR-3968: STUDY OF OPERATING PROCEDURES IN NUCLEAR POWER PLANTS. Practices And Problems.
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