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Vol. 12, No. 4

Regulatory and Technical Reports (Abstract Index Journal)

Annual Compilation for 1987

U.S. Nuclear Regulatory Commission

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Office of Administration and Resources Management
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PREFACE

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

Division of Publications Services
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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

Secondary Report Number Index
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NRC Originating Organization Index (International Agreements)
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Contractor Index
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Licensed Facility Index

A detailed explanation of the entries precedes each index.

The bibliographic elements of the main citations are the following:

Staff Report

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

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

Conference Report

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

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

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

Availability of NRC Publications

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

Superintendent of Documents
U.S. Government Printing Office
Post Office Box 37082
Washington, DC 20013-7082

You may charge any purchase to your GPO Deposit Account, MasterCard charge card, or VISA charge card by calling the GPO on (202)275-2060 or (202)275-2171. Non-U.S. customers must make payment in advance either by International Postal Money Order, payable to the Superintendent of Documents, or by draft on a United States or Canadian bank, payable to the Superintendent of Documents.

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. 450pp. 8704090032. 40468:027.

See NUREG-0020,V10,N09 abstract.

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

See NUREG-0020,V10,N09 abstract.

NUREG-0020 V10 N12: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of November 30,1986.(Gray Book I) ROSS,P.A. Division of Computer & Telecommunications Services (Post 870413). June 1987. 460pp. 8707060427. 41582:185.

See NUREG-0020,V10,N09 abstract.

NUREG-0020 V11 N01: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of December 31,1986.(Gray Book I) ROSS,P.A. Division of Computer & Telecommunications Services (Post 870413). September 1987. 473pp. 8710080382. 42994:112.

See NUREG-0020,V10,N09 abstract.

NUREG-0020 V11 N02: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of January 31,1987.(Gray Book I) SCHWARTZ,I. Division of Computer & Telecommunications Services (Post 870413). October 1987. 456pp. 8710200566. 43100:329.

See NUREG-0020,V10,N09 abstract.

NUREG-0020 V11 N03: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of February 28,1987.(Gray Book I) SCHWARTZ,I. Division of Computer & Telecommunications Services (Post 870413). October 1987. 465pp. 8710210040. 43123:291.

See NUREG-0020,V10,N09 abstract.

NUREG-0020 V11 N04: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of March 31,1987.(Gray Book I) SCHWARTZ,I. Division of Computer & Telecommunications Services (Post 870413). October 1987. 502pp. 8710230043. 43140:253.

See NUREG-0020,V10,N09 abstract.

NUREG-0020 V11 N05: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of April 30,1987.(Gray Book I) SCHWARTZ,I. Division of Computer & Telecommunications Services (Post 870413). October 1987. 495pp. 8711030219. 43244:047.

See NUREG-0020,V10,N09 abstract.

NUREG-0020 V11 N06: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of May 31,1987.(Gray Book I) SCHWARTZ,I. Division of Computer & Telecommunications Services (Post 870413). October 1987. 481pp. 8711030080. 43242:164.

See NUREG-0020,V10,N09 abstract.

NUREG-0020 V11 N07: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of June 30,1987.(Gray Book I) SCHWARTZ,I. Division of Computer & Telecommunications Services (Post 870413). October 1987. 508pp. 8710270253. 43185:327.

See NUREG-0020,V10,N09 abstract.

NUREG-0020 V11 N08: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of July 31,1987.(Gray Book I) SCHWARTZ,I. Division of Computer & Telecommunications Services (Post 870413). October 1987. 514pp. 8710280145. 43207:001.

See NUREG-0020,V10,N09 abstract.

NUREG-0020 V11 N09: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of August 31,1987.(Gray Book I) SCHWARTZ,I. Division of Computer & Telecommunications Services (Post 870413). December 1987. 496pp. 8712280075. 43822:105.

See NUREG-0020,V10,N09 abstract.

NUREG-0020 V11 N10: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of September 30,1987.(Gray Book I) SCHWARTZ,I. Division of Computer & Telecommunications Services (Post 870413). December 1987. 489pp. 8712300202. 43847:021.

See NUREG-0020,V10,N09 abstract.

NUREG-0020 V11 N11: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of October 31,1987.(Gray Book I) SCHWARTZ,I. Division of Computer & Telecommunications Services (Post 870413). December 1987. 425pp. 8801110233. 43973:167.

See NUREG-0020,V10,N09 abstract.

NUREG-0040 V10 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1986.(White Book) * Div of QA,Vendor & Technical Training Center Programs,IE (850212-870413). 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-0040 V11 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1987.(White Book) * Division of Reactor Inspection & Safeguards (Post 870411). May 1987. 235pp. 8706160106. 41311:032.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from January 1987 through March 1987. Also, included in this issue are the results of certain inspections performed prior to January 1987 that were not included in previous issues of NUREG-0040.

NUREG-0040 V11 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,April-June 1987.(White Book) * Division of Reactor Inspection & Safeguards (Post 870411). August 1987. 150pp. 8708270427. 42391:011.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from April 1987 through June 1987. Also, included in this issue are the results of certain inspections performed prior to April 1987 that were not included in previous issues of NUREG-0040.

NUREG-0040 V11 N03: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,July-September 1987.(White Book) * Division of Reactor Inspection & Safeguards (Post 870411). October 1987. 166pp. 8711090267. 43316:029.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from July 1987 through September 1987. Also included in this issue are the results of certain inspections performed prior to July 1987 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-0090 V09 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1986. * Office for Analysis & Evaluation of Operational Data, Director. April 1987. 60pp. 8705290301. 41115:296.

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

NUREG-0090 V09 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1986. * Office for Analysis & Evaluation of Operational Data, Director. July 1987. 150pp. 8708170105. 42171:217.

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 October 1 to December 31, 1986. During the report period, there were three abnormal occurrences at the nuclear power plants licensed to operate. The events were (1) the loss of low pressure service water systems at Oconee, (2) degraded safety systems due to incorrect torque switch settings on Rotork motor operators at Catawba and McGuire Nuclear Stations, and (3) a secondary system pipe break resulting in the death of four persons at Surry Unit 2. There were six abnormal occurrences at the other NRC licensees. One involved release of americium-241 inside a waste storage building at Wright-Patterson Air Force Base; three involved medical misadministrations, one therapeutic and two diagnostic; one involved a suspension of license for servicing teletherapy and radiography units, and one involved an immediate effective order modifying license and order to show cause issued to an industrial radiography company. There were no abnormal occurrences reported by the Agreement States. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0090 V10 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.January-March 1987. * Office for Analysis & Evaluation of Operational Data, Director. October 1987. 58pp. 8711230326. 43426:226.

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 January 1 to March 31, 1987. During the report period, there was one abnormal occurrence at the nuclear power plants licensed to operate. The item

involved the NRC suspension of power operations of the Peach Bottom Facility due to inattentiveness of the control room staff. There were seven abnormal occurrences at the other NRC licensees. Four involved diagnostic medical misadministrations; the other three involved breakdowns in management controls at three separate industrial radiography licensees. There were two abnormal occurrences reported by the Agreement States. Both involved breakdowns in management controls at industrial radiography licensees. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0090 V10 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1987. * Office for Analysis & Evaluation of Operational Data, Director. November 1987. 53pp. 8801070275. 43945:079.

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 to June 30, 1987. During the report period, there were no abnormal occurrences at the nuclear power plants licensed to operate. There were five abnormal occurrences at the other NRC licensees. Three involved medical misadministrations (two diagnostic and one therapeutic); one involved the issuance of an NRC Order to remove a hospital's radiation safety officer due to falsification of certain records; and one involved a significant breakdown in management and procedural controls at an industrial radiography licensee. There was one abnormal occurrence reported by an Agreement State (Idaho). The item involved radiographer overexposures. 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. 212pp. 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-0304 V12 N01: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1987,January-March. * Division of Publication Services (Post 870413). May 1987. 62pp. 8706030076. 41166:112.

See NUREG-0304,V11,N04 abstract.

NUREG-0304 V12 N02: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Second Quarter 1987,April-June. * Division of Publication Services (Post 870413). August 1987. 60pp. 8709040390. 42522:207.

See NUREG-0304,V11,N04 abstract.

NUREG-0304 V12 N03: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Third Quarter 1987,July-September. * Division of Publication Services (Post 870413). November 1987. 60pp. 8711250102. 43460:306.

See NUREG-0304,V11,N04 abstract.

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-0327 R04: OWNERS OF NUCLEAR POWER PLANTS.Percentage Ownership Of Commercial Nuclear Power Plants By Utility Companies. WOOD,R.S. Office of Nuclear Reactor Regulation, Director (Post 870411). August 1987. 38pp. 8708240084. 42316:072.

The report indicates percentage ownership of commercial nuclear power plants by utility companies. The report includes all plants operating, under construction, docketed for NRC safety and environmental reviews, or under NRC antitrust review, but does not include those plants announced but not yet under review or those plants formally cancelled. Part I of the report lists plants alphabetically with their associated applicants or licensees and percentage ownership. Part II list applicants or licensees alphabetically with their associated plants and percentage ownership. Part I also indicates which plants have received operating licenses (OLs).

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

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

NUREG-0383 V01 R10: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Packages. * Division of Safeguards & Transportation (Post 870413). November 1987. 603pp. 8712020110. 43524:244.

This directory contains a Summary Report of NRC Approved Packages (Volume 1), Certificates of Compliance (Volume 2), and a Summary Report of NRC Approved Quality Assurance Programs for Radioactive Material Packages (Volume 3). The purpose of this directory is to make available a convenient source of information on packagings which have been approved by the U.S. Nuclear Regulatory Commission. To assist in identifying packaging, an index by Model Number and corresponding Certificate of Compliance Number is included at the back of each volume of the directory. The Summary Report includes a listing of all users of each package design prior to the publication date of the directory. Shipments of radioactive material utilizing these packagings must be in accordance with the provisions of 49 CFR 173.471 and 10 CFR Part 71, as applicable. In satisfying the requirements of Section 71.12, it is the responsibility of the licensees to insure them that they have a copy of the current approval and conduct their transportation activities in accordance with an NRC approved quality assurance program. Copies of the current approval may be obtained from the U.S. Nuclear Regulatory Commission Public Document Room files (see Docket No. listed on each certificate) at 1717 H Street, Washington, DC 20555. Note that the general license of 10 CFR 71.12 does not authorize the receipt, possession, use or transfer of byproduct source, or special nuclear material; such authorization must be obtained pursuant to 10 CFR Parts 30 to 36, 40, 50, or 70.

NUREG-0383 V02 R10: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Certificates Of Compliance. * Division of Safeguards & Transportation (Post 870413). November 1987. 654pp. 8712020034. 43526:127.
See NUREG-0383,V01,R10 abstract.

NUREG-0383 V03 R07: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Summary Report Of NRC Approved Quality Assurance Programs For Radioactive Material Packages. * Division of Safeguards & Transportation (Post 870413). November 1987. 141pp. 8712020037. 43521:246.
See NUREG-0383,V01,R10 abstract.

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-0386 D04 R05: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.July 1972 - September 1986. * Office of the General Counsel. June 1987. 628pp. 8707130184. 41682:228.
This Revision 5 of the fourth edition of the NRC Staff Practice and Procedure Digest contains a digest of a number of Commission, Atomic Safety and Licensing Appeal Board, and Atomic Safety and Licensing Board decisions issued during the period July 1, 1972 to September 30, 1986, interpreting the NRC Rules of Practice in 10 CFR Part 2. This Revision 5 replaces in part earlier editions and supplements and includes appropriate changes reflecting the amendments to the Rules of Practice effective September 30, 1986.

NUREG-0386 D04 R06: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.July 1972 - Dec 1986. * Office of the General Counsel. November 1987. 597pp. 8712280136. 43819:001.
This revision 6 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 December 31, 1986, interpreting the NRC Rules of Practice in 10 CFR Part 2. This revision 6 replaces in part earlier editions and supplements and includes appropriate changes reflecting the amendments to the Rules of Practice effective December 31, 1986.

NUREG-0430 V07 N01: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.January-June 1986.(Gray Book II) * Ofc of Inspection & Enforcement, Director (820201-870413). 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-0430 V07 N02: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.July-December 1986.(Gray Book II) * Office of Nuclear Material Safety & Safeguards,Director. August 1987. 14pp. 8709090409. 42566:252.
See NUREG-0430,V07,N01 abstract.

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-0525 R13: SAFEGUARDS SUMMARY EVENT LIST (SSEL). * Division of Safeguards & Transportation (Post 870413). July 1987. 85pp. 8708240212. 42316:282.
See NUREG-0525,R12 abstract.

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. 342pp. 8704060097. 40398:060.
See NUREG-0540,V08,N10 abstract.

NUREG-0540 V09 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. February 1-28,1987. * Division of Publication Services (Post 870413). April 1987. 411pp. 8706240149. 41440:175.
See NUREG-0540,V08,N10 abstract.

NUREG-0540 V09 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. March 1-31,1987. * Division of Publication Services (Post 870413). May 1987. 469pp. 8706160119. 41315:256.
See NUREG-0540,V08,N10 abstract.

NUREG-0540 V09 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30,1987. * Division of Publication Services (Post 870413). June 1987. 482pp. 8706230495. 41434:117.
See NUREG-0540,V08,N10 abstract.

NUREG-0540 V09 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. May 1-31,1987. * Division of Publication Services (Post 870413). July 1987. 346pp. 8707210755. 41833:276.
See NUREG-0540,V08,N10 abstract.

NUREG-0540 V09 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. June 1-30,1987. * Division of Publication Services (Post 870413). August 1987. 505pp. 8709030196. 42471:111.
See NUREG-0540,V08,N10 abstract.

NUREG-0540 V09 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. July 1-31,1987. * Division of Publication Services (Post 870413). September 1987. 566pp. 8710220425. 43130:176.
See NUREG-0540,V08,N10 abstract.

NUREG-0540 V09 N08: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. August 1-31,1987. * Division of Publication Services (Post 870413). October 1987. 403pp. 8711030215. 43241:019.
See NUREG-0540,V08,N10 abstract.

NUREG-0540 V09 N09: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. September 1-30,1987. * Division of Publication Services (Post 870413). November 1987. 514pp. 8711230343. 43426:283.
See NUREG-0540,V08,N10 abstract.

NUREG-0540 V09 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. October 1-31,1987. * Division of Publication Services (Post 870413). December 1987. 434pp. 8801110224. 43975:315.
See NUREG-0540,V08,N10 abstract.

NUREG-0654 S01 R01: CRITERIA FOR PREPARATION AND EVALUATION OF RADIOLOGICAL EMERGENCY RESPONSE PLANS AND PREPAREDNESS IN SUPPORT OF NUCLEAR POWER PLANTS.Criteria For Utility Offsite Planning And Preparedness.Draft Report. * NRC - No Detailed Affiliation Given. * Federal Emergency Management Agency. November 1987. 35pp. 8712080435. FEMA REP-1. 43591:296.
This document has been developed for interim use in reviewing and evaluating utility-prepared offsite emergency plans and preparedness. This interim-use document is intended to be used with Section I and Appendices 1-5 of the existing NUREG-0654/FEMA-REP-1, Rev. 1. A notice has been provided in the "Federal Register" to announce the availability of this document and to invite public review and comment. Subsequent to the receipt of comments, this document will be issued in final. Except where specifically modified, the existing licensee-only evaluation criteria of the current Section II are not affected by this document. For those situations in which State and/or local governments are participating in the emergency planning process, the existing NUREG-0654/FEMA-REP-1 Rev. 1 evaluation criteria will apply.

NUREG-0683 S02: PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT RELATED TO DECONTAMINATION AND DISPOSAL OF RADIOACTIVE WASTES RESULTING FROM MARCH 28,1979 ACCIDENT AT THREE MILE ISLAND NUCLEAR STATION,UNIT 2.Final Supplement Dealing With Disposal Of.... * TMI-2 Cleanup Project Directorate. June 1987. 354pp. 8707080318. 41616:213.
In accordance with the National Environmental Policy Act, the Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Waste for the 1979 Accident at Three Mile Island Nuclear Station, Unit 2 (PEIS) has been supplemented. This supplement updates the environmental evaluation of accident-generated water disposal alternatives published in the PEIS, utilizing more complete and current information, and covering the licensee's proposal to dispose of the water by evaporation to the atmosphere. The staff concludes that the water can be disposed of without incurring significant environmental impact. The staff's evaluation of a number of disposal alternatives indicates that no alternative is clearly preferable to the others, and that the licensee's proposal method is satisfactory. The risks to the general public from exposure to radioactive effluents from any alternative have been quantitatively estimated and are very small fractions of the estimated normal incidence of cancer fatalities and genetic disorders. The most significant potential impact associated with any disposal alternative is the risk of physical injury associated with transportation accidents. Additionally, no significant impacts to aquatic or terrestrial biotic from any disposal alternative are expected.

NUREG-0728 R02: NRC INCIDENT RESPONSE PLAN. * Office for Analysis & Evaluation of Operational Data, Director. June 1987. 29pp. 8706240202. 41448:219.
The Nuclear Regulatory Commission (NRC) regulates civilian nuclear activities to protect the public health and safety and to preserve environmental quality. An Incident Response Plan had been developed and has now been revised for the second time to reflect current Commission policy. NUREG-0728, Rev 2 assigns responsibilities for responding to any potentially threatening incident involving NRC licensed activities and for assuring that the NRC will fulfill its statutory mission. Revision 2 was necessary to reflect organizational changes.

NUREG-0750 V24 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-September 1986. * Division of Publication Services (Post 870413). June 1987. 61pp. 8707020110. 41569:166.
Digests and indexes for issuances of the Commission, the Atomic Safety and Licensing Appeal Panel, the Atomic Safety and Licensing Board Panel, the Administrative Law Judge, the Director's Decisions, and the Denials of Petitions for Rulemaking are presented.

NUREG-0750 V24 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-December 1986. * Division of Publication Services (Post 870413). July 1987. 100pp. 8708270324. 42389:007.
See NUREG-0750,V24,I01 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. 101pp. 8704010165. 40319:061.
See NUREG-0750,V24,N01 abstract.

NUREG-0750 V24 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1986.Pages 489-679. * Division of Publication Services (Post 870413). April 1987. 202pp. 8705120123. 40899:257.
See NUREG-0750,V24,N01 abstract.

NUREG-0750 V24 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1986.Pages 681-768. * Division of Publication Services (Post 870413). May 1987. 99pp. 8706040313. 41180:238.
See NUREG-0750,V24,N01 abstract.

NUREG-0750 V24 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1986.Pages 769-930. * Division of Publication Services (Post 870413). June 1987. 170pp. 8707090378. 41633:182.
See NUREG-0750,V24,N01 abstract.

NUREG-0750 V25 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-March 1987. * Division of Publication Services (Post 870413). September 1987. 41pp. 8710290301. 43217:257.
See NUREG-0750,V24,I01 abstract.

NUREG-0750 V25 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1987.Pages 1-62. * Division of Publication Services (Post 870413). June 1987. 71pp. 8707210720. 41833:198.
See NUREG-0750,V24,N01 abstract.

NUREG-0750 V25 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1987.Pages 63-128. * Division of Publication Services (Post 870413). July 1987. 74pp. 8708060402. 42072:330.
See NUREG-0750,V24,N01 abstract.

NUREG-0750 V25 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1987.Pages 129-266. * Division of Publication Services (Post 870413). August 1987. 148pp. 8709110219. 42626:269.
See NUREG-0750,V24,N01 abstract.

NUREG-0750 V25 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1987.Pages 267-416. * Division of Publication Services (Post 870413). September 1987. 160pp. 8710080359. 42990:081.
See NUREG-0750,V24,N01 abstract.

NUREG-0762 R1 DRF FC: STANDARD FORMAT AND CONTENT FOR EMERGENCY PLANS FOR FUEL-CYCLE AND MATERIALS FACILITIES.Draft Report For Comment. * Division of Industrial & Medical Nuclear Safety (Post 870729). November 1987. 35pp. 8711250119. 43450:155.
This report is issued as guidance to those fuel-cycle and major materials licensees who are required by the NRC to prepare and submit an emergency plan. This Standard Format has been prepared to help ensure uniformity and completeness in the preparation of those plans.

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-0781 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SOUTH TEXAS PROJECT,UNITS 1 AND 2.Docket Nos. 50-498 And 50-499. (Houston Lighting And Power Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). May 1987. 171pp. 8706240260. 41441:226.
The Safety Evaluation Report issued in April 1986 provided the results of the NRC staff's review of the Houston Lighting and Power Company's application for licenses to operate the South Texas Project. The facility consists of two pressurized water nuclear reactors located in Matagorda County, Texas. Supplement No. 1 issued in September 1986 updated the information contained in the Safety Evaluation Report and addressed the ACRS Report issued on June 10, 1986. Supplement No. 2 issued in January 1987 addressed and resolved some of the outstanding issues remaining after issuance of the Safety Evaluation Report and Supplement No. 1. This Supplement No. 3 also addresses and resolves some of the outstanding issues remaining after issuance of the SER and Supplements 1 and 2.

NUREG-0781 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SOUTH TEXAS PROJECT,UNITS 1 AND 2.Docket Nos. 50-498 And 50-499. (Houston Lighting And Power Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). July 1987. 162pp. 8708240094. 42323:002.
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 reactors located in Matagorda County, Texas. Supplement No. 1, issued in September 1986 updated the information contained in the Safety Evaluation Report and addressed the ACRS Report issued on June 10, 1986. Supplement No. 2, issued in January 1987 addressed and resolved some of the outstanding issues remaining after issuance of the Safety Evaluation Report and Supplement No. 1. Supplement No. 3, issued in May 1987 addressed and resolved some of the outstanding issues remaining after issuance of the Safety Evaluation Report and Supplement Nos. 1 and 2. Supplement No. 4 also addresses and resolves some of the outstanding issues remaining after issuance of the Safety Evaluation Report and Supplement Nos. 1, 2, and 3.

NUREG-0800 06.5.2 R2: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Proposed Revision 2 To Section 6.5.2, "Containment Spray As A Fission Product Cleanup System." For Comment. * Office of Nuclear Reactor Regulation, Director (Post 870411). April 1987. 80pp. 8704270036. 40682:016.
Proposed revision 2 to SRP Section 6.5.2 would incorporate changes in the requirements for containment spray chemical additive systems, and explicitly states computational models which had only appeared in references in previous revisions. The requirement for immediate initiation of caustic addition to the spray would be deleted, and the minimum pH to be achieved would be reduced from 8.5 to 7. If adopted, this revision would be required to be used for future plants, and would be optional for present licensees. The proposed revision is accompanied by a regulatory analysis and two supporting technical documents.

NUREG-0800 06.5.5 R0: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Proposed Revision 0 To New SRP Section 6.5.5, "Pressure Suppression Pools As Fission Product Clean-Up Systems." For Comment. * Office of Nuclear Reactor Regulation, Director (Post 870411). April 1987. 51pp. 8704280166. 40716:350.
Proposed new SRP Section 6.5.5 would provide acceptance criteria and review procedures to be used in assessing the role of pressure suppression pools as fission product cleanup systems following potential reactor accidents. A calculational model to account for drywell bypass is given, and minimum fission product decontamination factors are listed for use in instances in which no detailed calculations of pool scrubbing have been performed. The proposed section is accompanied by a regulatory analysis and a supporting technical report.

NUREG-0825 S01: INTEGRATED PLANT SAFETY ASSESSMENT.SYSTEMATIC EVALUATION PROGRAM,YANKEE NUCLEAR POWER STATION.Docket No. 50-029. (Yankee Atomic Electric Company) * Division of Reactor Projects - I/II (Post 870411). October 1987. 34pp. 8710220292. 43130:143.
The U.S. Nuclear Regulatory Commission (NRC) has prepared Supplement 1 to the final Integrated Plant Safety Assess-

ment Report (IPSAR) (NUREG-0825), under the scope of the Systematic Evaluation Program (SEP), for Yankee Atomic Electric Company's Yankee Nuclear Power Station located in Rowe, Massachusetts. The SEP was initiated by the NRC to review the design of older operating nuclear power plants to reconfirm and document their safety. This report documents the review completed under the SEP for those issues that required refined engineering evaluations or the continuation of ongoing evaluations after the Final IPSAR for the Yankee plant was issued. The review has provided for (1) an assessment of the significance of differences between current technical positions on selected safety issues and those that existed when Yankee was licensed, (2) a basis for deciding how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety.

NUREG-0837 V06 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, July-September 1986. JANG,J.; RABATIN,K.; COHEN,L. Region 1, Ofc of the 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-0837 V06 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, October-December 1986. JANG,J.; COHEN,L. Region 1, Ofc of the Director. April 1987. 324pp. 8704280116. 40712:118.

This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facility sites throughout the country for the fourth quarter of 1986.

NUREG-0837 V07 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, January-March 1987. JANG,J.; MCNAMARA,N.; COHEN,L. Region 1, Ofc of the Director. July 1987. 226pp. 8708100398. 42095:325.

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 first quarter of 1987.

NUREG-0837 V07 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, April-June 1987. JANG,J.; MCNAMARA,N.; COHEN,L. Region 1, Ofc of the Director. October 1987. 226pp. 8710260458. 43176:145.

This report provides the status and results of the NRC Thermoluminescent (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facility sites throughout the country for the second quarter of 1987.

NUREG-0837 V07 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, July-September 1987. STRUCKMEYER,R.; MCNAMARA,N.; COHEN,L. Region 1, Ofc of the Director. December 1987. 229pp. 8801110268. 43977:343.

This report provides the status and results of the NRC Thermoluminescent Dosimetry (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 1987.

NUREG-0852 S03: SAFETY EVALUATION REPORT RELATED TO THE FINAL DESIGN OF THE STANDARD NUCLEAR STEAM SUPPLY REFERENCE SYSTEM,CESSAR System 80.Docket No. 50-470.(Combustion Engineering,Incorporated) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). December 1987. 78pp. 8801060284. 43940:294.

Supplement No. 3 to the Safety Evaluation Report for the application filed by Combustion Engineering, Inc., for a Final Design Approval for the Combustion Engineering Standard Safety Analysis Report (CESSAR) (Docket No. STN 50-470) 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 by providing (1) the evaluation of additional information submitted by the applicant since Supplement No. 2 to the Safety Evaluation Report was issued and (2) the evaluation of the matters that staff had under review when Supplement No. 2 to the Safety Evaluation Report was issued.

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-0857 S12: 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 Reactor Projects - III,IV,V & Special Projects (Post 870411). November 1987. 98pp. 8712150439. 43723:320.

Supplement No. 12 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 Stations, 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 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 licensee since Supplement No. 11 was issued and (2) other matters requiring staff review since Supplement No. 11 was issued, specifically those issues that required resolution before Unit 3 full 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-0885 I06: U.S. NUCLEAR REGULATORY COMMISSION POLICY AND PLANNING GUIDANCE 1987. * Commissioners. September 1987. 58pp. 8710010440. 42883:183.

The purposes of the Policy and Planning Guidance document are: to set forth the regulatory approach of the Nuclear Regulatory Commission and provide supporting principles to the approach; to state the major policies and planning objectives of the Commission; and to provide a common basis for the development of programs, the establishment of priorities, and the allocation of resources.

NUREG-0896 S07: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SEABROOK STATION,UNITS 1 AND 2.Docket Nos. 50-443 And 50-444.(Public Service Company Of New Hampshire,et al) * Division of Reactor Projects - I/II (Post 870411). October 1987. 76pp. 8711100384. 43341:178.

Supplement No. 7 to the Safety Evaluation Report related to operation of the Seabrook Station, Unit 1 addresses items relating to the issuance of a 5% low power license. The report relates to the application filed by the Public Service Company of New Hampshire for a license to operate the Seabrook Station, Unit 1 located in Rockingham County, New Hampshire.

NUREG-0904 S01: DRAFT SUPPLEMENT TO THE FINAL ENVIRONMENTAL STATEMENT RELATED TO THE DECOMMISSIONING OF THE RARE EARTHS FACILITY, WEST CHICAGO,ILLINOIS.Docket No. 40-2061.(Kerr-McGee) * Division of Fuel Cycle, Medical, Academic & Commercial Use Safety 870413-870728. June 1987. 492pp. 8707060327. 41584:239.

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

NUREG-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. 314pp. 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-0936 V05 N04: NRC REGULATORY AGENDA.Quarterly Report,October-December 1986. * Division of Rules & Records (Post 870413). May 1987. 160pp. 8706150112. 41300:247.

See NUREG-0936,V05,N03 abstract.

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

See NUREG-0936,V05,N03 abstract.

NUREG-0936 V06 N02: NRC REGULATORY AGENDA.Quarterly Report,April-June 1987. * Division of Rules & Records (Post 870413). July 1987. 128pp. 8708260001. 42333:062.

See NUREG-0936,V05,N03 abstract.

NUREG-0936 V06 N03: NRC REGULATORY AGENDA.Quarterly Report,July-September 1987. * Division of Rules & Records (Post 870413). November 1987. 117pp. 8712150231. 43724:153.

See NUREG-0936,V05,N03 abstract.

NUREG-0940 V05 N04: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,October-December 1986. * Ofc of Inspection & Enforcement, Director (820201-870413). 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-0940 V06 N01: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,January-March 1987. * Ofc of Enforcement (Post 870413). June 1987. 337pp. 8706240320. 41450:266.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January - March 1987) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

NUREG-0940 V06 N02: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June 1987. * Ofc of Enforcement (Post 870413). August 1987. 399pp. 8709090464. 42564:282.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1987) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoid-

ing future violations similar to those described in this publication.

NUREG-0940 V06 N03: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,July-September 1987. * Ofc of Enforcement (Post 870413). December 1987. 218pp. 8801110274. 43972:308.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (July - September 1987) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

NUREG-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. 409pp. 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-0980 R03: NUCLEAR REGULATORY LEGISLATION. HOSPODOR,S. Office of the General Counsel. April 1987. 115pp. 8705290061. 41116:266.

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

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

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

NUREG-1002 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BRAIDWOOD STATION,UNITS 1 AND 2.Docket Nos. 50-456 And 50-457.(Commonwealth Edison Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). July 1987. 22pp. 8708040187. 42039:049.

In November 1983, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1002) regarding the application filed by the Commonwealth Edison Company, as applicant and owner, for a license to operate Braidwood Station, Units 1 and 2 (Docket Nos. 50-456 and 50-457). The first supplement to NUREG-1002 was issued in September 1986; the second supplement to NUREG-1002 was issued in October 1986; the third supplement to NUREG-1002 was issued in May 1987. This fourth supplement to NUREG-1002 reports the status of certain items that remained unresolved at the time Supplement 3 was published. The facility is located in Reed Township, Will County, Illinois.

NUREG-1002 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BRAIDWOOD STATION,UNITS 1 AND 2.Docket Nos. 50-456 And 50-457.(Commonwealth Edison Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). December 1987. 21pp. 8801110217. 43975:207.

In November 1983, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1002) regarding the application filed by the Commonwealth Edison Company, as applicant and owner, for a license to operate Braidwood Station, Units 1 and 2 (Docket Nos. 50-456 and 50-457). The first supplement to NUREG-1002 was issued in September 1986; the second supplement was issued in October 1986; the third supplement was issued in May 1987; the fourth supplement was issued in July 1987. This fifth supplement to NUREG-1002 is in support of the low-power license for Unit 2 and provides the status of certain items that remained unresolved at the time Supplement 4 was published. The facility is located in Reed Township, Will County, Illinois.

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

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

NUREG-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 cri-

teria 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-1047 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION,UNIT 2.Docket No. 50-410.(Niagara Mohawk Power Corporation,et al) * Division of Reactor Projects - I/II (Post 870411). July 1987. 65pp. 8707230283. 41879:171.

This report supplements the Safety Evaluation Report (NUREG-1047, February 1985) for the application filed by Niagara Mohawk Power Corporation, as applicant and co-owner, for the license to operate Nine Mile Point Nuclear Station, Unit 2 (Docket No. 50-410). It has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located near Oswego, New York. This report supports the issuance of the full-power license for Nine Mile Point Nuclear Station, Unit No. 2.

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. 92pp. 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-1057 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BEAVER VALLEY POWER STATION,UNIT 2.Docket No. 50-412.(Duquesne Light Company,et al) * Division of Reactor Projects - I/II (Post 870411). May 1987. 176pp. 8706120079. 41186:104.

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

NUREG-1057 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BEAVER VALLEY POWER STATION,UNIT 2.Docket No. 50-412.(Duquesne Light Company,et al) * Division of Reactor Projects - I/II (Post 870411). August 1987. 83pp. 8709090412. 42566:356.

Supplement No. 6 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 licensees since Supplement No. 5 was issued, and (2) matters that the staff had under review when Supplement No. 5 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-1100 V03 ADD: BUDGET ESTIMATES.Fiscal Years 1988-1989. * Division of Budget & Analysis (Post 870413). May 1987. 44pp. 8706300181. 41491:060.

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

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

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

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

This compilation contains 58 ACRS reports submitted to the Commission or to the Executive Director for Operations during

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

NUREG-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, Ogelthorpe 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. 126pp. 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-1144 R01: NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PLAN. VORA,J.P. Division of Engineering (Post 870413). September 1987. 135pp. 8710090155. 43010:082.

The nuclear plant aging research described in this plan is intended to resolve issues related to the aging and service wear of equipment and systems and major components at commercial reactor facilities and their possible impact on plant safety. Emphasis has been placed on identification and characterization of the mechanisms of material and component degradation during service and evaluation of methods of inspection, surveillance, condition monitoring, and maintenance as means of mitigating such effects. Specifically, the goals of the program are as follows: (1) to identify and characterize aging and service wear effects which, if unchecked, could cause degradation of equipment, systems, and major components and thereby impair plant safety, (2) to identify methods of inspection, surveillance, and monitoring, or of evaluating residual life of equipment, systems, and major components, which will ensure timely detection of significant aging effects prior to loss of safety function, and (3) to evaluate the effectiveness of storage, maintenance, repair, and replacement practices in mitigating the rate and extent of degradation caused by aging and service wear.

NUREG-1145 V03: U.S. NUCLEAR REGULATORY COMMISSION 1986 ANNUAL REPORT. * Office of Administration & Resources Management, Director (Post 870413). June 1987. 258pp. 8707020359. 41567:001.

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

NUREG-1147 R01: SEISMIC SAFETY RESEARCH PROGRAM PLAN. * Division of Engineering (Post 870413). May 1987. 242pp. 8706240291. 41438:334.

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

NUREG-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 specif-

ic 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. 233pp. 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-1166: FINAL ENVIRONMENTAL STATEMENT FOR DECOMMISSIONING HUMBOLDT BAY POWER PLANT,UNIT 3.Docket No. 50-133.(Pacific Gas And Electric Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). April 1987. 164pp. 8705190499. 40978:319.

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

NUREG-1184 DRFT: INTEGRATED SAFETY ASSESSMENT REPORT,INTEGRATED SAFETY ASSESSMENT PROGRAM - MILLSTONE NUCLEAR POWER STATION,UNIT 1. Docket No. 50-245.(Northeast Nuclear Energy Co).Draft Report. * Office of Nuclear Reactor Regulation, Director (Post 870411). April 1987. 524pp. 8704270155. 40682:334.

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

NUREG-1185 V01 DRFT: INTEGRATED SAFETY ASSESSMENT REPORT.Integrated Safety Assessment Program.Haddam Neck Plant.Docket No. 50-213. (Connecticut Yankee Atomic Power Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). July 1987. 141pp. 8708260003. 42333:190.

The Integrated Safety Assessment Program (ISAP) was initiated in November 1984, by the U.S. Nuclear Regulatory Commission to conduct integrated assessments for operating nuclear power reactors. The integrated assessment is conducted on a plant-specific basis to evaluate all licensing actions, licensee initiated plant improvements and selected unresolved generic/safety issues to establish implementation schedules for each item. In addition, procedures will be established to allow for a periodic updating of the schedules to account for licensing issues that arise in the future. This report documents the review of Haddam Neck Plant, operated by Connecticut Yankee Atomic Power Company, which is one of two plants being reviewed under the pilot program for ISAP. This report indicates how 82 topics selected for review were addressed and presents the staff's recommendations regarding the corrective actions to resolve the 82 topics and other actions to enhance plant safety. The report is being issued in draft form to obtain comments from the licensee, nuclear safety experts, and the Advisory Committee for Reactor Safeguards. Once those comments have been resolved, the staff will present its positions, along with a long-term implementation schedule from the licensee, in the final version of this report.

NUREG-1185 V02 DRFT: INTEGRATED SAFETY ASSESSMENT REPORT.Integrated Safety Assessment Program.Haddam Neck Plant.Docket No. 50-213. (Connecticut Yankee Atomic Power Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). July 1987. 715pp. 8708260092. 42386:128.

See NUREG-1185,V01,DRFT abstract.

NUREG-1194: CONSTRUCTION APPRAISAL TEAM INSPECTION RESULTS ON WELDING AND NONDESTRUCTIVE EXAMINATION ACTIVITIES. WU,P.C.; SHAABAN,H.I. Division of Reactor Inspection & Safeguards (Post 870411). September 1987. 57pp. 8710050558. 42905:227.

This report summarizes data and findings of deficiencies and discrepancies in welding and nondestructive examination (NDE) activities identified by the U.S. Nuclear Regulatory Commission Construction Appraisal Team (CAT) during its inspection of 11 plants. The CAT reviewed selected welds and NDE packages in its inspection of the following plant areas: piping and pipe supports and/or restraints; modification and installation of reactor internals; electrical installations and electrical supports; instrumentation tubing and supports; heating, ventilation, and air conditioning (HVAC) systems and supports; fabrication and erection of structural steel; fabrication of refueling cavity and spent fuel pool liner; containment liner and containment penetrations; and fire protection systems. The CAT inspected both structural welds and pressure-retaining welds and reviewed welder qualification test records and welding procedure documents for code compliance. The NDE activities that were evaluated included visual examination, magnetic particle examination, liquid penetrant examination, ultrasonic examination, and radiographic examination of welds.

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. Div of Emergency Preparedness & Engineering Response,IE(850212-870413). 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. Div of Emergency Preparedness & Engineering Response,IE(850212-870413). 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. Div of Emergency Preparedness & Engineering Response,IE(850212-870413). 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. Div of Emergency Preparedness & Engineering Response,IE(850212-870413). 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. Div of Emergency Preparedness & Engineering Response,IE(850212-870413). 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-1213 R01: PLANS AND SCHEDULES FOR IMPLEMENTATION OF U.S. NUCLEAR REGULATORY COMMISSION RESPONSIBILITIES UNDER THE LOW-LEVEL RADIOACTIVE WASTE POLICY AMENDMENTS ACT OF 1985 (P.L. 99-240). DUNKELMAN,M.M. Division of Low Level Waste Management & Decommissioning (Post 870413). August 1987. 100pp. 8709090253. 42564:182.

The purpose of this document is to make available to the States and other interested parties, the plans and schedules for the U.S. Nuclear Regulatory Commission's (NRC's) implementation of its responsibilities under the Low-Level Radioactive Waste Policy Amendments Act of 1985 (P.L. 99-240) (LLRWPAA). This document identifies the provisions of the LLRWPAA which affect the programs of the NRC, identifies what the NRC must do to fulfill each of its requirements under the LLRWPAA, and establishes schedules for carrying out these requirements. Revision 1 of this document includes the accomplishments and schedule revisions made by NRC since July 1, 1986.

NUREG-1214 R01: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. JOHNSON,M.R.; WATERMAN,D.K. Div of Inspection Programs,IE (850212-870413). April 4, 1987. 69pp. 8704270378. 40704:218.

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

NUREG-1214 R02: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. RAMEY-SMITH,A. Division of Licensee Performance & Quality Evaluation (Post 870411). October 1987. 70pp. 8711030070. 43243:337.

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

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

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

NUREG-1231: SAFETY EVALUATION REPORT RELATED TO THE BABCOCK AND WILCOX OWNERS GROUP PLANT REASSESSMENT PROGRAM. SIEGEL,B. Office of Nuclear Reactor Regulation, Director (Post 870411). November 1987. 232pp. 8711250145. 43449:095.

After the accident at Three Mile Island, Unit 2, nuclear power plant owners made a number of improvements to their nuclear facilities. Despite these improvements, the U.S. Nuclear Regulatory Commission (NRC) staff is concerned that the number and complexity of events at Babcock & Wilcox (B&W) nuclear plants have not decreased as expected. This concern was reinforced by the June 9, 1985 total loss- of-feedwater event at Davis-Besse Nuclear Power Station and the December 26, 1985 overcooling transient at Rancho Seco Nuclear Generating Station. By letter dated January 24, 1986, the Executive Director of Operations (EDO) informed the Chairman of the B&W Owners Group (BWOG) that a number of recent events at B&W-designed reactors have led the NRC staff to conclude that the basic requirements of B&W reactors need to be reexamined. In its February 13, 1986 response to the EDO's letter, the BWOG committed to lead an effort to define concerns relative to reducing the frequency of reactor trips and the complexity of post-trip response in B&W plants. The BWOG submitted a description of the B&W program entitled "Safety and Performance Improvement Program" (BAW-1919) on May 15, 1986. Five revisions to BAW-1919 have also been submitted. The NRC staff has reviewed BAW- 1919 and its five revisions and presents its evaluation in this report.

NUREG-1232 V01: SAFETY EVALUATION REPORT ON TENNESSEE VALLEY AUTHORITY: Revised Corporate Nuclear Performance Plan. * Ofc of Special Projects. July 1987. 64pp. 8710080392. 42995:225.

This Safety Evaluation Report on the information submitted by the Tennessee Valley Authority (TVA) in its revised Corporate Nuclear Performance Plan, through Revision 4, has been prepared by the U.S. Nuclear Regulatory Commission staff. The TVA Corporate Nuclear Performance Plan addresses those corporate concerns identified by the staff in its letter to TVA dated September 17, 1985. On the basis of its review, the staff finds TVA's revised Corporate Nuclear Performance Plan (Revision 4) acceptable.

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

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

NUREG-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-1239: ENVIRONMENTAL ASSESSMENT FOR RENEWAL OF SOURCE MATERIAL LICENSE NO. STB-401.Docket No. 40-6563.(Columbium-Tantalum Division,Mallinckrodt,Inc.) * Division of Fuel Cycle, Medical, Academic & Commercial Use Safety 870413-870728. April 1987. 69pp. 8705190502. 40980:267.

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

NUREG-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 for:h 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-1244: PLAN FOR INTEGRATING TECHNICAL ACTIVITIES WITHIN THE U.S. NRC AND ITS CONTRACTORS IN THE AREA OF THERMAL HYDRAULICS. * Office of Nuclear Regulatory Research, Director (Post 860720). April 1987. 42pp. 8705120074. 40896:152.

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

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

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

NUREG-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. 458pp. 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. 490pp. 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-1250 R01: REPORT ON THE ACCIDENT AT THE CHERNOBYL NUCLEAR POWER STATION. * NRC - No Detailed Affiliation Given. * Energy, Dept. of. *; et al. Electric Power Research Institute. December 1987. 400pp. 8801110242. 43977:121.

See NUREG-1250 abstract.

NUREG-1251 DRFT FC: IMPLICATIONS OF THE ACCIDENT AT CHERNOBYL FOR SAFETY REGULATION OF COMMERCIAL NUCLEAR POWER PLANTS IN THE UNITED STATES.Draft For Comment. * NRC - No Detailed Affiliation Given. August 1987. 98pp. 8709040049. 42507:277.

This draft report issued for comment was prepared by the Nuclear Regulatory Commission (NRC) staff to assess the implications of the accident at the Chernobyl nuclear power plant as they relate to reactor safety regulation for commercial nuclear power plants in the United States. The facts used in this assessment have been drawn from the U.S. fact-finding report (NUREG-1250) and its sources.

NUREG-1253: TECHNICAL SPECIFICATIONS FOR NINE MILE POINT NUCLEAR STATION,UNIT 2.Docket No. 50-410.(Niagara Mohawk Power Corporation,et al) * Division of Reactor Projects - I/II (Post 870411). July 1987. 526pp. 8707210703. 41832:032.

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

NUREG-1255: TECHNICAL SPECIFICATIONS FOR SOUTH TEXAS PROJECT,UNIT 1. Docket No. 50-498.(Houston Lighting and Power Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). August 1987. 445pp. 8709090260. 42577:044.

The South Texas Project, 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 reactor facility as set forth in Section 50.36 of 10 CFR 50 for the protection of the health and safety of the public.

NUREG-1256 V01: INCENTIVE REGULATION OF NUCLEAR POWER PLANTS BY STATE PUBLIC UTILITY COMMISSIONS. PETERSEN,J.C. Program Management, Policy Development & Analysis Staff (Post 870411). December 1987. 59pp. 8712240211. 43793:245.

This is a report on incentive regulation of nuclear power plants by State public utility commissions (PUCs). Economic performance incentives established by State PUCs are applicable to the construction or operation of about 45 nuclear power reactors owned by 30 utilities in 17 States. The NRC staff monitors development of the incentives and periodically provides an updated report on all nuclear plan incentives to its Regional Offices. The staff maintains contact with the PUCs and the utilities responsible for implementing the incentives in order to obtain the updated information and to consider potential safety effects of the incentives. This report presents the NRC staff's concerns on potential safety effects of economic performance incentives. It also includes a plant-by-plant survey that describes the mechanics of each incentive and discusses the financial effects of the incentive on the utility-owner(s) of the plant.

NUREG-1258: EVALUATION PROCEDURE FOR SIMULATION FACILITIES CERTIFIED UNDER 10 CFR 55.Final Report. WACHTEL,J.; PLOTT,C.; LAUGHERY,K.R.; et al. Division of Licensee Performance & Quality Evaluation (Post 870411). December 1987. 141pp. 8801060431. 43929:099.

This document describes the procedure to be followed by NRC for the inspection of simulation facilities certified under 10 CFR 55. Inspections are divided into four areas based on types of evaluations conducted: 1) performance testing; 2) physical fidelity/human factors; 3) control capabilities; and 4) design, updating, modification, and testing. NRC staff representing several disciplines including license examiners, operations specialists, and human factors experts, under direction of a team leader, will perform these inspections. A simulation facility inspection may include on-site and off-site phases. The off-site phase will consist of an examination of simulation facility documentation and an identification of those operations and procedures which may be considered for use in performance testing during the on-site phase. In the on-site review, the staff will work closely with the facility licensee to conduct a sound and fair inspection and to evaluate the results of those tests that are conducted. Inspection findings will be based on staff judgment of the simulation facility's compliance w/10 CFR 55.45. Findings may range from "no adverse impact on the conduct of operating tests" through degrees of "adverse impact requiring correction" to "adverse impacts so serious that the simulation facility may not be used in the conduct of operating tests until the discrepancies are corrected and the simulation facility is recertified to the NRC."

NUREG-1258 DRFT: EVALUATION PROCEDURE FOR SIMULATION FACILITIES CERTIFIED UNDER 10 CFR 55.Draft Report. LAUGHERY,K.R.; PLOTT,C.; WACHTEL,J. Division of Human Factors Technology (851125-870411). March 1987. 141pp. 8704270073. 40682:096.

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

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

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

NUREG-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-1261: TECHNICAL SPECIFICATIONS FOR BRAIDWOOD STATION,UNITS 1 AND 2.Docket Nos. 50-456 And 50-457.(Commonwealth Edison Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). May 1987. 490pp. 8707140211. 41692:120.

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

NUREG-1262: ANSWERS TO QUESTIONS AT PUBLIC MEETINGS REGARDING IMPLEMENTATION OF TITLE 10,CODE OF FEDERAL REGULATIONS,PART 55 ON OPERATORS' LICENSES. * Division of Licensee Performance & Quality Evaluation (Post 870411). November 1987. 176pp. 8712230304. 43800:324.

This document presents questions and answers based on transcripts of four public meetings (and from written questions submitted after meetings) conducted from April 9 to April 20, 1987 by the staff of the U.S. Nuclear Regulatory Commission. The meetings discussed implementation of the Commission's final rule governing Operator's Licenses and Conforming Amendments (10 CFR Parts 55 and 50). The rule became effective May 26, 1987 and is intended to clarify the regulations for issuing licenses to operators and senior operators; revise the requirements and scope of written examinations and operating tests for operators and senior operators, including a requirement for a simulation facility; clarify procedures for administering requalification examinations; and describe the form and content for operator license applications.

NUREG-1264: CONTAINMENT INTEGRITY RESEARCH PROGRAM PLAN. * Division of Engineering (Post 870413). August 1987. 38pp. 8709160104. 42693:130.

This report presents a plan for research on the question of containment performance in postulated severe accident scenarios. It focuses on the research being performed by the Structural and Seismic Engineering Branch, Division of Engineering, Office of Nuclear Regulatory Research. Summaries of the plans for this work have previously been published in the "Nuclear Power Plant Severe Accident Research Plan" (NUREG-0900). This report provides an update to reflect current status. This plan provides a summary of results to date as well as an outline of planned activities and milestones to the contemplated completion of the program in FY 1989.

NUREG-1265: UNCERTAINTY PAPERS ON SEVERE ACCIDENT SOURCE TERMS. * Office of Nuclear Regulatory Research, Director (Post 860720). May 1987. 194pp. 8707140200. 41691:216.

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

NUREG-1266 V01: NRC SAFETY RESEARCH IN SUPPORT OF REGULATION - 1986. * Office of Nuclear Regulatory Research, Director (Post 860720). September 1987. 66pp. 8710080407. 42991:067.

This report is the second in a series of annual reports responding to congressional inquiries as to the utilization of nuclear regulatory research. NUREG-1175, "NRC Safety Research in Support of Regulation," published in May 1986, reported major research accomplishments between about FY 1980 & FY 1985. This report narrates the accomplishments of FY 1986 and does not restate earlier accomplishments. Earlier research results are mentioned in the context of current results in the interest of continuity. Both the direct contributions to scientific and technical knowledge and their regulatory applications, when there has been a definite regulatory outcome during FY 1986, have been described.

NUREG-1269: LOSS OF RESIDUAL HEAT REMOVAL SYSTEM,Diablo Canyon Unit 2, April 10,1987. CREWS,J.L.; TRAMMELL,C.M.; LYON,W.C.; et al. Region 5, Ofc of the Director. June 1987. 107pp. 8707060049. 41584:040.

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

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

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

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

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

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

This annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) is devoted to the activities performed during the calendar year 1986. Comments and observations are provided on operating experience at nuclear power plants and other NRC licensees, including results from selected AEOD studies; summaries of abnormal occurrences involving U.S. nuclear plants; reviews of licensee event reports and their quality, reactor scram experience from 1984 to 1986, engineered safety features actuations, and the trends and patterns analysis program; and assessments of nonreactor and medical misadministration events. In addition, the report provides the year-end status of all recommendations included in AEOD studies, and listings of all AEOD reports issued from 1980 through 1986.

NUREG-1274: REVIEW PROCESS FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL LICENSE APPLICATION UNDER LOW-LEVEL RADIOACTIVE WASTE POLICY AMENDMENTS ACT. PITTIGLIO,C.L. Division of Low Level Waste Management & Decommissioning (Post 870413). August 1987. 22pp. 8708240196. 42319:186.

This document identifies and describes the U.S. Nuclear Regulatory Commission's (NRC's) process for licensing a low-level radioactive waste disposal facility within the time required by the Low-Level Radioactive Waste Policy Amendments Act of 1985. This document also estimates the level of effort and expertise that is needed to review a license application within the required time. It is intended to be used by the NRC staff as well as States and interested parties to provide a better understanding

of what the NRC envisions will be involved in licensing a low-level radioactive waste facility.

NUREG-1275V1: OPERATING EXPERIENCE FEEDBACK REPORT - NEW PLANTS. Commercial Power Reactors. DENNIG,R.L.; O'REILLY,P.D. Office for Analysis & Evaluation of Operational Data, Director. July 1987. 285pp. 8709040358. 42522:267.

This report documents a detailed review of the cause of unplanned events during the early months of licensed operation for plants licensed between March 1983 and April 1986. The major lessons and corrective actions that appear to have the greatest potential for improving the effectiveness of plant startups are provided for consideration through the operating experience feedback programs and activities of the industry and the NRC staff.

NUREG-1275 V02: OPERATING EXPERIENCE FEEDBACK REPORT - AIR SYSTEMS PROBLEMS. Commercial Power Reactors. ORNSTEIN,H.L. Office for Analysis & Evaluation of Operational Data, Director. December 1987. 129pp. 8801070069. 43943:154.

This report highlights significant operating events involving observed or potential failures of safety-related systems in U.S. plants that resulted from degraded or malfunctioning non-safety grade air systems. Based upon the evaluation of these events, the Office for Analysis and Evaluation of Operational Data (AEOD) concludes that the issue of air systems problems is an important one which requires additional NRC and industry attention. This report also provides AEOD's recommendations for corrective actions to deal with the issue.

NUREG-1276: TECHNICAL SPECIFICATIONS FOR BRAIDWOOD STATION,UNITS 1 AND 2.Docket Nos. 50-456 And 50-457.(Commonwealth Edison Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). July 1987. 493pp. 8707290233. 41952:082.

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

NUREG-1278: VOGTLE UNIT 1 READINESS REVIEW.Assessment Of Georgia Power Company Readiness Review Pilot Program. LEWIS,G. Division of Licensee Performance & Quality Evaluation (Post 870411). September 1987. 29pp. 8710080317. 42990:320.

Georgia Power Company (GPC) performed a readiness review at Vogtle Unit 1 as a pilot program. The pilot program was a new and innovative approach for the systematic and disciplined review, with senior management involvement, of GPC's implementation of design, construction, and operational readiness processes. The program's principal objective was to increase the level of assurance that quality programs at Vogtle Unit 1 have been accomplished in accordance with regulatory requirements. This report assesses the effectiveness of the GPC's readiness review pilot program (RRPP) at Vogtle Unit 1. It includes (1) an overview of what was experienced during the program's implementation, (2) an assessment of how well program objectives were met, and (3) lessons learned on the future use of the readiness review concept. Overall, GPC and the NRC staff believe that the RRPP at Vogtle Unit 1 was a success and that the program provided significant added assurance that Vogtle Unit 1 licensing commitments and NRC regulations have been adequately implemented. Although altering the NRC licensing review process for the few plants still in the construction pipeline may not be appropriate, licensees may benefit significantly by performing readiness reviews on their own initiative as GPC did for Vogtle.

NUREG-1279: TECHNICAL SPECIFICATIONS FOR BEAVER VALLEY POWER STATION, UNIT 2.Docket No. 50-412.(Duquesne Light Company,et al) * Division of Reactor Projects - I/II (Post 870411). August 1987. 424pp. 8709090526. 42575:340.

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

NUREG-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. 102pp. 8704080097. AA50-2 032. 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-1281: EVALUATION OF THE QUALIFICATION OF SPERT FUEL FOR USE IN NON-POWER REACTORS. * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). August 1987. 89pp. 8708240112. 42319:001.

This report summarizes the U.S. Nuclear Regulatory Commission staff's evaluation of the qualification of the stainless-steel-clad uranium/oxide (UO2) fuel pins for use in non-power reactors. The fuel pins were originally procured in the 1960's as part of the Special Power Excursion Reactor Test (SPERT) program. Argonne National Laboratory (ANL) examined 600 SPERT fuel pins to verify that the pins were produced according to specification and to assess their present condition. The pins were visually inspected under 6X magnification and by X-radiographic, destructive, and metallographic examinations. Spectrographic and chemical analyses were performed on the UO2 fuel. The results of the qualification examinations indicated that the SPERT fuel pins meet the requirements of Phillips Specification No. F-1-SPT and have suffered no physical damage since fabrication. Therefore, the qualification results give reasonable assurance that the SPERT fuel rods are suitable for use in non-power reactors provided that the effects of thin-wall defects in the region of the upper end cap and low-density fuel pellets are evaluated for the intended operating conditions.

NUREG-1282: SAFETY EVALUATION REPORT ON HIGH-URANIUM CONTENT, LOW-ENRICHED URANIUM-ZIRCONIUM HYDRIDE FUELS FOR TRIGA REACTORS.Docket No. 50-163.(GA Technologies,Incorporated) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). August 1987. 17pp. 8708240190. 42319:207.

The properties and performance of the TRIGA higher weight percent (w%), low-enriched uranium fuels are compared with those of the currently licensed 8.5-w% fuels. Neutron physics considerations, materials properties, irradiation performance, fission product release, pulse heating, and limiting design basis were evaluated. The performance of uranium-zirconium hydride fuel is substantially independent of uranium content up to 45-w% uranium. The behavior of the proposed 20- and 30-w% uranium fuels is indistinguishable from that of the currently ap-

proved 8.5-w% uranium fuel. Both the 20-20 and 30-20 uranium-zirconium hydride fuels are acceptable for use in the GA Mark F TRIGA reactor, and these two types of fuel are generically acceptable for use in other licensed TRIGA reactors, with the provision that case-by-case analyses discuss individual reactor operating conditions in applications for authorization to use them.

NUREG-1284: PROGRAM PLAN FOR CORRECTION OF U.S. INSTRUMENT DEGRADATION OR FAILURE IN THE UPPER PLENUM TEST FACILITY (UPTF) IN THE FEDERAL REPUBLIC OF GERMANY. RHEE,G.S.; CHEN,Y.S.; SHOTKIN,L.M. Division of Reactor & Plant Systems (Post 870413). July 1987. 254pp. 8708040274. 42060:142.

This report documents the investigation of the failure or degradation of some of the advanced two-phase flow instruments supplied by the USNRC to the Upper Plenum Test Facility (UPTF), including Tie-Plate Drag Bodies, Breakthrough Detectors, Loop Drag Disc paddles, Fluid Distribution Grid sensors, and Liquid Level Detector sensors. The exact causes for these instrument degradations or failures are not known, but several potential causes have been identified. For DBs and BTDs, the primary mechanism for the degradation seems to be a leakage in the Inconel 600 strain gage encapsulation and the subsequent burnout of the strain gage elements. For DDs the degradation or failure cause seems to be excessive loads. The degradation cause for most of the FDGs and LLDs seems to be either steam/water erosion or mechanical abrasion of sensor tips, which are made of sapphire. However, some of the FDG tips were found to be cracked also. The corrective actions are being directed toward finding out what are the primary causes for the instrument degradation or failure and what should be done to prevent these degradations or failures from continuing or recurring.

NUREG-1285: NRC STAFF EVALUATION OF THE GENERAL ELECTRIC COMPANY NUCLEAR REACTOR STUDY ("REED REPORT"). VIRGILIO,M.J.; BEVAN,R. Office of Nuclear Reactor Regulation, Director (Post 870411). July 1987. 63pp. 8708060378. 42071:199.

In 1975, the General Electric Company (GE) published a Nuclear Reactor Study, also referred to as "the Reed Report," an internal product-improvement study. GE considered the document "proprietary" and thus, under the regulations of the Nuclear Regulatory Commission (NRC), exempt from mandatory public disclosure. Nonetheless, members of the NRC staff reviewed the document in 1976 and determined that it did not raise any significant new safety issues. The staff also reached the same conclusion in subsequent reviews. However, in response to recent inquiries about the report, the staff re-evaluated the Reed Report from a 1987 perspective. This re-evaluation, documented in this staff report, concluded that (1) there are no issues raised in the Reed Report that support a need to curtail the operation of any GE boiling water reactor (BWR); (2) there are no new safety issues raised in the Reed Report of which the staff was unaware; and (3) although certain issues addressed by the Reed Report are still being studied by the NRC and the industry, there is no basis for suspending licensing and operation of GE BWR plants while these issues are being resolved.

NUREG-1286: SAFETY EVALUATION REPORT RELATED TO THE RESTART OF RANCHO SECO NUCLEAR GENERATING STATION,UNIT 1 FOLLOWING THE EVENT OF DECEMBER 26,1985.Docket No. 50-312.(Sacramento Municipal Utility District) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). October 1987. 199pp. 8711030225. 43256:001.

On December 26, 1985, the Rancho Seco Nuclear Generating Station experienced a loss of dc power within the integrated control system while the plant was at 76% power. The ensuing reactor trip was followed by a rapid overcooling transient and automatic initiation of the safety features actuation system. The overcooling transient continued until integrated control system dc power was restored 26 minutes after its loss. This report presents the staff's evaluation of the corrective actions taken by the licensee to prevent recurrence and to improve overall performance of Rancho Seco with respect to safety.

NUREG-1287: TECHNICAL SPECIFICATIONS FOR PALO VERDE NUCLEAR GENERATING STATION,UNIT 3.Docket No. 50-530.(Arizona Nuclear Power Project) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). November 1987. 489pp. 8711180065. 43386:112.

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-1290: DIFFERING PROFESSIONAL OPINIONS.1987 Special Review Panel. * NRC - No Detailed Affiliation Given. November 1987. 166pp. 8712180023. 43756:025.

In mid-1987, the Executive Director for Operations of the U.S. Nuclear Regulatory Commission appointed a Special Review Panel to review the existing NRC policy for expressing differing professional views and to recommend possible improvements to the policy, if warranted. Through its own efforts and those of three subpanels and three consultants, the Panel developed recommendations for changes and improvements in five major areas. This report presents those recommendations, along with a detailed explanation of the Panel's findings, copies of the reports of the subpanels and consultants, and the results of a survey of NRC non-clerical employees on the issue.

NUREG-1291: BWR AND PWR OFF-NORMAL EVENT DESCRIPTIONS. * Division of Licensee Performance & Quality Evaluation (Post 870411). November 1987. 297pp. 8712230297. 43796:354.

This document chronicles a total of 87 reactor event descriptions for use by operator licensing examiners in the construction of simulator scenarios. Events are organized into four categories: (1) boiling-water reactor abnormal events; (2) boiling-water reactor emergency events; (3) pressurized-water reactor abnormal events; and (4) pressurized-water reactor emergency events. Each event described includes a cover sheet and a progression of operator actions flow chart. The cover sheet contains the following general parameters: initial plant state, sequence initiator, important plant parameters, major plant systems affected, tolerance ranges, final plant state, and competencies tested. The progression of operator actions flow chart depicts, in a flow chart manner, the representative sequence(s) of expected immediate and subsequent candidate actions, including communications, that can be observed during the event. These descriptions are intended to provide examiners with a reliable, performance-based source of information from which to design simulator scenarios that will provide a valid test of the candidates' ability to safely and competently perform all licensed duties and responsibilities.

NUREG-1293 DRFT FC: QUALITY ASSURANCE GUIDANCE FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY.Draft For Comment. PITTIGLIO,C.L. Division of Low Level Waste Management & Decommissioning (Post 870413). November 1987. 13pp. 8712170240. 43753:037.

This document provides guidance to an applicant on development of a low-level waste (LLW) disposal facility in meeting the quality control (QC) requirements of 10 CFR 61.12. The QC requirements are the basis for developing of a quality assurance (QA) program and for the guidance provided herein. The criteria developed for this document are similar to the criteria developed for 10 CFR Part 50 Appendix B. Although Appendix B of 10 CFR Part 50 is not a regulatory requirement for an LLW disposal facility, the criteria that were developed for 10 CFR Part 50 are basic to any QA program. The document specifically establishes QA guidance for the design, construction, and oper-

ation of those structures, systems, components, as well as, for site characterization activities necessary to meet the performance objectives of 10 CFR Part 61 and to limit exposure to or release of radioactivity. This document is issued as a draft and comments are requested. After the comments are reviewed and analyzed, the document will be revised. This draft was reviewed by the NRC staff, an NRC consultant, and personnel from the Idaho National Engineering Laboratory.

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

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

NUREG/CP-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-0078: PROCEEDINGS OF THE SYMPOSIUM ON CHEMICAL PHENOMENA ASSOCIATED WITH RADIOACTIVITY RELEASES DURING SEVERE NUCLEAR PLANT ACCIDENTS. NIEMCZYK,S.J. American Chemical Society. June 1987. 710pp. 8706250118. 41465:079.

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

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/CP-0086 V01: PROCEEDINGS OF THE 19TH DOE/NRC NUCLEAR AIR CLEANING CONFERENCE.Held In Seattle,Washington,August 18-21,1986. FIRST,M.W. Harvard School of Public Health, Boston, MA. May 1987. 644pp. 8706250268. CONF-860820. 41463:073.

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

NUREG/CP-0086 V02: PROCEEDINGS OF THE 19TH DOE/NRC NUCLEAR AIR CLEANING CONFERENCE.Held In Seattle,Washington,August 18-21,1986. FIRST,M.W. Harvard School of Public Health, Boston, MA. May 1987. 630pp. 8706240014. CONF-860820. 41442:037.

See NUREG/CP-0086,V01 abstract.

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

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

NUREG/CP-0088: TRANSACTIONS OF THE 9TH INTERNATIONAL CONFERENCE ON STRUCTURAL MECHANICS IN REACTOR TECHNOLOGY.Panel Session JK: Structural And Mechanical Engineering Research At The U.S. Nuclear Regulatory Commission. BROWZIN,B.S. Division of Engineering (Post 870413). July 1987. 265pp. 8707210681. 41830:271.

These transactions of the JK panel session include preprints of papers which are listed in the Second Announcement for the 9th International Conference on Structural Mechanics in Reactor Technology. These papers represent the body of the JK panel session, "Structural and Mechanical Engineering Research at the U.S. Nuclear Regulatory Commission."

NUREG/CP-0090: TRANSACTIONS OF THE FIFTEENTH WATER REACTOR SAFETY INFORMATION MEETING. WEISS,A.J. Office of Nuclear Regulatory Research, Director (Post 860720). October 1987. 270pp. 8710290263. 43215:350.

This report contains summaries of papers on reactor safety research to be presented at the 15th Water Reactor Safety Information Meeting held at the National Bureau of Standards in Gaithersburg, Maryland, October 26-29, 1987. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, USNRC. Summaries of invited papers concerning nuclear safety issues are included from the Office of Nuclear Regulatory Research, USNRC, in addition to summaries of invited papers that cover the highlights of reactor safety research conducted by the Department of Energy (DOE), the electric utilities through the Electric Power Research Institute (EPRI), the nuclear industry, and the research of government and industry in Europe and Japan. The summaries have been compiled in one report to provide a basis for meaningful discussion and information exchange during the course of the meeting, and are given in the order of their presentation in each session.

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-2000 V06 N2: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of February 1987. * Oak Ridge National Laboratory. March 1987. 123pp. 8704270093. ORNL/NSIC-200. 40678:297.
See NUREG/CR-2000,V05,N12 abstract.

NUREG/CR-2000 V06 N3: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of March 1987. * Oak Ridge National Laboratory. April 1987. 107pp. 8705190343. ORNL/NSIC-200. 40975:324.
See NUREG/CR-2000,V05,N12 abstract.

NUREG/CR-2000 V06 N4: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of April 1987. * Oak Ridge National Laboratory. May 1987. 145pp. 8706230059. ORNL/NSIC-200. 41421:297.
See NUREG/CR-2000,V05,N12 abstract.

NUREG/CR-2000 V06 N5: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of May 1987. * Oak Ridge National Laboratory. June 1987. 108pp. 8707090381. ORNL/NSIC-200. 41633:074.
See NUREG/CR-2000,V05,N12 abstract.

NUREG/CR-2000 V06 N6: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of June 1987. * Oak Ridge National Laboratory. July 1987. 138pp. 8708120301. ORNL/NSIC-200. 42128:253.
See NUREG/CR-2000,V05,N12 abstract.

NUREG/CR-2000 V06 N7: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of July 1987. * Oak Ridge National Laboratory. August 1987. 151pp. 8709150268. ORNL/NSIC-200. 42679:007.
See NUREG/CR-2000,V05,N12 abstract.

NUREG/CR-2000 V06 N8: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of August 1987. * Oak Ridge National Laboratory. September 1987. 120pp. 8710080189. ORNL/NSIC-200. 42987:075.
See NUREG/CR-2000,V05,N12 abstract.

NUREG/CR-2000 V06 N9: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of September 1987. * Oak Ridge National Laboratory. October 1987. 81pp. 8711250072. ORNL/NSIC-200. 43450:016.
See NUREG/CR-2000,V05,N12 abstract.

NUREG/CR-2000 06N10: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of October 1987. * Oak Ridge National Laboratory. November 1987. 74pp. 8712230196. ORNL/NSIC-200. 43796:280.
See NUREG/CR-2000,V05,N12 abstract.

NUREG/CR-2000 V06N11: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of November 1987. * Oak Ridge National Laboratory. December 1987. 86pp. 8801110221. ORNL/NSIC-200. 43975:230.
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. 8704060236. 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-2331 V06 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, July-September 1986. WEISS,A.J. Brookhaven National Laboratory. March 1987. 114pp. 8706120037. BNL-NUREG-51454. 41187:087.
See NUREG/CR-2331,V06,N02 abstract.

NUREG/CR-2331 V06 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October-December 1986. WEISS,A.J. Brookhaven National Laboratory. May 1987. 75pp. 8710080214. BNL-NUREG-51454. 42990:350.

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 and Operations of the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. The projects reported are the following: Advanced Reactor Review, Thermal-Hydraulic Reactor Safety Experiments, Thermodynamic Core-Concrete Interaction Experiments and Analysis, Plant Analyzer, Assessment and Application of TRAC-BF1 Code, Code Maintenance (RAMONA-3B), MELCOR Verification and Benchmarking, Source Term Code Package Verification and Benchmarking, Uncertainty Analysis of the Source Term (QUASAR), Soil-Structure Interaction Evaluation and Structural Benchmarks, Characterization and Detection of Age-Related Failures of Selected Components and Systems With Consideration for Aging/Seismic Interactions, Protective Action Decisionmaking, Risk and Risk Reduction for Zion, and Operational Safety Reliability Research.

NUREG/CR-2331 V07 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, January-March 1987. WEISS,A.J. Brookhaven National Laboratory. July 1987. 75pp. 8711100129. BNL-NUREG-51454. 43338:265.
See NUREG/CR-2331,V06,N04 abstract.

NUREG/CR-2452: RISK METHODOLOGY FOR GEOLOGIC DISPOSAL OF RADIOACTIVE WASTE. Final Report. CRANWELL,R.M.; CAMPBELL,J.E.; HELTON,J.C.; et al. Sandia National Laboratories. August 1987. 105pp. 8710010316. SAND81-2573. 42883:078.

This report contains the description of a risk assessment methodology developed for the US Nuclear Regulatory Commission for use in assessing the risk from the disposal of radioactive wastes in deep geologic formations. This methodology consists of (1) techniques for selecting and screening scenarios, (2) models for use in simulating the physical processes and estimating the consequences associated with the occurrence of these scenarios, (3) probabilistic and statistical techniques for use in risk estimates and sensitivity and uncertainty analysis, and (4) a procedure for utilizing these models and techniques to assess compliance with regulatory standards. The use of this methodology is demonstrated in this report by applying it in the analysis of a hypothetical disposal site containing a bedded-salt formation as the host medium for the waste.

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

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

NUREG/CR-2907 V05: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1984. TICHLER,J.; NORDEN,K.; CONGEMI,J. Brookhaven National Laboratory. August 1987. 256pp. 8708260142. BNL-NUREG-51581. 42332:001.

Releases of radioactive materials in airborne and liquid effluents from commercial light water reactors during 1984 have been compiled and reported. Data on solid waste shipments as well as selected operating information have been included. This report supplements earlier annual reports issued by the former Atomic Energy Commission and the Nuclear Regulatory Commission. The 1984 release data are summarized in tabular form. Data covering specific radionuclides are summarized.

NUREG/CR-3024: SUSTAINED CONCRETE ATTACK BY LOW-TEMPERATURE,FRAGMENTED CORE DEBRIS. TARBELL,W.W.; BRADLEY,D.R.; BLOSE,R.E.; et al. Sandia National Laboratories. July 1987. 247pp. 8709090444. SAND82-2476. 42565:321.

Four experiments were performed to study the interactions between low-temperature core debris and concretes typical of reactor structures. The tests addressed accident situations where the core debris is at elevated temperature, but not molten. Concrete crucibles were formed in right-circular cylinders with 45 kg of steel spheres (3-mm diameter) as the debris simulant. The debris was heated by an inductive power supply to nominal temperatures of 1473 K to 1673 K. Two tests were performed on each of two concrete types using either basalt or limestone aggregate. For each concrete, one test was performed with water atop the debris while the second had no water added. The results show that low-temperature core debris will erode either basalt or limestone-common sand concretes. Downward erosion rates of 3 to 4 cm/hr were recorded for both concrete types. The limestone concrete produced a crust layer within the debris bed that was effective in preventing the downward intrusion of water. The basalt concrete crust was formed above the debris and consisted of numerous, convoluted, thin layers. Carbon dioxide and water release from the decomposition of concrete were partially reduced by the metallic debris to yield carbon monoxide and hydrogen, respectively. The overlying water pool did not effect the reduction reactions.

NUREG/CR-3145 V05: GEOPHYSICAL INVESTIGATIONS OF THE WESTERN OHIO-INDIANA REGION.Final Report,October 1981 - September 1986. CHRISTENSEN,D.; POLLACK,H.N.; LAY,T.; et al. Michigan, Univ. of, Ann Arbor, MI. March 1987. 333pp. 8707310041. 41997:018.

Earthquake activity in the Western Ohio-Indiana region was monitored with a precision seismograph network consisting of nine stations located in west-central Ohio and four stations located in Indiana. Five local and near-regional earthquakes were recorded during the fiscal 1985-86 report period, ranging in magnitude from 0.5 to 5.0m(b). The two largest events (January 31, 1986 near Cleveland, Ohio, 5.0m(b); and July 12, 1986 near St. Marys, Ohio, 4.5m(b)) were felt with minor damage reported in each case. Focal mechanisms and isoseismal maps for these events are included in this report. These events are the largest to have occurred in Ohio since the events of March 2 and March 9, 1937 (magnitude = 4.5 and 4.9, respectively). The remaining three earthquakes all occurred in Ohio, north of the array. A total of 42 local and near-regional events, eleven of which were felt, have now been recorded by the Ohio-Indiana array since its initiation in 1976. Teleseismic P-wave arrival and residual tables were updated to include newly acquired data. The results are similar to those in previous years. The local velocity structure was investigated using data acquired during a refraction experiment in the summer of 1984, and travel times of local and near-regional earthquakes.

NUREG/CR-3228 V05: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1986. LOSS,F.J. Materials Engineering Associates, Inc. July 1987. 237pp. 8708110406. MEA-2207. 42109:337.

This program is being conducted for the NRC to provide analytical and experimental methods and data that are necessary to ensure the structural safety and reliability of pressure boundary components in U.S. commercial, light water reactor power systems. Emphasis is placed on characterization of material properties performance in a nuclear environment for the application to plant-life extension and mitigation of the consequences of postulated accident scenarios. Current work is organized into three major tasks: (1) fracture mechanics investigations, (2) environmentally-assisted crack growth in high temperature, primary reactor water, and (3) radiation sensitivity and postirradiation properties recovery. Research progress in these three tasks for 1986 is summarized in this report.

NUREG/CR-3231: PIPE-TO-PIPE IMPACT PROGRAM. ALZHEIMER,J.M.; BAMPTON,M.C.; FRILEY,J.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1987. 100pp. 8706240152. PNL-5779. 41452:059.

The objective of this research was to determine the extent of damage that occurs when two pipes experience an impact event due to one whipping against the other. The research was conducted through experimental and analytical approaches. The former required the development of a specialized impact machine that could accelerate a whipping pipe with sufficient energy to cause failure of a target pipe that was heated and pressurized to Pressurized Water Reactor (PWR) conditions. Damage was measured in terms of crushing, bending, and failure. The results of the tests permitted the correlation between

pipes of a certain size and the damage they could cause when impacting with a certain amount of known energy. These results were used to evaluate the pipe whip criteria in the Standard Review Plan 3.6.2-4. It was established that the criteria conditions did not fully represent the results obtained experimentally. An analysis procedure to model the pipe whip event was developed and used to establish the test matrix for the experimental program. This analytical procedure can also be used to predict deformation and rupture for postulated pipe whip scenarios.

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

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

NUREG/CR-3320 V03: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM,PSF Physics-Dosimetry Program. MCELROY,W.N.; GOLD,R. Hanford Engineering Development Laboratory. October 1987. 323pp. 8710230376. HEDL-TME 87-3. 43145:273.

The metallurgical irradiation experiment at the Oak Ridge Research Reactor Poolside Facility (ORR-PSF) is one of the series of benchmark experiments in the framework of the Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV- SDIP). The goal of this program is to test, against well-established benchmarks, the methodologies and data bases that are used to predict the irradiation embrittlement and fracture toughness of pressure vessel and support structure steels. The prediction methodology includes procedures for neutron physics calculations, dosimetry and spectrum adjustment methods, metallurgical tests, and damage correlations. The benchmark experiments serve to validate, improve, and standardize these procedures. The results of this program are implemented in a set of ASTM Standards on pressure vessel surveillance procedures. These, in turn, may be used as guides for the nuclear industry and for the Nuclear Regulatory Commission (NRC). To serve as a benchmark, a very careful characterization of the ORR-PSF experiment is necessary, both in terms of neutron flux-fluence spectra and of metallurgical tests results. Statistically determined uncertainties must be given in terms of variances and covariances to make comparisons between predictions and experimental results meaningful. Detailed descriptions of the PSF physics-dosimetry program and its results are reported.

NUREG/CR-3320 V04: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM,PSF Metallurgy Program. MCELROY,W.N.; GOLD,R. Hanford Engineering Development Laboratory. November 1987. 366pp. 8712240083. HEDL-TME 87-4. 43788:072.

The metallurgical irradiation experiment at the Oak Ridge Research Reactor Poolside Facility (ORR-PSF) is one of the series of benchmark experiments in the framework of the Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV- SDIP). The goal of this program is to test, against well-established benchmarks, the methodologies and data bases that are used to predict the irradiation embrittlement and fracture toughness of pressure vessel and support structure steels. The prediction methodology includes procedures for neutron physics calculations, dosimetry and spectrum adjustment methods, metallurgical tests, and damage correlations. The benchmark experiments serve to validate, improve, and standardize these procedures. The results of this program are implemented in a set of ASTM Standards on pressure vessel surveillance procedures. These, in turn, may be used as guides for the nuclear industry and for the Nuclear Regulatory Commission (NRC). To serve as a benchmark, a very careful characterization of the ORR-PSF experiment is necessary, both in terms of neutron flux-fluence spectra and of metallurgical tests results. Statistically determined uncertainties must be given in terms of variances and covariances to make comparisons between predictions and experimental results meaningful. Detailed descriptions of the PSF metallurgy program and its results are reported.

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-349[illegible]: [illegible]ON OF FAILURE IMPORTANCE MEASURES FOR BASIC EVENTS AND PLANT SYSTEMS IN NUCLEAR POWER PLANTS. WHEELER,T.A.; SPULAK,R.G. Sandia National Laboratories. December 1987. 450pp. 8801070265. SAND83-0329. 43941:292.

This is a report on the project "Determination of Event Sequences Contributing to Overall Frequency of Core Melt." It is an analysis of 13 existing probabilistic risk assessment (PRAs) covering 15 power plants. The object of this project is to identify the plant systems and system components which contribute most to the overall core melt frequency and to some measure of risk. The results of this analysis are grouped into two major types of commercial reactors, boiling water reactors (BWRs), and pressurized water reactors (PWRs). Within each type of reactor, plants are grouped by manufacturer. The results are first analyzed on a plant-by-plant basis. Then, the results are discussed for all plants within each plant manufacturer type to study any interrelationships or common trends which might exist among plants built by the same manufacturer. The results are then discussed among all PWRs and all BWRs.

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. EMI-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 likeli-

...nd that the concentrating mechanisms will operate and to ...e upper limits of concentration for each mechanism.

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

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

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

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

NUREG/CR-3966: METHODS FOR IMPACT ANALYSIS OF SHIPPING CONTAINERS. NELSON,T.A.; CHUN,R.C. Lawrence Livermore National Laboratory. November 1987. 60pp. 8712010381. UCID-20639. 43487:123.

This report reviews methods for performing impact stress analyses of shipping containers used to transport spent fuel. Three methods are discussed: quasi-static, dynamic lumped parameter and dynamic finite element. These methods are used by industry for performing impact analyses for Safety Analysis Reports. The approach for each method is described including assumptions and limitations and modeling considerations. The effects of uncertainties in the modeling and analyzing of casks are identified. Each of the methods uses linear elastic structural analysis principles. Methods for interfacing impact stresses with the design and load combinations criteria specified in Regulatory Guides 7.6 and 7.8 are outlined. The quasi- static method is based on D'Alembert's principle to substitute equivalent static forces for inertial forces created by the impact. The lumped parameter method is based on using a discrete number of stiffness elements and masses to represent the cask during impact. The dynamic finite element method uses finite element techniques combined with time integration to analyze the cask impact. Each of these methods can provide an acceptable means, within certain limitations, for analyzing cask impact on unyielding surfaces.

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. ...02190449. ANL-AA-30. 39690:235.

Seasonal replacement energy costs are estimated for potential short-term shutdowns of 116 nuclear electricity-generating units. These estimates ... 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 under-

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

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

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

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

NUREG/CR-4134 R01: REPOSITORY ENVIRONMENTAL PARAMETERS AND MODELS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES (BASALT,TUFF AND SALT). CLAIBORNE,H.C.; CROFF,A.G.; GRIESS,J.C.; et al. Oak Ridge National Laboratory. October 1987. 247pp. 8711060223. ORNL/TM-9522. 43309:091.

This document provides specifications for models/methodologies that could be employed in determining postclosure repository environmental parameters relevant to the performance of high-level waste packages for the Basalt Waste Isolation Project (BWIP) at Richland, Washington, the tuff at Yucca Mountain by the Nevada Test Site, and the bedded salt in Deaf Smith County, Texas. Guidance is provided on (1) the identity of the relevant repository environmental parameters (groundwater characteristics, temperature, radiation, and pressure), (2) the models/methodologies employed to determine the parameters, and (3) the input data base for the models/methodologies. Supporting studies included are (1) an analysis of potential waste package failure modes leading to identification of the relevant repository environmental parameters, (2) an evaluation of the credible range of the repository environmental parameters, and (3) a summary of the review of existing models/methodologies currently employed in determining repository environmental parameters relevant to waste package performance.

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

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

NUREG/CR-4161 V03: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 3:Salt. DIDWALL,E.M. Lawrence Livermore National Laboratory. WOLLENBERG,H.A.; BINNALL,E.P.; et al. Lawrence Berkeley Laboratory. July 1987. 41pp. 8708120128. UCID-20092. 42128:057.

This is the third report of a three-volume series addressing critical parameters for an underground high-level nuclear waste repository in basalt, tuff, or salt. This report covers the identification and prioritization of critical parameters for a repository in bedded salt. For purposes of this work, a parameter is a physical property whose value helps determine the characteristics or behavior of a repository system. A parameter is considered to be critical if a mistake in its measurement, or in the inability to measure it, could lead to a wrong conclusion about the adequacy of a repository. Consideration was given to the relative im-

portance of critical parameters in four specific discipline areas: geomechanics, geology, hydrology, and geochemistry. Relative importance of the parameters was independently considered for each of the four major phases of repository activity: site characterization, site construction, site operation, and site closure and decommissioning. Important activities that cover all phases are determination of hydrologic characteristics and salt dissolution rates, and their long- term monitoring.

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

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

NUREG/CR-4216: EXPERIMENTAL RESULTS FOR A 1:8-SCALE STEEL MODEL NUCLEAR POWER PLANT CONTAINMENT PRESSURIZED TO FAILURE. KOENIG,L.N. Sandia National Laboratories. December 1986. 634pp. 8710010431. SAND85-0790. 42878:040.

A 1:8-scale steel model of a light-water reactor containment building was tested as part of an NRC-sponsored program whose objective is to qualify methods for predicting the response of containment buildings subjected to severe accidents. The model was pressurized in steps to 195 psig, and its response was recorded by strain and displacement transducers located throughout the model. Periodic leak-rate measurements were also made. An overview of the test is given, and the response of the model and specific features (such as equipment hatches, airlocks, and piping penetrations) are presented in this report.

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

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

NUREG/CR-4219 V04 N1: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For October 1986 - March 1987. PUGH,C.E. Oak Ridge Nat[illegible]al Laboratory. August 1987. 266pp. 8710060244. ORNL/TM[illegible] 42922:293.

The Heavy-Section Steel Technology ([illegible]ram is an engineering research activity conducted by the Oak Ridge National Laboratory for the Nuclear Regulatory Commission. The Program comprises studies related to all areas of the technology of materials fabricated into thick-section primary-coolant containment systems of light-water-cooled nuclear power reactors. The investigation focuses on the behavior and structural integrity of steel pressure vessels containing cracklike flaws. Current work is organized into twelve tasks: (1) program management, (2) fracture-methodology and analysis, (3) material characterization and properties, (4) environmentally assisted crack-growth studies, (5) crack-arrest technology, (6) irradiation effects studies, (7) cladding evaluations, (8) intermediate vessel tests and analysis, (9) thermal-shock technology, (10) pressurized thermal-shock technology, (11) Pressure Vessel Research Users' Facility, and (12) shipping-cask material evaluations.

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

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

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

The Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) has been established by the U.S. Nuclear Regulatory Commission (NRC) to improve, test, verify, and standardize the physics-dosimetry-metallurgy, damage correlation, and associated reactor analysis methods, procedures and data used to predict the integrated

effect of neutron exposure to LWR pressure vessels and their support structures. A vigorous research effort attacking the same measurement and analysis problems exists worldwide, and strong cooperative links between the U.S. NRC-supported activities at HEDL, ORNL, NBS, and MEA and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Julich, Germany), and several United Kingdom Laboratories have been extended to a number of other countries and laboratories. These cooperative links are strengthened by the active membership of the scientific staff from many participating countries and laboratories in the ASTM E10 Committee on Nuclear Technology and Applications. Several subcommittees of ASTM E10 are responsible for the preparation of LWR surveillance standards. Results of FY-86 research by a number of LWR-PV-SDIP participants are reported in this progress report.

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

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

NUREG/CR-4405: PROBABILISTIC RISK ASSESSMENT (PRA) INSIGHTS. FITZPATRICK,R.; ARRIETA,L.; TEICHMANN,T.; et al. Brookhaven National Laboratory. December 1987. 142pp. 8801070031. BNL-NUREG-51931. 43944:018.

Four different probabilistic risk assessments (PRAs) have been briefly reviewed with the broad objective of ascertaining what insights might be gained (beyond those already documented in the PRAs) by an independent evaluation. This effort was not intended to verify the specific details and results of each PRA but rather, having accepted the results, to see what they might mean on a plant-specific and/or generic level. The four PRAs evaluated were those for Millstone 3, Seabrook, Shoreham, and Oconee 3. Full detailed reviews of each of these four PRAs have been commissioned by the NRC, but only two have been completed and available as further input to this study: the review of Millstone 3 by LLNL and the review of Shoreham by BNL. The review reported here focused on identifying the dominant (leading) initiators, failure modes, plant systems, and specific components that affect the overall core melt probability and/or risk to the public. In addition, the various elements of the methodologies employed by the four PRAs are discussed and ranked (per NUREG/CR-3852). PRA-specific insights are presented within the report section addressing that PRA, and overall insights are presented in the Summary.

NUREG/CR-4407: PIPE BREAK FREQUENCY ESTIMATION FOR NUCLEAR POWER PLANTS. WRIGHT,R.E.; STEVERSON,J.A.; ZUROFF,W.F. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1987. 255pp. 8708040304. EGG-2421. 42037:260.

This study empirically develops frequencies of safety-significant pipe failures in commercial nuclear power plants (NPPs). Its primary purpose is to update the pipe break frequencies reported in the Reactor Safety Study, WASH-1400, which are used in many risk analyses. The study involved reviewing various data sources for actual piping failure events of significant magnitude. When extant in the documentation reviewed, information was extracted concerning conditional factors such as the system in which the failure occurred, operational mode of the plant, and size of the pipe involved to estimate conditional pipe break frequencies useful to risk analysts. Because of the high quality piping used in NPPs, there have been few significant pipe failures. An attempt was made to augment the analysis with synthetic data from a Delphi approach, but the wide uncertainty bounds on the resulting estimates rendered the results unsuitable for combining data.

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. 234pp. 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-4418: DOSE CALCULATION FOR CONTAMINATION OF THE SKIN USING THE COMPUTER CODE VARSKIN. TRAUB,R.J.; REECE,W.D.; SCHERPELZ,R.I.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1987. 101pp. 8708280132. PNL-5610. 42399:289.

This report describes a method for computing the radiation dose to the skin from radioactive contamination on the skin. The calculational method is based on the tables of absorbed energy distributions around point sources in water that have been developed by M.J. Berger and published in 1971 in Pamphlet No. 7 of the "Journal of Nuclear Medicine." The method

has been implemented in a computer code, VARSKIN, that will compute the radiation dose at a specified depth in the skin from a radiation source ranging in size from a point to a disc having a 100-cm(2) area. This report contains a source code listing of the computer code and tables of dose factors that have been derived from the code, as well as sample runs and user instructions. This report also discusses the contamination level that corresponds to the recording level described by the International Commission on Radiological Protection.

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. 781pp. 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-4457: AGING OF CLASS 1E BATTERIES IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS. EDSON,J.L.; HARDIN,J.E. EG&G Idaho, Inc. (subs. of EG&G, Inc.). July 1987. 43pp. 8709040461. EGG-2488. 42522:019.

This report presents the results of a study of aging effects on safety-related batteries in nuclear power plants. The purpose is to evaluate the aging effects caused by operation within a nuclear facility and to evaluate maintenance, testing, and monitoring practices with respect to their effectiveness in detecting and mitigating the effects of aging. The study follows the U.S. Nuclear Regulatory Commission's (NRC's) Nuclear Plant-Aging Research approach and investigates the materials used in battery construction, identifies stressors and aging mechanisms, presents operating and testing experience with aging effects, analyzes battery-failure events reported in various data bases, and evaluates recommended maintenance practices. Data bases that were analyzed included the NRC's Licensee Event Report system, the Institute for Nuclear Power Operations' Nuclear Plant Reliability Data System, the Oak Ridge National Laboratory's In-Plant Reliability Data System, and The S.M. Stoller Corporation's Nuclear Power Experience data base.

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. 994pp. 8704060155. SAND85-2496. 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 analyses 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. 90pp. 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-4469 V05: NONDESTRUCTIVE EXAMINATION (NDE) RELIABILITY FOR INSERVICE INSPECTION OF LIGHT WATER REACTORS.Semiannual Report, April-September 1986. DOCTOR,S.R.; BATES,D.J.; DEFFENBAUGH,J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1987. 39pp. 8705280412. PNL-5711. 41097:183.

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

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

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

NUREG/CR-4485 V02: THE IMPACT OF FUEL CLADDING FAILURE EVENTS ON OCCUPATIONAL RADIATION EXPOSURES AT NUCLEAR POWER PLANTS.Case Study:PWR During An Outage. MOELLER,M.P.; MARTIN,G.F.; KENOYER,J.L. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1987. 145pp. 8710080227. PNL-5606. 42993:327.

This report is the second in a series of case studies designed to evaluate the magnitude of increase in occupational radiation exposures at commercial U.S. nuclear power plants resulting from small incidents or abnormal events. The event evaluated is fuel cladding failure, which can result in elevated primary coolant activity and increased radiation exposure rates within a plant. For this case study, radiation measurements were made at a pressurized-water reactor (PWR) during a maintenance and refueling outage. The PWR had been operating for 22 months with fuel cladding failure characterized as 105 pin-hole leakers, the equivalent of 0.212 percent failed fuel. Gamma spectroscopy measurements, radiation exposure rate determination, thermoluminescent dosimeter (TLD) assessments, and air sample analyses were made in the plant's radwaste, pipe penetration, and containment buildings. Based on the data collected, evaluations indicate that the relative contributions of activation products and fission products to the total exposure rates were constant over the duration of the outage. This constancy is due to the significant contribution from the longer-lived isotopes of cesium (a fission product) and cobalt (an activation product). For this reason, fuel cladding failure events remain as significant to occupational radiation exposure during an outage as during routine operations. As documented in the previous case study (NUREG/CR-4485 Vol. 1), fuel cladding failure events increased radiation exposure rates an estimated 540 percent in some areas of the plant during routine operations. Consequently, such events can result in significantly greater radiation exposure rates in many areas of the plant during the maintenance and refueling outages than would have been present under normal fuel conditions.

NUREG/CR-4491: DEVELOPMENT OF MODELS FOR WARM PRESTRESSING. * Materials Engineering Associates, Inc. RYBICKI,E.F. PROSIG, Inc. STONESIFER,R.B. Computational Mechanics. 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) (Atlun) 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-4527 V01: AN EXPERIMENTAL INVESTIGATION OF INTERNALLY IGNITIED FIRES IN NUCLEAR POWER PLANT CONTROL CABINETS.Part 1:Cabinet Effects Tests. CHAVEZ,J.M. Sandia National Laboratories. April 1987. 115pp. 8707060099. SAND86-0336. 41583:285.

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

does not appear that the fire poses a threat outside the burning cabinet ex.ept the resulting smoke.

NUREG/CR-4530 V02: U.S./FRENCH JOINT RESEARCH PROGRAM REGARDING THE BEHAVIOR OF POLYMER BASE MATERIALS SUBJECTED TO BETA RADIATION.Volume 2: Phase-2a Screening Tests. BUCKALEW,W.H.; WYANT,F.J.; CHENION,J.; et al. Sandia National Laboratories. September 1987. 59pp. 8712280236. SAND86-0366. 43815:242.

As part of the ongoing joint NRC/CEA cooperative test program to investigate the relative effectiveness of beta and gamma irradiation to produce damage in polymer base materials, ethylene propylene rubber (EPR) specimens, in slab geometry, were exposed to Cobalt-60 gamma rays and accelerator produced electron beams. Specimens were irradiated and evaluated at research facilities in the U.S. (Sandia National Laboratories) and France (Compagnie OR15 Industrie). These tests included several electron beam energies, sample thicknesses, exposure doses, and dose rates. Based on changes in the tensile properties of the test specimens, results of these studies suggest that material damage resulting from electron and gamma irradiations can be correlated on the basis of absorbed radiation dose for several EPR formulations in slab geometry.

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-4534: ANALYSIS OF DIFFUSION FLAME TESTS. SHEPHARD,J.E. Sandia National Laboratories. August 1987. 89pp. 8710010109. SAND86-0419. 42881:333.

This report discusses the results and analysis of hydrogen diffusion flame tests conducted at the Nevada Test Site by EPRI and the USNRC. Those tests were designed to simulate the effects of hydrogen combustion inside a nuclear power plant containment following a degraded-core accident. Test initial conditions and sample data plots are given for 16 tests. Mixing and ignition phenomena are discussed in terms of the source parameters and igniter location. A simple model is developed for simulating the heat transfer and computing convective heat transfer coefficients for experimental pressure measurements. Convective heat transfer coefficients are reported for four tests. The effect of stagnation point heat transfer is estimated.

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. 191pp. 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 V01: ANALYSIS OF CORE DAMAGE FREQUENCY FROM INTERNAL EVENTS: METHODOLOGY GUIDELINES. HARPER,F.T.; CAMP,A.L. Sandia National Laboratories. DROUIN,M.T. Science Applications International Corp. (formerly Science Applications, Inc.). September 1987. 292pp. 8711250143. SAND86-2084. 43450:190.

NUREG-1150 examines the risk to the public from a selected group of nuclear power plants. This report describes the methodology used to estimate the internal event core damage frequencies of four plants in support of NUREG-1150. In principle, this methodology is similar to methods used in past probabilistic risk assessments; however, based on past studies and using analysts that are experienced in these techniques, the analyses can be focused in certain areas. In this approach, only the most important systems and failure modes are modeled in detail. Further, the data and human reliability analyses are simplified, with emphasis on the most important components and human actions. Using these methods, an analysis can be completed in six to nine months using two to three full-time systems analysts and part-time personnel in other areas, such as data analysis and human reliability analysis. This is significantly faster and less costly than previous analyses and provides most of the insights that are obtained by the more costly studies.

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. 466pp. 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 fail-

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

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

NUREG/CR-4550 V06PT1: ANALYSIS OF CORE DAMAGE FREQUENCY FROM INTERNAL EVENTS:GRAND GULF,UNIT 1.Main Report. DROUIN,M.T.; LACHANCE,J.L.; SHAPIRO,B.J.; et al. Sandia National Laboratories. April 1987. 514pp. 8707080284. SAND86-2084. 41618:163.

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

NUREG/CR-4550 V06PT2: ANALYSIS OF CORE DAMAGE FREQUENCY FROM INTERNAL EVENTS:GRAND GULF,UNIT 1.Appendices. DROUIN,M.T.; LACHANCE,J.L.; SHAPIRO,B.J.; et al. Sandia National Laboratories. April 1987. 653pp. 8707080295. SAND86-2084. 41619:317.

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

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

The Severe Accident Risk Reduction Program (SARRP) has completed a rebaselining of the risks to the public from a particular pressurized water reactor with an ice-condenser containment (Sequoyah, Unit 1). Emphasis was placed on determining the magnitude and character of the uncertainties, rather than focusing on a point estimate. The risk-reduction potential of a set of proposed safety option backfits was also studied, and their costs and benefits were also evaluated. It was found that the risks from internal events are generally comparable to those evaluated in the Reactor Safety Study for a different pressurized water reactor, although the calculated uncertainty bands indicate that the risk could be higher or lower by as much as an order of magnitude. Principle sources of uncertainty include the modeling of common-cause failure in the component cooling water system, the characteristics of hydrogen generation and

burning, and the possibility of fission product releases that bypass the ice condenser. Most of the postulated safety options do not appear to be cost effective for the Sequoyah plant; however, certain relatively inexpensive hardware and procedural changes to prevent core damage appear to be marginally cost effective. It should be noted that this work is based on a draft report of the ASEP study of core-damage frequency for Sequoyah. The final ASEP results have lower frequencies for station-blackout sequences. These final ASEP results will be incorporated in the final version of this study. The differences are not expected to cause significant changes in the conclusions of the report because risk is influenced by a variety of accident sequences, not just station blackout. This work supports the Nuclear Regulatory Commission's assessment of severe accidents in NUREG-1150.

NUREG/CR-4551 V3 PT1: EVALUATION OF SEVERE ACCIDENT RISKS AND THE POTENTIAL FOR RISK REDUCTION:PEACH BOTTOM,UNIT 2.Main Report.Draft For Comment. AMOS,C.N.; BENJAMIN,A.S.; BOYD,G.J.; et al. Sandia National Labora' es. April 1987. 244pp. 8707060116. SAND86-1309. 41593:078.

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

NUREG/CR-4551 V3 PT2: EVALUATION OF SEVERE ACCIDENT RISKS AND THE POTENTIAL FOR RISK REDUCTION:PEACH BOTTOM,UNIT 2.Appendices.Draft For Comment. AMOS,C.N.; BENJAMIN,A.S.; BOYD,G.J.; et al. Sandia National Laboratories. May 1987. 531pp. 8707060135. SAND86-1309. 41591:267.

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

NUREG/CR-4551 V4 DRF: EVALUATION OF SEVERE ACCIDENT RISKS AND THE POTENTIAL FOR RISK REDUCTION:GRAND GULF,UNIT 1.Draft For Comment. AMOS,C.N.; BENJAMIN,A.S.; BOYD,G.J.; et al. Sandia National Laboratories. April 1987. 761pp. 8707060161. SAND86-1309. 41589:226.

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

NUREG/CR-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-4577: AUTOMATED LONG-TERM SURVEILLANCE OF A COMMERCIAL NUCLEAR POWER PLANT. SMITH,C.M.; GONZALEZ,R.C. Oak Ridge National Laboratory. August 1987. 66pp. 8710080198. ORNL/TM-10015. 42995:290.

This report presents and describes a pattern recognition system for monitoring nuclear reactor signals. The system is based on detecting deviations from baseline signatures identified during normal plant operation. The capabilities and limitations of this pattern recognition approach were investigated during a 2 1/2-year series of continuous online experiments at the Sequoyah-1 nuclear power plant.

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

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

NUREG/CR-4583 V03: DEVELOPMENT AND VALIDATION OF A REAL-TIME SAFT-UT SYSTEM FOR THE INSPECTION OF LIGHT WATER REACTOR COMPONENTS.Annual Report,October 1985 - September 1986. DOCTOR,S.R.; HALL,T.E.; REID,L.D.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1987. 73pp. 8709210489. PNL-5822. 42738:004.

The Pacific Northwest Laboratory is working to design, fabricate, and evaluate a real-time flaw detection and characterization system based on the synthetic aperture focusing technique for ultrasonic testing (SAFT-UT). The system is designed to perform inservice inspection of light-water reactor components. Included objectives of this program for the Nuclear Regulatory Commission are to develop procedures for system calibration and field operation, to validate the system through laboratory and field inspections, and to generate an engineering data base to support ASME Code acceptance of the technology. This progress report covers the programmatic work from October 1985 through September 1986.

NUREG/CR-4590 V01: AGING OF NUCLEAR STATION DIESEL GENERATORS.Evaluation Of Operating And Expert Experience.Phase I Study. HOOPINGARNER,K.; VAUSE,J.W.; DINGEE,D.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1987. 300pp. 8709150302. PNL-5832. 42679:158.

Pacific Northwest Laboratory evaluated operational and expert experience pertaining to the aging degradation of diesel generators in nuclear service. The research, sponsored by the U.S. Nuclear Regulatory Commission, identified and characterized the contribution of aging to emergency diesel generator failures. Volume 1 reviews diesel generator experience to identify the systems and components most subject to aging degradation and isolates the major causes of failure that may affect future operational readiness. Evaluations show that as plants age, the percent of aging-related failures increases and failure modes change. A compilation is presented of recommended corrective actions for the failures identified. A review of current relevant industry programs, research, and standards is included. Volume 2 reports results of an industry-wide workshop.

NUREG/CR-4590 V02: AGING OF NUCLEAR STATION DIESEL GENERATORS.Evaluation Of Operating And Expert Experience.Workshop. HOOPINGARNER,K.; VAUSE,J.W. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1987. 89pp. 8709150331. PNL-5832. 42678:301.

Pacific Northwest Laboratory evaluated operational and expert experience pertaining to the aging degradation of diesel generators in nuclear service. The research, sponsored by the U.S. Nuclear Regulatory Commission, identified and characterized the contribution of aging to emergency diesel generator failures. This report, Volume 2, reports the results of an industry-wide workshop held on May 28 and 29, 1986 to discuss the technical issues associated with aging of nuclear service emergency diesel generators. The technical issues discussed most extensively were: man/machine interfaces, component interfaces, thermal gradients of startup and cooldown and the need for an accurate industry database for trend analysis of the diesel generator system. Volume 1 reports the results of the Phase I research.

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

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

NUREG/CR-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-4617: ONSITE ASSESSMENTS OF THE EFFECTIVENESS AND IMPACTS OF UPGRADED EMERGENCY OPERATING PROCEDURES. MEYER,O.R.; BLACKMAN,H.S.; FORD,R.E.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). March 1987. 167pp. 8706220014. EGG-2456. 41415:226.

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

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

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

NUREG/CR-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-4640: HANDBOOK OF SOFTWARE QUALITY ASSURANCE TECHNIQUES APPLICABLE TO THE NUCLEAR INDUSTRY. BRYANT,J.L.; WILBURN,N.P. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1987. 113pp. 8709090601. PNL-5784. 42594:001.

Pacific Northwest Laboratory is conducting a research project to recommend good engineering practices in the application of 10 CFR 50, Appendix B requirements to assure quality in the development and use of computer software for the design and operation of nuclear power plants for NRC and industry. This handbook defines the content of a software quality assurance program by enumerating the techniques applicable. Definitions, descriptions, and references where further information may be obtained are provided for each topic.

NUREG/CR-4645: TOUGH USER'S GUIDE. PRUESS,K. Sandia National Laboratories. * Lawrence Berkeley Laboratory. August 1987. 348pp. 8710300024. SAND86-7104. 43225:075.

This document contains a technical description of the TOUGH computer program, which was developed at Lawrence Berkeley Laboratory for simulating the coupled transport of water, vapor, air and heat in porous and fractured media. The physical processes taken into account in TOUGH are discussed, and the governing equations actually solved by the simulator are stated in full detail. A brief overview is given of the mathematical and numerical methods, and the code architecture. The report provides detailed instructions for preparing input decks. Code applications are illustrated by means of six sample problems.

NUREG/CR-4651: DEVELOPMENT OF RIPRAP DESIGN CRITERIA BY RIPRAP TESTING IN FLUMES:Phase I. LEE,D.W.; HINKLE,N.E. Oak Ridge National Laboratory. ABT,S.R.; et al. Colorado State Univ., Fort Collins, CO. May 1987. 120pp. 8705280379. ORNL/TM-10100. 41100:230.

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

NUREG/CR-4653: GASPAR II - TECHNICAL REFERENCE AND USER GUIDE. STRENGE,D.L.; BANDER,T.J.; SOLDAT,J.K. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1987. 615pp. 8707090389. PNL-5907. 41631:179.

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

NUREG/CR-4654: RADIONUCLIDE TRANSPORT AS VAPOR THROUGH UNSATURATED FRACTURED ROCK. GREEN,R.T.; EVANS,D.D. Arizona, Univ. of, Tucson, AZ. July 1987. 199pp. 8708040263. 42036:028.

The objective of this study is to identify and examine potential mechanisms of radionuclide transport as vapor at a high-level radioactive waste repository located in unsaturated fractured rock. Transport mechanisms and processes have been investigated near the repository and at larger distances. Transport mechanisms potentially important at larger distances include or-

dinary diffusion, viscous flow and free convection. Ordinary diffusion includes self and binary diffusion, Knudsen flow and surface diffusion. Pressure flow and slip flow comprise viscous flow. Free convective flow results from a gas density contrast. Transport mechanisms or processes dominant near the repository include ordinary diffusion, viscous flow plus several mechanisms whose driving forces arise from the non-isothermal, radioactive nature of high-level waste. The additional mechanisms include forced diffusion, aerosol transport, thermal diffusion and thermophoresis. Near a repository vapor transport mechanisms and processes can provide a significant means of transport from a failed canister to the geologic medium from which other processes can transport radionuclides to the accessible environment. These issues are believed to be important factors that must be addressed in the assessment of specific engineering designs and site selection of any proposed HLW repository.

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

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

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

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

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

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

NUREG/CR-4667 V03: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April-September 1986. SHACK,W.J.; KASSNER,T.F.; MAIYA,P.S.; et al. Argonne National Laboratory. October 1987. 89pp. 8801070270. ANL-87-37. 43944:351.

This progress report summarizes work performed by Argonne National Laboratory on environmentally assisted cracking in light water reactors during the six months from April 1986 through September 1986.

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

Forty-eight operational events, reported in Licensee Event Reports (LERs) and occurring at commercial light-water reac-

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

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

See NUREG/CR-4674,V03 abstract.

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

This report presents the findings of a task in which currently available fire research literature was reviewed for quantitative data on the burning characteristics of combustible materials that are found in nuclear power plants. The materials considered for which quantitative data were available include cable insulation materials, flammable liquids, furniture, trash and general refuse, and wood and wood products. A total of 90 figures and tables, taken primarily from the referenced works, which summarize the available quantitative fire characterization information for these materials is presented.

NUREG/CR-4681: ENCLOSURE ENVIRONMENT CHARACTERIZATION TESTING FOR THE BASE LINE VALIDATION OF COMPUTER FIRE SIMULATION CODES. NOWLEN,S.P. Sandia National Laboratories. March 1987. 82pp. 8706150161. SAND86-1296. 41301:047.

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

NUREG/CR-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. 8702[illegible]116. 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 applications. 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-4690 V01 N1: GENERIC COMMUNICATIONS INDEX.Listings Of Communications (1971-1986). DEAN,R.S.; STEINBRECHER,D.; HENNICK,A. Parameter, Inc. December 1987. 320pp. 8712310192. 43874:27[illegible].

As part of its program to feed back information on operating experience to industry, the U.S. Nuclear Regulatory Commission (NRC) issues certain generic communications called bulletins (about 10/yr), information notices (about 100/yr), and circulars (now discontinued). The report presents an index of all such communications from 1971, when such documentation started, to 1986. The index is the printout from the computer database developed at the NRC for easy access to information previously published in these generic communications. The record (row) entries each consist of twenty fields (columns) of categorized information about a particular document. Included are fields for the document identification number, title, and NRC technical contact, and fields for general system or topic, specific component or topic, cause or defect, potential affect, remarks, and vendors involved. The report also contains lists of communications sorted by each subject category of the General System or Topic and the Specific Component or Topic fields. A list of the category subjects used and a list of vendors involved in the communications are included, also. The NRC intends to periodically update this report to index later communications.

NUREG/CR-4690 V02: GENERIC COMMUNICATIONS INDEX.User's Manual. DEAN,R.S.; STEINBRECHER,D.; HENNICK,A.; et al. Parameter, Inc. December 1987. 20pp. 8712310211. 43874:253.

This report is a manual for providing information required to use a special computer program developed by the NRC for indexing generic communications. The program is written in a

user-friendly menu driven form using dBASE III programming language. It facilitates use of the required dBASE III search and sort capabilities to access records in a database called Generic Communications Index. This index is made up of one record each for all bulletins, circulars, and information notices, including revisions and supplements, from 1971, when such documentation started, through 1986 (or to the latest update). The program is designed for use by anyone modestly acquainted with the general use of IBM-compatible personal computers. The manual contains both a brief overview and a detailed description of the program, as well as detailed instructions for getting started using the program on a personal computer with either a two-floppy disk or a hard disk system. Included at the end are a brief description of how to handle problems which might occur, and notes on the makeup of the program and database files for help in adding records of communications for future years.

NUREG/CR-4692: OPERATING EXPERIENCE REVIEW OF FAILURES OF POWER OPERATED RELIEF VALVES AND BLOCK VALVES IN NUCLEAR POWER PLANTS. MURPHY,G.A. Oak Ridge National Laboratory. CLETCHER,J.W. Professional Analysis, Inc. October 1987. 82pp. 8712080440. ORNL/NOAC-233. 43588:214.

This report contains a review of nuclear power plant operating events involving failures of power-operated relief valves (PORVs) and associated block valves (BVs). Of the 230 events identified, 101 involved PORV mechanical failure, 91 were attributable to PORV control failure, 6 events involved design or fabrication of PORVs, and 32 events involved BV failures. The report contains a compilation of the PORV and BV failure events, including failure cause and severity. The events are identified as to plant and valve manufacturer. An assessment of the need to upgrade PORVs and BVs to safety grade status concludes that such action would improve PORV and BV reliability. The greatest improvement in reliability would result from using newer, more reliable PORV designs and improving testing, diagnostics and maintenance applied to PORVs and BVs, particularly the BV motor operator. A summary of interviews conducted with four PORV manufacturers is also included in the report.

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. Battelle Memorial Institute, Columbus Laboratories. February 1987. 62pp. 8704070470. EGG-2404. 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-4700 V2 DRF: CONTAINMENT EVENT ANALYSIS FOR POSTULATED SEVERE ACCIDENTS: SEQUOYAH POWER STATION,UNIT 1.Draft Report For Comment. BEHR,V.L.; BENJAMIN,A.S.; KUNSMAN,D.M.; et al. Sandia National Laboratories. February 1987. 208pp. 8705200197. SAND86-1135. 40986:165.

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

cess in the injection phase. This work supports NRC's assessment of severe accident risks to be published in NUREG-1150.

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

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

NUREG/CR-4701 V02: SAFETY ASSESSMENT OF ALTERNATIVES TO SHALLOW-LAND BURIAL OF LOW-LEVEL RADIOACTIVE WASTE.Volume 2:Environmental Conditions Affecting Reliability Of Engineered Barriers. CERVEN,F.; OTIS,M.D. EG&G Idaho, Inc. (subs. of EG&G, Inc.). September 1987. 45pp. 8711090388. EGG-2465. 43323:232.

The need for new disposal capacity for low-level radioactive waste (LLW) has led to a re-examination of disposal practices. A number of enhancements and alternatives to traditional shallow-land burial have been proposed to meet the need for new capacity and to address various concerns about the performance history of existing commercial LLW sites. This document builds on the results of the Volume I effort, which identified the important LLW disposal facility engineered barriers of cover and structure. Fifteen potentially important degradation mechanisms for a LLW facility are identified, categorized, and analyzed to determine their importance to the proper functioning of the disposal facility over its 500-year lifetime. Wind storms, biological intrusion, mechanical settling, freeze/thaw cycling, chemical degradation, wind erosion, and water erosion were considered the most important mechanisms. Data supporting concrete structure long-term performance in sulfate environments and long-term cover performance in erosive and biological intrusion environments were obtained. Research on the performance of covers and concrete structures in the presence of the other listed degradation mechanisms is recommended.

NUREG/CR-4704 V01: RELATIVE BIOLOGICAL EFFECTIVENESS (RBE) OF FISSION NEUTRONS AND GAMMA RAYS AT OCCUPATIONAL EXPOSURE LEVELS.Volume 1: Studies On The Genetic Effects In Mice Of 60 Equal Once-Weekly Exposures To Fisson Neutrons And Gamma Rays. GRAHN,D.; CARNES,B.A. Argonne National Laboratory. October 1987. 71pp. 8712170245. ANL-86-33. 43761:209.

The relative biological effectiveness (RBE) values for low doses of fission neutrons compared to (60) Co gamma rays were determined with four separate assessments of genetic damage induced in young hybrid male mice. Both radiations were delivered at low dose levels over about one-half the adult lifetime as 60 once-weekly exposures. Genetic damage assessed included both transient and residual injury. The latter is more critical, as residual genetic injury can be transmitted to subsequent generations long after the radiation exposures have ceased. Assays were performed periodically during the 60-week exposure period and at 10 or more weeks after the irradiations had terminated. RBE values, with few exceptions, ranged between 5 and 15 for transient injury and between 25 and 50 for different types of residual genetic injury. The most important form of residual genetic damage in this study was the balanced reciprocal chromosome translocation. These translocations continue to be transmitted throughout reproductive life and can lead to reduced fertility and increased prenatal mortality. The best estimate of the RBE value for translocations was 45 plus minus 10. Implications and recommendations with regard to the neutron quality factor will be presented conjointly with the findings from the data obtained in this same project on life shortening and on the risks of incidence or death from neoplastic disease.

NUREG/CR-4704 V02: RELATIVE BIOLOGICAL EFFECTIVENESS (RBE) OF FISSION NEUTRONS AND GAMMA RAYS AT OCCUPATIONAL EXPOSURE LEVELS.Volume II: Studies On The Effects Of 60 Equal Once-Weekly Exposures To Fission Neutrons And Gamma Rays On Survival Of Mice. THOMSON,J.F.; GRAHN,D. Argonne National Laboratory. October 1987. 45pp. 8712310077. ANL-86-33. 43871:252.

A total of 6316 mice (C57BL/6JAnl x BALB/cJAnl F1 hybrids) were exposed to 60 once-weekly doses of 0.85 MeV fission neutrons (0.033 to 0.67 cGy per weekly fraction) or (60)Co gamma rays (1.67 to 10 cGy per weekly fraction) and observed until they died. An additional 1404 mice were entered into the experiment and followed for part of their lifetimes; a few of these mice were lost accidentally, but most were removed for genetic testing. The mean aftersurvival (MAS) times showed dose-response curves for both neutron and gamma-ray exposures to be linear over all doses except the highest neutron dose. The relative biological effectiveness (RBE) value for neutrons, calculated as the ratio of the linear slopes of the dose-response curves for MAS times, was about 20 for both males and females. Essentially the same value was obtained by other analyses of the data. This RBE value of 20 is specific for deaths from all causes after 60 once-weekly exposures to 0.85 MeV fission neutrons, with once-weekly (60)Co gamma- ray exposures as the reference radiation. The value for the RBE will probably be different for some, but not all, of the other end points (i.e., specific causes of death, especially tumors).

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 approxi-

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

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

See NUREG/CR-4674,V03 abstract.

NUREG/CR-4679: QUANTITATIVE DATA ON THE FIRE BEHAVIOR OF COMBUSTIBLE MATERIALS FOUND IN NUCLEAR POWER PLANTS:A Literature Review. NOWLEN,S.P. Sandia National Laboratories. February 1987. 166pp. 8706120209. SAND86-0311. 41285:215.

This report presents the findings of a task in which currently available fire research literature was reviewed for quantitative data on the burning characteristics of combustible materials that are found in nuclear power plants. The materials considered for which quantitative data were available include cable insulation materials, flammable liquids, furniture, trash and general refuse, and wood and wood products. A total of 90 figures and tables, taken primarily from the referenced works, which summarize the available quantitative fire characterization information for these materials is presented.

NUREG/CR-4681: ENCLOSURE ENVIRONMENT CHARACTERIZATION TESTING FOR THE BASE LINE VALIDATION OF COMPUTER FIRE SIMULATION CODES. NOWLEN,S.P. Sandia National Laboratories. March 1987. 82pp. 8706150161. SAND86-1296. 41301:047.

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

NUREG/CR-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. 39527:138.

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-4690 V01 N1: GENERIC COMMUNICATIONS INDEX.Listings Of Communications (1971-1986). DEAN,R.S.; STEINBRECHER,D.; HENNICK,A. Parameter, Inc. December 1987. 320pp. 8712310192. 43874:272.

As part of its program to feed back information on operating experience to industry, the U.S. Nuclear Regulatory Commission (NRC) issues certain generic communications called bulletins (about 10/yr), information notices (about 100/yr), and circulars (now discontinued). The report presents an index of all such communications from 1971, when such documentation started, to 1986. The index is the printout from the computer database developed at the NRC for easy access to information previously published in these generic communications. The record (row) entries each consist of twenty fields (columns) of categorized information about a particular document. Included are fields for the document identification number, title, and NRC technical contact, and fields for general system or topic, specific component or topic, cause or defect, potential effect, remarks, and vendors involved. The report also contains lists of communications sorted by each subject category of the General System or Topic and the Specific Component or Topic fields. A list of the category subjects used and a list of vendors involved in the communications are included, also. The NRC intends to periodically update this report to index later communications.

NUREG/CR-4690 V02: GENERIC COMMUNICATIONS INDEX.User's Manual. DEAN,R.S.; STEINBRECHER,D.; HENNICK,A.; et al. Parameter, Inc. December 1987. 20pp. 8712310211. 43874:253.

This report is a manual for providing information required to use a special computer program developed by the NRC for indexing generic communications. The program is written in a

mately 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-4710: SHUTDOWN DECAY HEAT REMOVAL ANALYSIS OF A COMBUSTION ENGINEERING PRESSURIZED WATER REACTOR.Case Study. CRAMOND,W.R.; ERICSON,D.M.; SANDERS,G.A. Sandia National Laboratories. July 1987. 775pp. 8708110412. SAND86-1797. 42107:282.

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 Combustion Engineering 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 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-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. Oak Ridge National Laboratory. * Babcock & Wilcox Co. 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. 122pp. 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. 903pp. 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-4715: AN AGING ASSESSMENT OF RELAYS AND CIRCUIT BREAKERS AND SYSTEM INTERACTIONS. TOMAN,G.J.; BACANSKAS,V.P.; SHOOK,T.A.; et al. Brookhaven National Laboratory. June 1987. 181pp. 8711090389. BNL-NUREG-52017. 43323:050.

This report provides an assessment of the aging of circuit breakers and relays which are vital components of the nuclear power plant electrical safety system, and was conducted under the auspices of the NRC Nuclear Aging Research (NPAR) Program. The study included protective, control, and logic relays, and molded-case and metal-clad switchgear circuit breakers. The relay failures were attributable to coil deterioration, changes in dimensions of critical organic components, and changes in characteristics of timing diaphragms from thermal deterioration. Some of the failure modes will prevent fail- safe operation. The electrical control and mechanical portions of metal-clad switchgear were found to be more failure prone than the main contacts and arc extinguishing systems. Circuit breakers and relays in a PWR safety injection system were evaluated with respect to the aging induced by system operation. The effect of circuit breaker and relay deterioration on the ability of the system to perform its safety functions was also evaluated.

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

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

NUREG/CR-4722: SOURCE TERM ESTIMATION USING MENU-TACT. SJOREEN,A.; MCKENNA,T.; JULIUS,J. Oak Ridge National Laboratory. October 1987. 65pp. 8711060294. ORNL/6314. 43311:025.

MENU-TACT is a computer code designed to be used for source term estimation during reactor emergencies. The code allows the user to define concentrations of the nuclides in the reactor and the release pathway and timing. Filtering and other in-plant removal processes may be included. MENU-TACT, a modification of the TACT-III code, is simple to use. To provide quick results of bounding calculations, it incorporates only those processes that can make major changes in reactor releases.

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-4726: EVALUATION OF PROTECTIVE ACTION RISKS. WITZIG,W.F.; SHILLENN,J.K. Pennsylvania State Univ., University Park, PA. June 1987. 212pp. 8707070524. 41600:074.

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

NUREG/CR-4727: SWISS:SUSTAINED HEATED METALLIC MELT/CONCRETE INTERACTIONS WITH OVERLYING WATER POOLS. BLOSE,R.E.; GRONAGER,J.E.; SUO-ANTTILA,A.; et al. Sandia National Laboratories. June 1987. 267pp. 8711170043. SAND85-1546. 43372:153.

The SWISS tests of melt/concrete interactions with an overlying water pool are described. Each test involved 46 Kg of molten stainless steel sustained by induction heating while in contact with limestone/common sand concrete. A water pool was formed over the melt in test SWISS 1 after about 12 cm of concrete had been eroded. In test SWISS 2, a water pool was formed immediately after melt contacted the concrete. Concrete erosion, temperatures, gas generation rates, and aerosol generation rates are reported.

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-4731 V01: RESIDUAL LIFE ASSESSMENT OF MAJOR LIGHT WATER REACTOR COMPONENTS - OVERVIEW.Volume 1. SHAH,V.N.; MACDONALD,P.E. EG&G Idaho, Inc. (subs. of EG&G, Inc.). June 1987. 198pp. 8708060423. EGG-2469. 42071:001.

This report presents an assessment of the aging (time-dependent degradation) of selected major light water reactor components and structures. The stressors, possible degradation sites and mechanisms, potential failure modes and currently used non-destructive examinations, in-service inspection (ISI) and life assessment methods are discussed for seven major light water reactor components: pressurized water reactor (PWR) and boiling water reactor (BWR) pressure vessels, PWR containment structures, PWR reactor coolant piping, PWR steam generators, BWR recirculation piping, and reactor pressure vessel supports. Unresolved technical issues related to life extension of these components, including requirements for advanced ISI and life assessment methods, are also discussed.

NUREG/CR-4734: SEISMIC TESTING OF TYPICAL CONTAINMENT PIPING PENETRATION SYSTEMS. CLOSE,J.A.; HILL,R.C.; STEELE,R. EG&G Idaho, Inc. (subs. of EG&G, Inc.). 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. 139pp. 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-4735 V02: EVALUATION AND COMPILATION OF DOE WASTE PACKAGE TEST DATA. Biannual Report:August 1986 - January 1987. INTERRANTE,C.; ESCALANTE,E.; FRAKER,A.; et al. Commerce, Dept. of, National Bureau of Standards. October 1987. 137pp. 8711090211. 43315:244.

This report summarizes results of the National Bureau of Standards (NBS) evaluations of Department of Energy (DOE) activities on waste packages designed for containment of radioactive high-level nuclear waste (HLW). The waste package is a proposed engineered barrier that is part of a permanent repository for HLW. Metal alloys are the principal barriers within the engineered system. Technical discussions are given for the corrosion of metals proposed for the canister, particularly carbon and stainless steels, and copper. In the section on tuff, the current level of understanding of several canister materials is questioned. Within the Basalt Waste Isolation Project (BWIP) section, discussions are given on problems concerning groundwater, materials for use in the metallic overpack, and diffusion through the packing. For the proposed salt site, questions are raised on the work on both the primary candidate material (ASTM A216 Steel) and the alternate alloy (Ti-Code 12). NBS work related to the vitrification of HLW borosilicate glass at the West Valley Demonstration Project (WVDP) and the Defense Waste Processing Facility (DWPF) is covered. NBS reviews of the Materials Characterization Center (MCC) is presented. Using a database management system, a computerized database for storage and retrieval of reviews and evaluations of HLW data has been developed and is described.

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

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

NUREG/CR-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. 121pp. 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-4744 V01 N2: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,April-September 1986. CHOPRA,O.K.; CHUNG,H.M. Argonne National Laboratory. May 1987. 54pp. 8709040241. ANL-87-16. 42526:101.

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

NUREG/CR-4744 V02 N1: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,October 1986 - March 1987. CHOPRA,O.K.; CHUNG,H.M. Argonne National Laboratory. November 1987. 52pp. 8801110207. ANL-87-45. 43975:157.

This progress report summarizes work performed by Argonne National Laboratory during the six months from October 1986-March 1987 on long-term embrittlement of cast duplex stainless steels used in light-water reactors.

NUREG/CR-4747 V01: AN AGING FAILURE SURVEY OF LIGHT WATER REACTOR SAFETY SYSTEMS AND COMPONENTS. MEALE,B.M.; SATTERWHITE,D. EG&G Idaho, Inc. (subs. of EG&G, Inc.). July 1987. 295pp. 8709090266. EGG-2473. 42578:129.

This report describes the methods, analyses, results, and conclusions of two different aging studies. The first analysis consists of a survey of light water reactor component failures associated with selected safety and support systems. Tables are presented, indicating the systems and the components within those systems most affected by aging. Also provided are engineering insights drawn from the data. The second analysis consists of identifying and categorizing the reported failure causes of component failures. The systems used in the failure-cause analysis were service water systems and Class 1E electrical power distribution systems for Babcock & Wilcox Company pressurized water reactors and service water systems for boiling 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. 686pp. 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-4753 V02: CANADIAN SEISMIC AGREEMENT.Annual Report. WETMILLER,R.J.; LYONS,J.A.; SHANNON,W.E.; et al. Canadian Commercial Corp. October 1987. 39pp. 8711100368. 43338:226.

This is the second report under the terms of the contract, NRC- 04-85-110, for a research agreement entitled "Canadian Seismic Agreement" between the U.S. Nuclear Regulatory Commission (NRC) and the Canadian Commercial Corporation. Activities undertaken by the Geophysics Division (GD) of the Geological Survey of Canada (GSC) during the period June 1986 to June 1987 and supported in part by the NRC agreement are described below under four headings: ECTN and portable network developments, data lab developments, strong motion network developments and earthquake activity. During the period of this report, the contract resources were spent on operation and maintenance of ECTN, servicing and maintenance (by contract) of the strong-motion seismograph network, operation and expansion of the Ottawa data lab and earthquake monitoring and reporting. The development of special purpose networks for Sudbury and Charlevoix are also reported on here because they are closely related to ECTN but they are supported wholly or in part by other sources.

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

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

NUREG/CR-4760: TEST OF 6-IN.-THICK PRESSURE VESSELS.Series 3:Intermediate Test Vessel V-8A:Tearing Behavior Of Low Upper-Shelf Material. BRYAN,R.H.; BASS,B.R.; BOLT,S.E.; et al. Oak Ridge National Laboratory. May 1987. 404pp. 8710060100. ORNL-6187. 42920:003.

Tests of several 152-mm-thick vessels have been performed to study the behavior of flaws under stress states similar to those in full-scale reactor pressure vessels. The objective of the latest test, V-8A, was to provide accurate quantitative data concerning the growth by ductile tearing and final instability of a flaw in a low- upper-shelf toughness weld located in a cylinder of reactor vessel steel. This test is important because there are vessels in service that contain welds which, because of high copper content, may have their Charpy upper-shelf energy values reduced to relatively low levels by neutron irradiation. The results of the V-8A test are intended to provide an experimental basis for judging the accuracy of vessel fracture safety analysis procedures for low-upper-shelf toughness conditions. The objective of the V-8A test was attained, with a tearing instability observed at a pressure of 139 MPa (2 x design pressure). The flaw, which was initially a fatigue-sharpened notch with an approximately elliptical profile, grew in depth and length to 101.4 mm and 453 mm, respectively. Pretest and posttest fracture-mechanics and stress analyses were made by simplified methods, convenient for investigating a wide range of parameters, and by three-dimensional finite element methods, which modeled the material properties and geometry more precisely. Ductile flaw growth and instability predictions based upon measured J-resistance and tensile properties were made. Results of analyses based on J(R)- controlled crack growth agree reasonably well with experimental observations.

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. 827pp. 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-4765: MXS CROSS-SECTION PREPROCESSOR USER'S MANUAL. PARKER,F.; LUCK,L. Los Alamos National Laboratory. ISHIKAWA,M. FBR Safety Laboratory, Ibaraki Prefecture, Japan. March 1987. 58pp. 8705190577. LA-10856-M. 40966:217.

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

NUREG/CR-4766: USER'S MANUAL FOR THE NEFTRAN COMPUTER CODE. LONGSINE,D.E.; BONANO,E.J.; HARLAN,C.P. Sandia National Laboratories. September 1987. 230pp. 8712280248. SAND86-2405. 43816:051.

This document describes the NEFTRAN (NEtwork Flow and TRANsport) computer code and it is intended to provide the reader with detailed enough information in order to use the code. This code is a successor to NWFT/DVM (Campbell et al., 1981) and contains several new capabilities. These are: 1) generalized flow network, 2) matrix diffusion, 3) leg transfer, 4) mixing cell, and 5) multiple chains. This document is divided into four main sections describing 1) mathematical representation of ground-water flow, radionuclide transport, matrix diffusion, source term, and leg transfer models; 2) the program and subprograms in the code; 3) the input to the code; and 4) sample problems highlighting each of the new capabilities of the code.

NUREG/CR-4767: SHUTDOWN DECAY HEAT REMOVAL ANALYSIS OF A GENERAL ELECTRIC BWR4/MARK I.Case Study. HATCH,S.W.; ERICSON,D.M.; SANDERS,G.A. Sandia National Laboratories. July 1987. 745pp. 8710140105. SAND86-2419. 43051:001.

A General Electric Boiling Water Reactor (BWR4) 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 implementa-

tion 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 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-4768 V01: METHODOLOGY AND APPLICATION OF SURROGATE PLANT PRA ANALYSIS TO THE RANCHO SECO POWER PLANT.Task 1 - Analysis Of ANO-1 And Oconee PRAs. GORE,B.F. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1987. 75pp. 8707310128. PNL-6032-1. 41996:304.

This two-volume report presents the development and first application of a methodology for using generic PRA information to identify risk-important systems and components at a plant lacking a PRA. The methodology requires the detailed analysis of both similarities and differences between related plants. It is applied in an analysis of the Rancho Seco plant using information from PRAs for the ANO-1 and Oconee plants. It is generic, drawing upon the functional and design similarities of B&W plants, yet it incorporates considerable plant specificity through the detailed comparative analysis of systems. Volume 1 presents the analysis of the surrogate plant PRAs. Dominant cut sets leading to core melt are identified and analyzed to determine the Fussel-Vesely importance of plant systems. In Volume 2 the dominant cut sets are further analyzed to identify and categorize important system failure modes. The Rancho Seco plant is then studied to determine the plausibility and importance of similar failure modes for: High Pressure Injection, Low Pressure Injection, Emergency Feedwater, Service Water, Vital AC power and DC Power systems. Plant-specific information is then presented identifying Rancho Seco components, power supplies, and operating modes associated with the failure modes for the first four of these systems.

NUREG/CR-4768 V02: METHODOLOGY AND APPLICATION OF SURROGATE PLANT PRA ANALYSIS TO THE RANCHO SECO POWER PLANT.Final Report. GORE,B.F.; HUENEFELD,J.C. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1987. 87pp. 8707310413. PNL-6032-2. 41994:017.

See NUREG/CR-4768,V01 abstract.

NUREG/CR-4769: RISK EVALUATIONS OF AGING PHENOMENA.The Linear Aging Reliability Model And Its Extensions. VESELY,W.E. EG&G Idaho, Inc. (subs. of EG&G, Inc.). April 1987. 176pp. 8708130348. EGG-2476. 42154:110.

A model for light water reactor safety system component failure rates due to aging mechanisms has been developed from basic phenomenological considerations. In the treatment, the occurrences of deterioration are modeled as following a Poisson process. The severity of damage is allowed to have any distribution, however, the damage is assumed to accumulate independently. Finally, the failure rate is modeled as being proportional to the accumulated damage. Using this treatment, the linear aging failure rate model is obtained. The applicability of the linear aging model to various mechanisms is discussed. The model is also extended to cover nonlinear and dependent aging phenomena. The implementation of the linear aging model is demonstrated by applying it to the aging data collected in the U.S. Nuclear Regulatory Commission's Nuclear Plant Aging Research Program.

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

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

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

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

NUREG/CR-4776: RESPONSE OF SEISMIC CATEGORY I TANKS TO EARTHQUAKE EXCITATION. BUTLER,T.A.; BENNETT,J.G.; BABCOCK,C.D.; et al. Los Alamos National 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-4779: NEW DATA FOR AEROSOLS GENERATED BY RELEASES OF PRESSURIZED POWDERS AND SOLUTIONS IN STATIC AIR. BALLINGER,M.Y.; SUTTER,S.L.; HODGSON,W.H. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1987. 51pp. 8706240046. PNL-6065. 41448:248.

Safety assessments and environmental impact statements for nuclear fuel cycle facilities require an estimate of potential airborne releases. Aerosols generated by accidents are being investigated by Pacific Northwest Laboratory to develop radioactive source-term estimation methods. Experiments measuring the mass airborne and particle size distribution of aerosols produced by pressurized releases were run. Carbon dioxide was used to pressurize uranine solutions to 50, 250, and 500 psig

before release. The mass airborne from these experiments was higher than for comparable air- pressurized systems, but not as great as expected based on the amount of gas dissolved in the liquid and the volume of liquid ejected from the release equipment. Flashing sprays of uranine at 60, 125, and 240 psig produced a much larger source term than all other pressurized releases performed under this program. Low-pressure releases of depleted uranium dioxide at 9, 17.5, and 24.5 psig provided data in the energy region between 3-m spills and 50-psig pressurized releases.

NUREG/CR-4781 DRFT: STUDY OF SEVERE ACCIDENT MITIGATION SYSTEMS. CHERDACK,R.; HESS,C.; LEE,K.; et al. Sandia National Laboratories. May 1987. 702pp. 8707300021. SAND87-7064. 41977:020.

This study has been performed as part of the Severe Accident Risk Reduction Program (SARRP) to investigate the feasibility and costs of installing various systems designed to mitigate the effects of severe accidents in light water reactors (LWRs). The primary function of the systems is to maintain the integrity of the reactor containment in the event of a severe accident - one in which substantial core melting and reactor vessel failure occur. The systems evaluated include (1) a hydrogen ignition system with capabilities for detecting hydrogen concentrations, initiating hydrogen burning, and avoiding the triggering of detonations; (2) a reactor cavity flooding system capable of introducing large quantities of water in the reactor cavity and adjacent regions; (3) a core debris management system that would prevent the core debris from a failed reactor vessel from interacting with significant quantities of water and with the concrete of the cavity; (4) an alternate containment spray system that would operate without relying on any existing plant electrical power supplies; and (5) a passive containment heat removal system that would remove heat from the containment atmosphere by condensing steam. These systems were considered for installation in three types of LWR containments. Conceptual designs and associated cost estimates were developed for two proposed combinations of these systems. This work provides an input to the cost-benefit analyses conducted in the SARRP Program in support of NRC's assessment of severe accident risks to be published in NUREG-1150.

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

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

NUREG/CR-4786: TRANSPORT BEHAVIOR OF IODINE IN EFFLUENT RADIOACTIVITY MONITORING SYSTEMS. EDSON,J.L.; DUCE,S.W.; TKACHYK,J.W. EG&G Idaho, Inc. (subs. of EG&G, Inc.). October 1987. 35pp. 8711100261. EGG-2480. 43341:253.

This report presents the results of tests conducted on a mockup of the sample line of an effluent radioactivity monitoring system at a nuclear power plant. The purpose of the tests was to evaluate the ability of the radioactivity monitoring system sample lines to transmit radioactive elemental iodine during transient conditions. The sample line was tested initially with manufacturing contamination left in the line; then was cleaned and retested to evaluate the effect of cleaning on transient performance. The sample line tested was of a size [2.54 cm (1 in.) diameter and 4572 cm (150 ft) long] that was expected to have acceptable performance during a nuclear reactor accident. The tests were conducted in a laboratory where radioactive iodine could be generated safely, test variables closely controlled, and test data measured.

NUREG/CR-4787: CONFERENCE OF RADIATION CONTROL PROGRAM DIRECTORS' 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 circu-

lation 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-4795: LONG-TERM PERFORMANCE OF HIGH-LEVEL GLASS WASTE FORMS. MEANS,J.L.; SPINOSA,E.D.; MARKWORTH,A.J.; et al. Battelle Memorial Institute, Columbus Laboratories. November 1987. 176pp. 8712240049. BMI-2143. 43794:276.

This report investigates the long-term performance of materials used for high-level waste packages. Glass waste-form studies, the topic of this report, focused on borosilicate glasses for both defense and commercial high-level wastes. Numerical modeling efforts describe the kinetics of glass dissolution, including the effects of water chemistry, and the kinetics of glass leaching assuming that silica precipitates as a more stable phase subsequent to dissolution from the glass. Glass leaching experiments conducted in this task included experimental validation of the numerical model for glass dissolution and precipitation, an evaluation of the effects of two natural organic acids on leaching of simulated waste glass, and an experimental study of the leaching behavior of devitrified MCC 76-68 glass specimens. Literature reviews also were conducted on the effects of both devitrification and radiation on glass-leaching behavior. Other experiments documented the presence of experimental artifacts in glass leach tests under certain experimental conditions and examined the synergistic effects of temperature, pressure, surface-area-to-volume ratio, and leachate composition on glass leach rates.

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-4799: RELAP5/MOD2 ASSESSMENT SIMULATION OF SEMISCALE MOD-2C TEST S-NH-3. MEGAHED,M.M. EG&G Idaho, Inc. (subs. of EG&G, Inc.). October 1987. 53pp. 8711090336. EGG-2519. 43316:195.

This report documents an evaluation of the RELAP5/MOD2/Cycle 36.05 thermal-hydraulic computer code for a simulation of a small- break loss-of-coolant accident transient (SBLOCA). The experimental data base for the evaluation is the result of Test S-NH-3 performed in the Semiscale MOD-2C test facility. The test modeled a 0.5% SBLOCA with an accompanying failure of the high-pressure injection emergency core cooling system. The test facility and RELAP5/MOD2 model used in the calculations are described. Evaluations of the accuracy of the calculations are presented in the form of comparisons of measured and calculated histories of selected parameters associated with the primary and secondary systems. A conclusion was reached that the code is capable of making SBLOCA calculations efficiently. However, some of the SBLOCA-related phenomena were not

properly predicted by the code, suggesting a need for code improvement.

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

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

NUREG/CR-4801: CLIMATOLOGY OF EXTREME WINDS IN SOUTHERN CALIFORNIA. RAMSDELL,J.V.; HUBBE,J.M.; ELLIOTT,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-4802: AN EVALUATION OF TRAC-PF1/MOD1 COMPUTER CODE PERFORMANCE DURING POSTTEST SIMULATIONS OF SEMISCALE MOD-2C FEEDWATER LINE BREAK TRANSIENTS. HALL,D.G.; WATKINS,J.C. EG&G Idaho, Inc. (subs. of EG&G, Inc.). February 1987. 204pp. 8704270202. EGG-2486. 40681:173.

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

NUREG/CR-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-4805 V01: REACTOR SAFETY RESEARCH SEMI-ANNUAL REPORT.January-June 1986. * Sandia National Laboratories. May 1987. 359pp. 8707300091. SAND86-2752. 41983:130.

Sandia National Laboratories is conducting, under USNRC sponsorship, phenomenological research related to the safety of commercial nuclear power reactors. The research includes experiments to simulate the phenomenology of the accident conditions and the development of analytical models, verified by experiment, which can be used to predict reactor and safety systems performance and behavior under abnormal conditions. The objective of this work is to provide NRC requisite data bases and analytical methods to (1) identify and define safety issues, (2) understand the progression of risk- significant accident sequences, and (3) conduct safety assessments. The collective NRC-sponsored effort at Sandia National Laboratories is directed at enhancing the technology base supporting licensing decisions.

NUREG/CR-4808: MINTEQ USER'S MANUAL. PETERSON,S.R.; HOSTETLER,C.J.; DEUTSCH,W.J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1987. 159pp. 8710220424. PNL-6106. 43132:130.

This MINTEQ user's manual was written to aid the user in applying the MINTEQ geochemical computer code to model aqueous solutions and the interactions of aqueous solutions with hypothesized assemblages of solid phases. The purpose of the user's manual is twofold: (1) to provide a basic understanding of how the MINTEQ computer code operates and the important principles that are incorporated into the code and (2) to instruct a user of the MINTEQ code on how to create input files to simulate a variety of geochemical problems. Chapters 2 through 8 are directed toward the user who has some experience with or wishes to review the principles important to geochemical computer codes. Also contained in these chapters is information on the methodology MINTEQ uses to incorporate these principles into the code. Chapters 9 through 11 are directed toward the user who wants to know how to create input data files to model various types of problems.

NUREG/CR-4810: EVALUATION OF DIESEL UNAVAILABILITY AND RISK EFFECTIVE SURVEILLANCE TEST INTERVALS. VESELY,W.E.; DEMOSS,G.; LOFGREN,E.; et al. Brookhaven National Laboratory. May 1987. 83pp. 8708260121. BNL-NUREG-52022. 42331:180.

In this report risk and reliability approaches are presented which allow risk acceptable test intervals to be determined for any diesel. Data required to apply the approaches are also de-

scribed. The approaches can be applied not only to diesels, but to any component with suitable data. Incorporation of the approaches in personal computer (PC) software is discussed, which can provide tools for the regulator or plant personnel for any plant specific or generic application. The FRANTIC III computer code was run to validate the approaches and to evaluate specific issues associated with determining risk-effective test intervals for diesels. Using the approaches presented, diesel accident unavailability can be more effectively monitored and controlled on a plant-specific or generic basis. Test intervals can be made more risk effective than they are now, producing more acceptable accident unavailabilities, and avoiding the possible deleterious effects of Regulatory Guide 1.108. The methods presented are one step toward performance-based technical specifications, which more directly control risks.

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

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

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

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

NUREG/CR-4817: IODINE PARTITION COEFFICIENT MEASUREMENTS AT SIMULATED PWR STEAM GENERATOR CONDITIONS.Interim Data Report. CLINTON,S.D.; SIMMONS,C.M. Oak Ridge National Laboratory. May 1987. 28pp. 8708060384. ORNL/TM-10330. 42075:002.

Iodine partition coefficients (defined as the ratio of the concentration of iodine species in the aqueous solution to the iodine concentration in the vapor phase) were measured at simulated PWR steam generator conditions (285 degrees C and 6.9 MPa), using carrier-free radioactive I-131 in the form of sodium iodide. The iodine tracer concentration was maintained at 6 x 10(-11) mol/L; boric acid concentration was varied from 0 to 0.4 mol/L; and the solution pH (measured at 25 degrees C) was adjusted from 4 to 9 by the addition of lithium hydroxide. Iodine partition coefficients decrease with increasing boric acid concentration; however, the iodine volatility is essentially independent of the solution pH for a given boric acid concentration. Sparging the solutions with air at room temperature increases the iodine volatility by an order of magnitude, compared to that achieved with argon sparging. Iodine partition coefficient measurements ranged from a low of 200 (in 0.2 molarity boric acid sparged with air) to 400,000 (in purified water sparged with argon).

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 in present in the static test.

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

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

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

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

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

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

NUREG/CR-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. 217pp. 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. Furthermore, 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. 190pp. 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 seis-

mic 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. Lawrence Livermore National Laboratory. 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. Lawrence Livermore National Laboratory. March 1987. 236pp. 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-4830: MELCOR VALIDATION AND VERIFICATION 1986 PAPERS. LEIGH,C.D. Sandia National Laboratories. March 1987. 223pp. 8706120166. SAND86-2689. 41281:049.

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

NUREG/CR-4834 V01: RECOVERY ACTIONS IN PRA FOR THE RISK METHODS INTEGRATION AND EVALUATION PROGRAM (RMIEP).Volume 1:Development Of The Data-Based Method. WESTON,L.M.; WHITEHEAD,D.W. Sandia National Laboratories. GRAVES,N.L. Energy, Inc. June 1987. 295pp. 8709090367. SAND87-0179. 42572:071.

In a probabilistic risk assessment (PRA) for a nuclear power plant, the analyst identifies a set of potential core damage events consisting of equipment failures and human errors and their estimated probabilities of occurrence. If operator recovery from an event within some specified time is considered, the probability of this recovery can be included in the PRA. This report provides PRA analysis with an improved methodology for including recovery actions in a PRA. A recovery action can be divided into two distinct phases, a Diagnosis Phase (realizing that there is a problem with a critical parameter and deciding upon the correct course of action), and an Action Phase (physically accomplishing the required action). In this methodology, simulator data are used to estimate recovery probabilities for the diagnosis phase. Different time-reliability curves showing the probability of failure of diagnosis as a function of time from the compelling cue for the event are presented. These curves are based on simulator exercises, and the actions are grouped based upon their operational similarities. This is an improvement over existing diagnosis models that rely greatly upon subjective judgment to obtain such estimates. The action phase is modeled using estimates from available sources. The methodology also includes a recommendation on where and when to apply the recovery action in the PRA process.

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

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

results. Characterization of the irradiated material will be presented in a subsequent report.

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

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

NUREG/CR-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. 300pp. 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-4844 DRFT: INTEGRATED RELIABILITY AND RISK ANALYSIS SYSTEM (IRRAS) USER'S GUIDE - VERSION 1.0 (DRAFT). RUSSELL,K.D.; SNIDER,D.M.; SATTISON,M.B.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). June 1987. 149pp. 8707300200. EGG-2495. 41986:016.

The Integrated Reliability and Risk Analysis System (IRRAS) is an integrated PRA software tool which gives the user the ability to create and analyze fault trees using an IBM PC. This program provides functions that range from graphical fault tree construction to cut set generation and quantification. Also provided in the system is an integrated full-screen editor for use when interfacing with remote computer systems. IRRAS-PC is being developed at the INEL as the USNRC's state-of-the-art microcomputer based probabilistic risk assessment (PRA) model development and analysis tool to address key nuclear plant safety issues. The INEL role in the IRRAS program is that of software developer and interfacer to the user community, including training and technology transfer. To support this role, this user's manual was developed as a guideline for use of the IRRAS software. Presented in the manual is an explanation of the installation procedures, the hardware and software requirements, the forms and menus used by the program, and detailed explanations of the programs's capabilities and functions. The level of detail presented in this manual is intended to guide the beginning or infrequent user. IRRAS-PC has all the capabilities and functions required to create, modify, reduce, and analyze fault tree models used in the analysis of complex systems and processes. IRRAS-PC uses advanced graphic and analytical techniques to achieve the greatest possible realization of the potential of the microcomputer and when the needs of the user exceed this potential IRRAS-PC can call upon the power of the mainframe.

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

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

NUREG/CR-4846: HIGH-LEVEL WASTE PRECLOSURE SYSTEMS SAFETY ANALYSIS.Phase 2. Final Report. LIGON,D.M.; STAMATELATOS,M.; BARSELL,A.W.; et al. Sandia National Laboratories. June 1987. 163pp. 8708250176. SAND87-7029. 42330:277.

The major effort of this phase was the demonstration of the methodology developed in Phase I. A sample problem consisting of six scenarios was quantified. The purpose of this sample problem is to check the application of the assembled methods, particularly the importance ranking scheme. All of the features of the complete analytical technique are applied including uncertainty and sensitivity analysis, common cause failure evaluation and human error contribution. Other topics addressed include Mine-Related Data Development, Human Reliability Analysis and Common Cause Failure Analysis.

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 that may be involved in future spent fuel shipping campaigns.

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

This report is a summary of the Steam Generator Group Project progress for 1985. Statistical analyses of data from nondestructive examination (NDE) round robins performed on the Surry generator are presented. Criteria are listed for selection of tube specimens to be removed from the generator for validation of the NDE round robin results. A sampling plan is described along with the initial steps taken to implement this plan. Special tooling fabricated for specimen removals is discussed. Removal of three 9-tube sections of tube sheet is presented. Preliminary

results from destructive analysis of one section are reported. Validation activities were initiated by metallurgical evaluation of two tubes pulled from the hot leg tube sheet. Results from destructive measurement of maximum defect depth from these two tubes indicated that in situ bobbin coil eddy current measurements consistently undersized the defects.

NUREG/CR-4850: STEAM GENERATOR GROUP PROJECT.Task 10 - Secondary Side Examination.Final Report. SCHWENK,E.B. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1987. 107pp. 8708240213. PNL-6045. 42316:155.

The Steam Generator Group Project (SGGP) is using the retired- from-service Surry 2A pressurized water reactor steam generator as a test bed to investigate the reliability and effectiveness of in- service nondestructive eddy current inspection equipment and procedures. In addition, service degraded tubes from the generator will be used to validate models of remaining tube integrity developed in Phase I of the program. This information will provide the technical basis for revision of Regulatory Guides 1.83 and 1.121 governing in-service inspection and tube plugging criteria, respectively. This report is the second of two reports covering examination and characterization of the secondary side of the Surry unit. It includes a photographic survey of typical Surry generator material and structural conditions, metallurgical failure analysis of a ruptured U-bend tube, a description on how secondary side access penetrations were made and an experimental stress analysis of a bent Row 1 tube.

NUREG/CR-4851: SEISMICITY 1886-89 IN THE SOUTHEASTERN UNITED STATES.The Aftershock Sequence Of The Charleston,South Carolina Earthquake. ARMBRUSTER,J.G.; SEEBER,L. Lamont-Doherty Geological Observatory. May 1987. 202pp. 8709090380. 42569:334.

A search of contemporary newspapers in the Carolinas, Georgia and Eastern Tennessee during the 1886-1889 (inclusive) aftershock sequence of the Charleston earthquake of 1886 has provided more than 3000 intensity reports for 522 earthquakes as compared to 144 previously known earthquakes for the same period. Of these 144 events, 138 were felt in Charleston/Summerville and had been assigned epicenters in that area. The new data provide 112 well-constrained macroseismic epicenters. The 1886-1889 seismicity is characterized by a linear relation between log frequency and magnitude with a slope b1, a temporal decay of earthquake frequency proportional to time (-1), and a low level of seismicity prior to the main shock. These are frequently observed characteristics of aftershock sequences. By 1889, the level of seismicity had decreased more than 2 orders of magnitude, reaching approximately the current level in the same area. The 1886-1889 epicenters delineate a large aftershock zone that extends northwest about 250 km from the coast into the Piedmont and at least 100 km along the Fall Line. The aftershock zone occupies the same area as a zone of recent seismicity with unique characteristics. The portion of this zone in the Piedmont is the only area of Southeastern United States where reservoir-induced seismicity is unambiguously recognized. Seismicity elsewhere in the Southeast, such as in the Virginia seismic zone, is deeper and apparently unrelated to reservoir loading.

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-4855: DEVELOPMENT AND APPLICATION OF A COMPUTER MODEL FOR LARGE-SCALE FLAME ACCELERATION EXPERIMENTS. MARX,K.D. Sandia National Laboratories. July 1987. 80pp. 8801070052. SAND87-8203. 43943:297.

A new computational model for large-scale premixed flames is developed and applied to the simulation of flame acceleration experiments. The primary objective is to circumvent the necessity for resolving turbulent flame fronts; this is imperative because of the relatively coarse computational grids which must be used in engineering calculations. The essence of the model is to artificially thicken the flame by increasing the appropriate diffusivities and decreasing the combustion rate, but to do this in such a way that the burn velocity varies with pressure, temperature, and turbulence intensity according to prespecified phenomenological characteristics. The model is particularly aimed at implementation in computer codes which simulate compressible flows. To this end, it is applied to the two-dimensional simulation of hydrogen-air flame acceleration experiments in which the flame speeds and gas flow velocities attain or exceed the speed of sound in the gas. It is shown that many of the features of the flame trajectories and pressure histories in the experiments are simulated quite well by the model. Using the comparison of experimental and computational results as a guide, some insight is developed into the processes which occur in such experiments.

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-4860: FLAW DENSITY EXAMINATIONS OF A CLAD BOILING WATER REACTOR PRESSURE VESSEL SEGMENT. COOK,K.V.; MCCLUNG,R.W. Oak Ridge National Laboratory. April 1987. 33pp. 8707310061. ORNL/TM-10364. 41996:179.

As part of the Oak Ridge National Laboratory's Heavy-Section Steel Technology Program, studies have been conducted to determine flaw density in a section of reactor pressure vessel cut from the Hope Creek Unit 2 vessel. This boiling water reactor vessel was never in service. One objective was to evaluate the approximate 0.7- by 3-m (2- by 10-ft) segment of the vessel provided using ultrasonic flaw detection methods performed with both ASME Code techniques and supplemental ultrasonic methods. A second objective was to evaluate the inner surface stainless steel cladding for cracks with a high sensitivity penetrant examination. Both objectives were successfully completed. Five Code-recordable indications were detected ultrasonically; however, all were found to be anomalies associated with the cladding. One flaw was detected by supplemental ultrasonic tests, and it was analyzed destructively. This flaw was a pipelike indication, about 20 mm (0.8 in.) long extending along the length of the longitudinal weld in which it was located and was about 20 mm below the cladding surface. The flaw had a through-wall dimension (or length) of about 6 mm (0.24 in.) for an approximate 3-mm (0.1 in.) distance along the 20-mm major length. No flaws were detected by the penetrant examination of the cladding surface.

NUREG/CR-4861: DEVELOPMENT OF SITE SPECIFIC RESPONSE SPECTRA. BERNREUTER,D.J.; CHEN,J.C.; SAVY,J.B. Lawrence Livermore National Laboratory. March 1987. 140pp. 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-4862 V01: COGNITIVE ENVIRONMENT SIMULATION:AN ARTIFICIAL INTELLIGENCE SYSTEM FOR HUMAN PERFORMANCE ASSESSMENT.Volume 1:Summary And Overview. WOODS,D.D.; ROTH,E.M.; POPLE,H. Westinghouse Electric Corp. November 1987. 56pp. 8712090100. 43631:312.

This report documents the results of Phase II of a three-phase research program to develop and validate improved methods to model the cognitive behavior of nuclear power plant (NPP) personnel. In Phase II a dynamic simulation capability for modeling how people form intentions to act in NPP emergency situations was developed based on techniques from artificial intelligence. This modeling tool, Cognitive Environment Simulation or CES, simulates the cognitive processes that determine situation assessment and intention formation. It can be used to investigate analytically what situations and factors lead to intention failures, what actions follow from intention failures (e.g., errors of omission, error of commission, common mode errors, the ability to recover from errors or additional machine failures, and the effects of changes in the NPP person-machine system. The Cognitive Reliability Assessment Technique (or CREATE) was also developed in Phase II to specify how CES can be used to enhance the measurement of the human contribution to risk in probabilistic risk assessment (PRA) studies. The results are reported in three self- contained volumes that describe the research from different perspectives. Volume 1 provides an overview of both CES and CREATE. Volume 2 gives a detailed description of the structure and content of the CES modeling environment and is intended for those who want to know how CES models successful and erroneous intention formation. Volume 3 describes the CREATE methodology for using CES to provide enhanced human reliability estimates. Volume 3 is intended for those who are interested in how the modeling capabilities of CES can be utilized in human reliability assessment and PRA.

NUREG/CR-4862 V02: COGNITIVE ENVIRONMENT SIMULATION:AN ARTIFICIAL INTELLIGENCE SYSTEM FOR HUMAN PERFORMANCE ASSESSMENT.Volume 2:Modeling Human Intention Formulation. WOODS,D.D.; ROTH,E.M.; POPLE,H. Westinghouse Electric Corp. November 1987. 138pp. 8712080216. 43588:008.

See NUREG/CR-4862,V01 abstract.

NUREG/CR-4862 V03: COGNITIVE ENVIRONMENT SIMULATION:AN ARTIFICIAL INTELLIGENCE SYSTEM FOR HUMAN PERFORMANCE ASSESSMENT.Volume 3:Cognitive Reliability Analysis Technique. WOODS,D.D.; ROTH,E.M. Westinghouse Electric Corp. November 1987. 79pp. 8712080200. 43587:195.

See NUREG/CR-4862,V01 abstract.

NUREG/CR-4863: SURFACE SPECTROSCOPY OF PRESSURE VESSEL STEEL FATIGUE FRACTURE SURFACE FILMS FORMED IN PWR ENVIRONMENTS. HANNINEN,H.E.; VULLI,M.; CULLEN,W.H. Materials Engineering Associates, Inc. July 1987. 100pp. 8708040253. MEA-2194. 42036:227.

The composition and structure of corrosion products formed on corrosion fatigue fracture surfaces of pressure vessel steels tested in PWR-water conditions have been analyzed by using X-Ray Photoelectron Spectroscopy (XPS) and Auger Electron Spectroscopy (AES) techniques. This was the first time these electron spectroscopic techniques have been applied to corrosion fatigue fracture surface studies. The oxide phase on the corrosion fatigue fracture surfaces was Fe(3)O(4) (magnetite) in all the specimens. Small amounts of sulfur (typically about 3 atomic percent) were present in the oxide film mainly as FeS(2). In specimens showing the highest crack growth rates, the amount of sulfur was about doubled near the crack tip compared to the value obtained further behind the crack tip in the middle of the fracture surface. It was possible to locate the crack- tip condition on the high-temperature Pourbaix diagram in the area where both magnetite and FeS(2) are stable phases. The anticipated crack-tip conditions are that the corrosion potential is about -500 mV(SHE) or less and the pH value is neutral or slightly acid. XPS and AES analyses of the corrosion fatigue fracture surfaces were found to reveal the possible water chemistry impurities during the corrosion fatigue tests, like Cl, Na, Ca, etc.

NUREG/CR-4865: THE SOLUBILITY OF ELECTRODEPOSITED TC(IV) OXIDES. MEYER,R.E.; ARNOLD,W.D.; CASE,F.I. Oak Ridge National Laboratory. July 1987. 31pp. 8710010154. ORNL-6374. 42882:060.

Solubilities of electrodeposited Tc(IV) oxides have been determined in solutions of NaCl, HCl, and synthetic groundwater in the pH range 0 to 10. Oxides were electrodeposited onto platinum electrodes, and the oxide-covered platinum was then placed into a small stirred cell. Solubilities were determined by counting the beta radiation of (99)Tc in the solution in the stirred cell. The solubilities are approximately constant in the pH range 3 to 10; the values in this region for deposits from acid solution average (1.32 plus or minus 0.68) x 10(-8) mol/L. For oxides deposited in basic solutions the average is (2.56 plus or minus 0.54) x 10(-9) mol/L. The electrodeposited oxides are hydrated and experiments show that they have the composition TcO(2).nH(2)O where n has an average value of 1.63 plus or minus 0.28. The oxides appear to vary in structure and/or composition depending on their method of preparation and history. Solubility products between 10(-32) and 10(-33) can account for most of the data. These data can be used to estimate solubilities for cases where solubility limits transport of technetium in reducing high-level waste repository environments.

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

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

NUREG/CR-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/CR-4870: AN EVALUATION OF THE EFFECTS OF DESIGN DETAILS ON THE CAPACITY OF LWR STEEL CONTAINMENT BUILDINGS. GREIMANN,L.; FANOUS,F.; ROGERS,J.; et al. Sandia National Laboratories. May 1987. 96pp. 8709090402. SAND87-7066. 42569:238.

As part of their work with severe accident loadings, the Containment Integrity Division at Sandia National Laboratories has been conducting a program to evaluate the performance of containment buildings with internal pressure. Sandia has suggested using a strain criterion for predicting rupture of steel containments: the strain limit used to correlate with failure must be consistent with the level of detail included in the analytical model. If detailed 3D models are developed, the strain limit may be as large as the material's ultimate strain. This criterion is applied to strains calculated with a shell theory which includes material and geometric nonlinearities and is applied to the actual containment. In this work, drawings from nine steel containments were studied and several significant strain concentration regions were identified and classified as: eccentricities in stiffener patterns around penetrations, eccentricities in the shell middle surface, flat plate covers used on spare penetrations, containment base connection details, and ellipsoidal and torispherical heads and covers. The 1:8-scale model was analyzed using the shell finite element in ANSYS, and the results agreed well with Sandia's analytical results and the experimental results. Examples of each classification were analyzed by finite element and/or simplified equations. In the case of middle surface eccentricities, the strains are self-limiting. Although flat plates have primary bending strains, they are typically designed so as not to control. Bolts in the base connection have primary strains and may control.

NUREG/CR-4871: RESULTS FROM THE DCH-1 EXPERIMENT. TARBELL,W.W.; ROSS,J.W.; ARELLANO,F.E.; et al. Sandia National Laboratories. June 1987. 67pp. 8707300035. SAND86-2483. 41983:063.

The DCH-1 (Direct Containment Heating) test was the first experiment performed in the Surtsey Direct Heating Test Facility. The test involved 20 kg of molten core debris simulant ejected into a 1:10 scale model of the Zion reactor cavity. The melt was produced by a metallothermic reaction of iron oxide and aluminum powers to yield molten iron and alumina. The cavity model was placed so that the emerging debris propagated directly upwards along the vertical centerline of the chamber. Results from the experiment showed that the molten material was ejected from the cavity as a cloud of particles and aerosol. The dispersed debris caused a rapid pressurization of the 103-m(3) chamber atmosphere. Peak pressure from the six transducers ranged from 0.09 to 0.13 MPa (13.4 to 19.4 psig) above the initial value in the chamber. Posttest debris collection yielded 11.6 kg of material outside the cavity, of which approximately 1.6 kg was attributed to the uptake of oxygen by the iron particles. Mechanical sieving of the recovered debris showed a lognormal size distribution with a mass mean size of 0.55 mm. Aerosol measurements indicated a substantial portion (2 to 16%) of the ejected mass was in the size range less than 10 micrometer aerodynamic equivalent diameter.

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

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

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

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

NUREG/CR-4876: SILVER-INDIUM-CADMIUM CONTROL ROD BEHAVIOR AND AEROSOL FORMATION IN SEVERE REACTOR ACCIDENTS. PETTI,D.A. EG&G Idaho, Inc. (subs. of EG&G, Inc.). April 1987. 185pp. 8706130364. EGG-2501. 42154:286.

Ag-In-Cd control rod behavior and aerosol formation in severe reactor accidents are examined. Control rod behavior in in-pile and out-of-pile experiments is reviewed. A mechanistic model named VAPOR is developed that calculates downward relocation and simultaneous vaporization behavior of Ag-In-Cd alloy expected after control rod failure. Although cadmium is found to be the most volatile constituent of the alloy, all calculations using VAPOR predict that rapid relocation of the alloy down to cooler portions of the core results in a small release for all three control rod alloy vapors. Potential aerosol formation mechanisms in severe reactor accidents are reviewed. Models for homogeneous, ion-induced, and heterogeneous nucleation are investigated. Examination of these models indicates that aerosol formation occurs in three stages: (1) ion-induced nucleation causes aerosol generation; (2) ion-induced and heterogeneous nucleation operate simultaneously; and (3) heterogeneous nucleation is the dominant mechanism of gas-to-particle conversion until equilibrium is established. Preliminary results of the control rod and aerosol behavior observed in PBF Test SFD 1-4 are presented. Control rod material release in the test is compared to VAPOR predictions. Conclusions from this work are presented, and their impact on source term estimation is assessed.

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

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

experiments the maximum stress was less than that predicted by the IWB-3640 Source Equations. The greatest difference was 8 percent. For the flaw sizes evaluated in this study, an overlay design based on actual flaw dimensions would be considered a "Limited Service" overlay, suitable only for short-term plant operation, not to exceed one fuel cycle.

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

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

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

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

NUREG/CR-4884: INTERPRETATION OF BIOASSAY MEASUREMENTS. LESSARD,E.T.; YIHUA,X.; SKRABLE,K.W.; et al. Brookhaven National Laboratory. July 1987. 914pp. 8710080369. BNL-NUREG-52063. 42991:133.

This is a comprehensive manual describing how to compute intakes from both in-vivo and in-vitro bioassay measurements. To date, interpretations of intake have been inconsistent, particularly in the early phases after an accidental intake. This manual is aimed at completely describing a consistent approach and instructing others on how to compute intakes and committed organ dose equivalents. Tables for the interpretation of bioassay results are compiled for several hundred radionuclides. Measurements which employ a whole-body counter, a thyroid counter, a lung counter, or measurements on excreta can be converted into estimates of intake based on the tables presented in the appendices. The values in the tables were determined by using lung, gastrointestinal tract and systemic retention models published by the International Commission on Radiological Protection (ICRP79). In a few cases, pseudo-retention functions, organ retention functions, and excretion functions were used to generate the tabulated values. The biological and radiological input parameters are included in an appendix, and a description of the mathematical approach that was used to derive the tabulated data is included in the methods section. Calculations for various particle sizes are addressed along with methods to interpret multiple or continuous exposures. Examples of use are based on actual bioassay measurements following accidental intakes, including tritium, Mn-54, Co-60, Sr-90, Nb-95, radioiodines, Cs-137, Ce-141, Ce-144, U-233, U-Nat, and Am-241.

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

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

NUREG/CR-4886: ANALYSIS OF THE NESDIP2 AND NESDIP3 RADIAL SHIELD AND CAVITY EXPERIMENTS. MAERKER,R.E. Oak Ridge National Laboratory. May 1987. 57pp. 8708060387. ORNL/TM-10389. 42072:274.

Discrete ordinates calculations were made of a series of measurements performed using the NESTOR reactor at AEE Winfrith. These measurements are part of the NESDIP experimental program designed to benchmark methods and data commonly used in interpreting pressure vessel surveillance dosimetry placed at either in-vessel or ex-vessel locations. Results obtained using the LEPRICON procedures and updated ELXSIR cross sections indicate agreement to within 10 percent with measured threshold dosimeter activities behind various components of a radial shield containing up to 25 cm of water and 24 cm of steel. Significant discrepancies in the energy range between 0.1 and 2 MeV were found to exist with measurements made on the centerline of a 29-cm-wide cavity and at vertical locations considerably above the centerline of a 21-cm-wide cavity, however. A logical explanation of the latter discrepancy is the failure of two independent two-dimensional calculations to properly treat the three- dimensional effects of vertical cavity streaming. No satisfactory explanation has yet surfaced for the first discrepancy, however, which is more important of the two in possibly affecting the interpretation of ex-vessel, low-threshold (100 KeV) dosimetry.

NUREG/CR-4887: POSTCON:A POSTPROCESSOR AND UNIT CONVERSION PROGRAM FOR THE CONTAIN COMPUTER CODE. WASHINGTON,K.E. Sandia National Laboratories. May 1987. 230pp. 8711100407. SAND87-0562. 43340:308.

The numerical predictions from use of the CONTAIN severe reactor accident containment analysis computer code normally take the form of massive quantities of output data. The purpose of the POSTCON computer program is to provide an easy-to-use and efficient method for examining such results. In this report the capabilities of POSTCON are described and instructions for the use of the program are given. In order to clarify the discussion of the input options and output format, several examples are presented including actual input, output, and vector files. The summary sections of this document serve as a user's manual and can be consulted for the construction of simple POSTCON input files. The detailed sections and Appendix A

serve as a comprehensive reference manual that can be consulted for advanced POSTCON include extraction of all transient data from CONTAIN binary plot files, multiple file handling, a flexible unit conversion system, snapshot and histogram options, automatic pagination and labeling of tables, and vector output files for plot program interfacing.

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

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

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

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

NUREG/CR-4891: PROPERTIES OF REACTOR FUEL ROD MATERIALS AT HIGH TEMPERATURES.Final Summary Report - Severe Core Damage Property Tests Program. PRATER,J.T.; COURTRIGHT,E.L. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1987. 45pp. 8708240077. PNL-6164. 42316:110.

This report summarizes work sponsored by the U.S. Nuclear Regulatory Commission Division of Accident Evaluation to investigate those physical properties that are needed to predict the behavior of fuel-rod assemblies during a loss-of-coolant accident. The results include a determination of the oxidation kinetics of Zircaloy and Zircaloy-uranium oxide mixtures in steam and steam-hydrogen gas mixtures at 1300 to 2400 degrees C, viscosity measurements of zirconium-oxide mixtures at 1800 to 2100 degrees C, an estimate of the heat of reaction for the dissolution of uranium oxide by molten zirconium at 2000 degrees C, and thermal diffusivity measurements on prereacted Zircaloy-uranium oxide mixtures at 800 to 1500 degrees C.

NUREG/CR-4892: A STUDY OF THE EFFECTS OF PENETRATION FRAMING ON STEEL CONTAINMENT BUCKLING CAPACITY. BAKER,W.E.; BUTLER,T.A. Los Alamos National Laboratory. May 1987. 59pp. 8707300287. LA-10977-MS. 41981:309.

Polycarbonate cylinders modeling steel containment structures were tested to study the effects of different framing designs around large penetrations on the static buckling capacity of containments. Two of the four models had equipment hatch penetrations and two had personnel airlock penetrations. Both types of models were tested with axial and shear loads as framing was incrementally added. Results indicate that, for the models constructed of polycarbonate, buckling is influenced minimally with added framing. Numerical results support the experimental results. Extrapolation of the results to containment constructed under field conditions with prototypic steel materials is discussed and further testing is recommended.

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

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

NUREG/CR-4896: CONTAINMENT LOADS DUE TO DIRECT CONTAINMENT HEATING AND ASSOCIATED HYDROGEN BEHAVIOR.Analysis And Calculations With The CONTAIN Code. WILLIAMS,D.C.; BERGERON,K.D.; CARROLL,D.E.; et al. Sandia National Laboratories. May 1987. 164pp. 8708060351. SAND87-0633. 42071:261.

One of the most important unresolved issues governing risk in many nuclear power plants involves the phenomenon called direct containment heating (DCH), in which it is postulated that molten corium ejected under high pressure from the reactor vessel is dispersed into the containment atmosphere, thereby causing sufficient heating and pressurization to threaten containment integrity. Models for the calculation of potential DCH loads have been developed and incorporated into the CONTAIN code for severe accident analysis. Using CONTAIN, DCH scenarios in PWR plants having three different representative containment types have been analyzed: Surry (subatmospheric large dry containment), Sequoyah (ice condenser containment), and Bellefonte (atmospheric large dry containment). A large number of parameter variation and phenomenological uncertainty studies were performed. Response of DCH loads to these variations was found to be quite complex; often the results differ substantially from what has been previously assumed concerning DCH. Containment compartmentalization offers the potential of greatly mitigating DCH loads relative to what might be calculated using single-cell representations of containments, but the actual degree of mitigation to be expected is sensitive to many uncertainties. Dominant uncertainties include hydrogen combustion phenomena in the extreme environments produced by DCH scenarios, and factors which affect the rate of transport of DCH energy to the upper containment. The importance of hydrogen behavior is partly due to the fact that most of the metallic content of the dispersed corium is calculated to react rapidly with steam in the oxygen-starved lower containment; hence, the immediate energy release is reduced (relative to oxygen reactions) but large quantities of hydrogen are rapidly generated. In addition, DCH loads can be aggravated by rapid blowdown of the primary system, co- dispersal of moderate quantities of water with the debris, and quenching of de-entrained debris in water; these factors act by increasing steam flows which, in turn, accelerates energy transport. Assessment of the actual contribution of DCH scenarios to plant risk would require substantial additional work, but it may be noted that containment-threatening loads were calculated for a substantial portion of the scenarios treated for some of the plants considered.

NUREG/CR-4897: LOW-LEVEL WASTE SOURCE TERM EVALUATION.Review of Published Modeling And Experimental Work,And Presentation Of Low-Level Waste Source Term Modeling Framework And Preliminary Model Development. SULLIVAN,T.; KEMPF,C.R. Brookhaven National Laboratory. February 1987. 105pp. 8708060330. BNL-NUREG-52066. 42072:065.

This report contains information on the efforts performed to date on the Low-Level Waste Source Term Evaluation Project, the objective of which is development of a model to predict radionuclide release rates from a low-level waste disposal unit. The approach for model development has been based on a compartmentalized scheme focused on the four major processes of water flow, container degradation, waste leaching and waste radionuclide transport to the trench boundaries. This stage of the project is focused primarily on modeling release rates from shallow land burial as currently practiced. Research efforts to this point include characterization work (of burial trenches themselves, of soils and structural features, and of waste forms and containers), review of published modeling work, review of several waste package performance system models, and development of original container degradation and waste leaching models. Characterization of the wastes, containers, and of the site (trench soils and structure) has been based on the premise that NRC guidance has been put into effect.

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

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

NUREG/CR-4900 V01: COMPONENT FRAGILITY RESEARCH PROGRAM.Phase I Demonstration Tests.Summary Report. HOLMAN,G.S.; CHOU,C.K. Lawrence Livermore National Laboratory. SHIPWAY,G.D.; et al. Wyle Laboratories. August 1987. 210pp. 8709090542. UCID-21002. 42563:001.

This report describes tests performed in Phase I of the NRC Component Fragility Research Program. The purpose of these tests was to demonstrate procedures for characterizing the seismic fragility of a selected component, investigating how various parameters affect fragility, and finally using test data to develop practical fragility descriptions suitable for application in probabilistic risk assessments. A three-column motor control center housing motor controllers of various types and sizes as well as relays of different types and manufacturers was subjected to seismic input motions up to 2.5g zero period acceleration. To investigate the effect of base flexibility on the structural behavior of the MCC and on the functional behavior of the electrical devises, multiple tests were performed on each of four mounting configurations: four bolts per column with internal diagonal bracing, and two bolts per column with no top or internal bracing. Device fragility was characterized by contact chatter correlated to local in-cabinet response at the device location. Seismic capacities were developed for each device on the basis of local input motion required to cause chatter; these results were then applied to develop probabilistic fragility curves for each type of device, including estimates of the "high-confidence low probability of failure" capacity of each.

NUREG/CR-4900 V02: COMPONENT FRAGILITY RESEARCH PROGRAM.Phase I Demonstration Tests.Appendices. HOLMAN,G.S.; CHOU,C.K. Lawrence Livermore National Laboratory. SHIPWAY,G.D.; et al. Wyle Laboratories. August 1987. 316pp. 8709090300. UCID-21002. 42573:006.

See NUREG/CR-4900,V01 abstract.

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

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

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

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

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

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

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

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

NUREG/CR-4904: INVESTIGATION OF STEEL CONTAINMENT BUCKLING FROM DYNAMIC LOADS. BUTLER,T.A.; BAKER,W.E.; BABCOCK,C.D. Los Alamos National Laboratory. May 1987. 54pp. 8708170218. LA-10985-MS. 42171:076.

Buckling of free-standing nuclear steel containment buildings from dynamic base excitation was investigated in a combined experimental/numerical program. A polycarbonate scale model of a containment building was excited with scaled earthquake transients and single-frequency harmonic transients to determine the peak base acceleration levels required to induce buckling. Buckling was identified using recorded signals from strain gages and accelerometers, with high-speed video records, and by audibility. Experimental results are compared with numerical results obtained using a freezing-in-time technique. Results indicate that, depending on the criterion used, predictions can range from being quite conservative to being unconservative. It was also found that excitation of higher steel harmonics may influence the buckling capacity of containments. Results of this study are considered preliminary in that additional tests need to be performed using models fabricated from steel rather than polycarbonate.

NUREG/CR-4905: DETONABILITY OF H2-AIR-DILUENT MIXTURES. TIESZEN,S.R.; SHERMAN,M.P.; BENEDICK,W.B.; et al. Sandia National Laboratories. June 1987. 215pp. 8707300370. SAND85-1263. 41985:133.

This report describes the Heated Detonation Tube (HDT). Detonation cell width and velocity results are presented for H(2)-air mixtures, undiluted and diluted with CO(2) and H(2)O for a range of H(2) concentration, initial temperature and pressure. The results show that the addition of either CO(2) or H(2)O significantly increases the detonation cell width and hence reduces the detonability of the mixture. The results also show that the detonation cell width is reduced (detonability is increased) for increased initial temperature and/or pressure.

NUREG/CR-4909: MELPROG-PWR/MOD0:A MECHANISTIC CODE FOR ANALYSIS OF REACTOR CORE MELT PROGRESSION AND VESSEL ATTACK UNDER SEVERE ACCIDENT CONDITIONS. CAMP,W.J.; YOUNG,M.F.; TOMKINS,J.L.; et al. Sandia National Laboratories. April 1987. 156pp. 8711100095. SAND85-0237. 43339:008.

Since the Three Mile Island accident, increased emphasis has been placed on the analysis of severe reactor accidents since such accidents represent the dominant public risk from nuclear reactors. The U.S. Nuclear Regulatory Commission has made the development of mechanistic models for severe accident progression a major priority. The purpose of these models is to provide detailed, best-estimate, coupled analyses of all the major phenomena involved in the course of the accident. To meet this objective, the MELPROG computer code is being developed. The initial version of this code, MELPROG-PWR/MOD0, is described in this report. The purpose of MELPROG is to provide at any time during an accident sequence a description of (1) the state of the reactor core and surrounding in-vessel environment, and (2) the disposition of core materials (in particular, fission products) contained within the reactor coolant system boundary. MELPROG is coupled to the TRAC-PF1 RCS thermal-hydraulics code to provide an integrated analysis of the behavior of core, vessel, and reactor coolant system during severe accidents. MELPROG treats core degradation and loss of geometry, debris formation, core melting, attack on supporting structures, slumping, melt/water interactions and vessel failure. The key element in MELPROG is the use of detailed modeling for the entire damage progression and failure sequence. Emphasis is also placed on the rates of hydrogen, steam and fission product formation, and transport to containment during damage progression.

NUREG/CR-4910: RELAY CHATTER AND OPERATOR RESPONSE AFTER A LARGE EARTHQUAKE:An Improved PRA Methodology With Case Studies. BUDNITZ,R.J.; LAMBERT,H.E.; HILL,E.E. Future Resources Associates, Inc. August 1987. 204pp. 8708270393. 42390:117.

The report addresses methodological weaknesses in the PRA systems analysis used for studying post-earthquake relay chatter and for quantifying human response under high stress. An improved PRA methodology for relay-chatter analysis is developed, and its use is demonstrated through analysis of the Zion-1 and LaSalle-2 reactors as case studies. This demonstration analysis is intended to show that the methodology can be applied in actual cases. The analysis relies on SSMRP-based methodologies and data bases. For both Zion-1 and LaSalle-2, it is assumed that loss of offsite power (LOSP) occurs after a large earthquake and that there are no operator recovery actions. The report also presents an improved PRA methodology for quantifying operator error under high-stress conditions such as after a large earthquake. Single-operator error rates are developed, and a case study involving an 8-step procedure (establishing feed-and-bleed in a PWR after an earthquake-initiated accident) is used to demonstrate the methodology.

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

The age of ground water is the length of time that the water has been isolated from the atmospheric portion of the hydrologic cycle. It is a theoretical concept only, because all ground water to some extent is a mixture of waters of different ages. In simple systems, however, relative ages of ground water from different portions of an aquifer can be determined by different methods, and the dates obtained are commonly in accord with each other and reflect systematic increases of water ages in downgradient directions. At least nine independent methods can

be used to approximate ground-water ages. The methods vary widely in precision but all give useful information. In complex ground-water systems, as many dating methods as possible should be used. Discordant "dates" will result which, when properly interpreted, will not give a single water age but will give valuable information concerning the hydrodynamics of the ground-water system. Dating methods which use isotopic and other geochemical techniques will read the actual history of the water and will give direct information on average ground-water conditions over long periods of time. If these periods exceed several hundred years, geochemical methods which use past conditions to predict the future are superior to hydrodynamic methods which use an extrapolation of short-term data to predict long-term hydrogeologic conditions.

NUREG/CR-4913: ROUND-ROBIN PRETEST ANALYSES OF A 1:6-SCALE REINFORCED CONCRETE CONTAINMENT MODEL SUBJECT TO STATIC INTERNAL PRESSURIZATION. CLAUSS,D.B. Sandia National Laboratories. May 1987. 532pp. 8710050162. SAND87-0891. 42908:037.

Analyses of a 1:6-scale reinforced concrete containment model that will be tested to failure at Sandia National Laboratories in the spring of 1987 were conduced by the following organizations in the United states and Europe: Sandia National Laboratories (USA), Argonne National Laboratory (USA), Electric Power Research Institute (USA), Commissariat a L'Energie Atomique (France), HM Nuclear Installations Inspectorate (U.K.), Comitato Nazionale per la ricera e per lo sviluppo dell'Energia Nucleare e delle Energie Alternative (Italy), U.K. Atomic Energy Authority, Safety & Reliability Directorate (U.K.), Gesellschaft fuer Reaktorsicherheit (FRG), Brookhaven National Laboratory (USA), Central Electricity Generating Board (U.K.). Each organization was supplied with a standard information package, which included construction drawings and actual material properties for most of the materials used in the model. Each organization worked independently using their own analytical methods. This report

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

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

NUREG/CR-4919: FIELD TESTING OF BENTONITE AND CEMENT BOREHOLE PLUGS IN GRANITE. KIMBRELL,A.F.; AVERY,T.S.; DAEMEN,J.J. Arizona, Univ. of, Tucson, AZ. July 1987. 342pp. 8708110501. 42123:253.

This technical report describes in-situ flow tests on bentonite and cement borehole plugs installed in granite, and laboratory back-up experiments on similar plugs. Prior to sealing, the hydraulic conductivity of the boreholes is tested. These measurements, together with core and borehole videologs, permit the selection of seal test intervals. Commercial expansive cement and standard waterwell sealing bentonite products and emplacement procedures are used. Transient (short-term) and steady-state (long-term) tests determine the sealing performance of the plugs under various stress conditions. Laboratory and field experiments confirm the difficulty of obtaining even moderately accurate performance values for bentonite plugs as installed here, due to the simultaneous saturation, swelling, and consolidation. Nevertheless, conventional installation of readily available seals can provide adequate borehole seals (i.e., seals with a hydraulic conductivity of about the same magnitude as that of intact granite). Bentonite plugs of this type are heterogeneous and weak. Improved testing procedures and analyses are needed if actual conductivity values are to be obtained in a reasonable testing time (e.g., months). Laboratory investigations on failure mechanisms in cementitious and bentonitic plugs, not covered in this research study, are needed to identify and prevent plug failure.

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

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

NUREG/CR-4922: STEAM SEPARATOR MODELING FOR VARIOUS NUCLEAR REACTOR TRANSIENTS. PAIK,C.Y.; MULLEN,G.; KNOESS,C.; et al. Massachusetts Institute of Technology, Cambridge, MA. June 1987. 226pp. 8706240352. EPRI NP-5272. 41439:289.

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

NUREG/CR-4925: FISSION PRODUCT BEHAVIOR DURING THE PBF SEVERE FUEL DAMAGE TEST 1-1. HARTWELL,J.K.; PETTI,D.A.; HAGRMAN,D.L.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). May 1987. 672pp. 8708140041. EGG-2462. 42159:038.

In response to the accident at Three Mile Island Unit 2, the United States Nuclear Regulatory Commission initiated a series of Severe Fuel Damage (SFD) tests that were performed in the Power Burst Facility (PBF) at the Idaho National Engineering Laboratory to obtain data necessary to understand (a) fission product release, transport, and deposition; (b) hydrogen generation; and (c) fuel/cladding material behavior during degraded core accidents. During the second experiment of this series, the SFD Test 1-1, real-time release and transport data of certain fission products were obtained from on-line gamma spectroscopy measurements. Liquid and gas effluent grab samples were collected at selected periods during the test transient. Additional information was obtained from steamline deposition analysis. From these and other data, fission product release rates and total release fractions are estimated and compared with predicted release behavior using current models; fission product distributions and a mass balance are summarized; and certain probable chemical forms are predicted for iodine, cesium, and tellurium. Analysis indicates that volatile release from fuel is strongly influenced by parameters other than fuel temperature; these are fuel/fission product morphology, fuel and cladding oxidation state, extent of fuel liquefaction and shattering. Tellurium release behavior is examined relative to the extent of zircaloy cladding oxidation.

NUREG/CR-4926: AN ASSESSMENT OF LOSS-OF-CONTAINMENT POTENTIAL BECAUSE OF KNUCKLE BUCKLING FOR 4:1 STEEL CONTAINMENT HEADS. BENNETT,J.G. Los Alamos National Laboratory. April 1987. 224pp. 8712010143. LA-10972-MS. 43485:153.

4:1 diameter-to-height rise ratio torispherical nuclear steel containment head geometries were investigated experimentally and analytically for effects of knuckl[illegible] from internal pressure. The particular focus of the [illegible] to assess the loss-of- containment potential. Sc th[illegible] were carried to rupture when feasible. Results o[illegible]ale tests and two large-scale (1/8) tests are reported. Significant ratios of knuckle buckling to design pressures were found (1.6-2.4). In addition, very large rupture pressure to design pressure ratios (7-7.5) indicate adequate reserve margin for loss-of-containment for these geometries and materials.

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

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

NUREG/CR-4930: CRACK-ARREST BEHAVIOR IN SEN WIDE PLATES OF QUENCHED AND TEMPERED A 533 GRADE B STEEL TESTED UNDER NONISOTHERMAL CONDITIONS. NAUS,D.J.; BASS,B.R.; PUGH,C.E.; et al. Oak Ridge National Laboratory. August 1987. 210pp. 8711160022. ORNL-6388. 43361:147.

The first series of six wide-plate crack-arrest tests (WP-1 Series) are discussed in this report. Each test utilized a 1-m x 1-m x 0.1-m-thick single-edge notch specimen (a/w = 0.2), fabricated from A533 grade B class 1 steel, that was subjected to a linear thermal gradient along the plane of crack propagation. The tests were conducted at the National Bureau of Standards (Gaithersburg) and were designed to provide fracture-toughness measurements at temperatures approaching or above the onset of the Charpy upper-shelf regime, in a rising toughness region and with an increasing driving force. Results from this series of tests produced crack-arrest toughness values well above the limit recognized by current ASME Code guidelines(220 MPa$\cdot\sqrt{m}$) with arrest occurring at 52 degrees C to 115 degrees C above the material RT(NDT) (-23 degrees C). The fracture data support (1) the use of current LEFM concepts to analyze cleavage run-arrest events, (2) the treatment of cleavage run-arrest and ductile fracture modes as separate events, and (3) the fact that cleavage arrest occurs above the ASME limit.

NUREG/CR-4931: RESPONSE OF CENTRIFUGAL AND AXI-VANE BLOWERS TO LARGE PRESSURE TRANSIENTS. GREGORY,W.S.; SMITH,P.R. Los Alamos National Laboratory. June 1987. 619pp. 8709090188. LA-11019-MS. 42561:102.

The effect of large pressure pulses on the operation of centrifugal and axi-vane blowers of the types found in ventilation systems used in the mining and nuclear industries was investigated using the Los Alamos National Laboratory/New Mexico State University fluid dynamics test facility. Three blowers were tested for both quasi-steady and transient pressures: a 24-in. and a 12-in. centrifugal blower and a 33-in. axi-vane blower were subjected to pressure pulses at their exhaust and inlet, which caused backflow and outrunning flow, respectively. Performance curves were obtained for the first, second, and fourth quadrants.

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

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

NUREG/CR-4937: INVESTIGATION OF THE MEERS FAULT IN SOUTHWESTERN OKLAHOMA. LUZA,K.V.; MADOLE,R.F.; CRONE,A.J. Oklahoma, Univ. of, Norman, OK. August 1987. 65pp. 8709150280. 42680:291.

The Meers Fault is part of a major system of NW-trending faults that form the boundary between the Wichita Mountains and the Anadarko basin in southwestern Oklahoma. The Meers Fault is exposed at the surface over a length of at least 26 km; it strikes N 60 degrees W and offsets Permian conglomerate and shale. The fault is consistently down to the south, with a maximum relief of 5 m near the center of the fault trace. Quaternary stratigraphy and 10 (14)C age dates constrain the age of the last movement of the Meers Fault. The last movement postdates the Browns Creek Alluvium, late Pleistocene to early Holocene, and predates the East Cache Alluvium, 100-800 yr B.P. Fan alluvium, produced by the last fault movement, buried a soil that dates between 1,400 and 1,100 yr B.P. Two trenches excavated across the scarp near Canyon Creek document the near-surface deformation and provide some information on recurrence. Flexing and warping was the dominant mode of deformation. The stratigraphy in both trenches indicates one surface-faulting event, which implies a lengthy recurrence interval for surface faulting on this part of the fault. Two samples that postdate the last fault movement yielded (14)C ages between 1,600 and 1,300 yr B.P. These dates are in agreement with dates obtained from the soil buried by fault-related fan alluvium.

NUREG/CR-4938: OCCUPATIONAL RADIATION EXPOSURES ASSOCIATED WITH ALTERNATIVE METHODS OF LOW-LEVEL WASTE DISPOSAL. HERRINGTON,W.N.; HARTY,R.; MERWIN,S.E. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1987. 90pp. 8706240313. PNL-6217. 41450:176.

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

NUREG/CR-4941: THE APPLICATION OF VALUE-IMPACT ANALYSIS TO USI A-45.Summary Report Of UCLA Studies On Value-Impact Analysis In Relation To USI A-45. CAVE,L.; KASTENBERG,W.E. Sandia National Laboratories. * California, Univ. of, Los Angeles, CA. September 1987. 226pp. 8711160040. SAND87-7116. 43363:045.

Value-impact analyses of potential resolutions to USI A-45 were an essential part of the program. This report describes: (1) development of a V-I analysis methodology for the example plant case studies, (2) the main sources of uncertainty in V-I analyses and their treatment in USI A-45, (3) the investigation of additional avertable costs such as economic impact of nuclear moratoria, (4) the inter- relation of V-I analysis and possible use of quantified design objectives for the probability of core melt and other safety parameters, and (5) the application of V-I analysis to regulatory decisions in relation to specific problems.

NUREG/CR-4942: EQUIPMENT OPERABILITY DURING STATION BLACKOUT EVENTS. JACOBUS,M.J.; NICOLETTE,V.F.; PAYNE,A.C. Sandia National Laboratories. October 1987. 51pp. 8711030077. SAND87-0750. 43243:285.

This report is an initial look at a possible analysis method for determining if equipment expected to operate during and after a station blackout can be "reasonably" expected to survive environments generated as a result of the station blackout. The general conclusion is that, given the current qualification levels of representative equipment, "reasonable" assurance of operability should be demonstrable for most equipment in most locations.

NUREG/CR-4943: PREPARATION OF DESIGN SPECIFICATIONS AND DESIGN REPORTS FOR PUMPS,VALVES,PIPING,AND PIPING SUPPORTS USED IN SAFETY-RELATED PORTIONS OF NUCLEAR POWER PLANTS. RODABAUGH,E.C.; MOORE,S.E. Oak Ridge National Laboratory. June 1987. 78pp. 8710060234. ORNL/TM-10425. 42922:217.

Section III of the ASME Boiler and Pressure Vessel Code requires the preparation of Design Specifications and Design Reports as part of the design process leading to construction of a nuclear power plant, in compliance with provisions of Title 10 of the Code of Federal Regulations (10 CFR). Guidelines for preparing this documentation are contained in nonmandatory Appendixes B and C of the ASME Code. This report gives an in-depth review of the ASME Code requirements and guidance, beginning with the first edition of the Code in 1963 through the 1983 editions, Summer 1985 Addenda. Recommendations for substantial revisions to the Code are presented based on the authors' experience in conducting design documentation audits of pumps, valves, piping, and piping supports for nuclear power plants undergoing NRC review for Operating Licenses. It is concluded that adequate Design Specifications and Design Reports are absolutely necessary for the normal operating life of a plant and are vital if plant life extension is planned.

NUREG/CR-4944: CONTAINMENT PENETRATION ELASTOMER SEAL LEAK RATE TESTS. BRIDGES,T.L. Sandia National Laboratories. BRIDGES,T.L. EG&G Idaho, Inc. (subs. of EG&G, Inc.). July 1987. 58pp. 8712020035. SAND87-7118. 43519:282.

Tests were performed on three elastomer seal designs commonly used for nuclear plant containment mechanical penetrations. The objective of this research project is to obtain an understanding of the integrity and leakage behavior of these seal designs under severe accident temperature and pressure conditions. The three designs tested and the seal materials used in the tests were: (a) double tongue-and-groove design with silicone rubber seals, (b) double-O-ring design with neoprene and ethylene-propylene (EPDM) seals, and (c) double gumdrop design with neoprene and EPDM seals. The effects of thermal aging and angular rotations of flange mating surfaces were determined. The test results provide information required to characterize the leakage behavior of penetrations under severe accident conditions.

NUREG/CR-4945: SUMMARY OF THE SEMISCALE PROGRAM (1965-1986). LOOMIS,G.G. EG&G Idaho, Inc. (subs. of EG&G, Inc.). July 1987. 273pp. 8708140083. EGG-2509. 42161:001.

This document summarizes significant results from the Semiscale Program, which examined pressurized water reactor (PWR) safety issues from 1965 to 1986. Most of these issues were related to plant response during loss-of-coolant accidents and operational transients. The Semiscale program utilized a series of non-nuclear, scaled, PWR plant simulators to provide thermal-hydraulic data at prototypical pressures and temperatures for a wide range of nuclear safety issues. Presented are: a his-

torical perspective of the Semiscale Program relative to reactor safety with a catalog of the Semiscale experimental facilities and data bases, the relationship of Semiscale results to 10 CFR Part 50 (Appendix K), the impact of Semiscale results on scaling experimental results to full size operating plants, a summary of safety issues that were addressed in Semiscale testing as they arouse throughout the operational lifetime of Semiscale, the contributions of the Semiscale Program to safety technology, a description of phenomena observed during Semiscale testing, the impact of Semiscale data on code development and assessment, and finally, major conclusions and accomplishments of the Semiscale program.

NUREG/CR-4946: DAVIS-BESSE UNCERTAINTY STUDY. DAVIS,C.B. EG&G Idaho, Inc. (subs. of EG&G, Inc.). August 1987. 83pp. 8709090483. EGG-2510. 42564:099.

The uncertainties of calculations of loss-of-feedwater transients at Davis-Besse Unit 1 were determined to address concerns of the U.S. Nuclear Regulatory Commission relative to the effectiveness of feed and bleed cooling. Davis-Besse Unit 1 is a pressurized water reactor of the raised-loop Babcock & Wilcox design. A detailed, quality- assured RELAP5/MOD2 model of Davis-Besse was developed at the Idaho National Engineering Laboratory. The model was used to perform an analysis of the loss-of-feedwater transient that occurred at Davis- Besse on June 9, 1985. A loss-of-feedwater transient followed by feed and bleed cooling was also calculated. The evaluation of uncertainty was based on the comparisons of calculations and data, comparisons of different calculations of the same transient, sensitivity calculations, and the propagation of the estimated uncertainty in initial and boundary conditions to the final calculated results.

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

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

NUREG/CR-4950 V02: THE SHORELINE ENVIRONMENT ATMOSPHERIC DISPERSION EXPERIMENT (SEADEX).Meteorological And Gas Tracer Data. JOHNSON,W.B.; CANTRELL,B.K.; MORLEY,B.M.; et al. SRI International. October 1987. 495pp. 8711180047. 43387:241.

The SEADEX atmospheric dispersion field study was conducted during the period May 28 - June 8, 1972, in northeastern Wisconsin, in the vicinity of the Kewaunee Power Plant on the western shore of Lake Michigan. The specific objectives of SEADEX were to characterize (1) the atmospheric dispersion and (2) the meteorological conditions influencing this dispersion as completely as possible during the test period. This field study included a series of controlled tracer tests utilizing state-of-the-art tracer measurement technology to determine horizontal and vertical dispersion over both land and water. Extensive meteorological measurements were obtained to thoroughly characterize the three-dimensional structure of the atmospheric boundary layer controlling the dispersion process. This volume presents the meteorological and gas tracer data collected during the field study.

NUREG/CR-4951: NEPHROTOXICITY OF URANYL FLUORIDE AND REVERSIBILITY OF RENAL INJURY IN THE RAT. DIAMOND,G.L.; GELEIN,R.M.; MORROW,P.E.; et al. Rochester, Univ. of, Rochester, NY. September 1987. 104pp. 8710060148. 42921:137.

The objective of the study was to examine the severity and duration of renal injury produced in the rat from exposure to low levels of uranyl fluoride (UO(2)F(2)). Rats received multiple i.p. injections of UO(2)F(2) (cumulative dose: 0.66 or 1.32 mg U/kg body wt). Renal injury was characterized primarily by cellular and tubular necrosis of the pars recta of the proximal tubule (S(2) and S(3)), with less severe cellular injury to the thick ascending limb of the loop of Henle and collecting tubule. The injury was apparent early in the dosing phase of the study, at a time when renal uranium levels were between 0.7 - 1.4 ug U/g, and was most severe when the renal uranium burden was between 3.4 - 5.6 ug U/g. Repair of the injury was rapid, with complete restoration within 35 days after the exposure. Associated with the injury were numerous abnormalities in kidney function, including impaired tubular reabsorption, proteinuria and enzymuria, which appear to be temporally related, to variable degrees, to the progression of renal injury. Renal injury preceded and outlasted functional abnormalities as assessed by urinalysis and clearance measurements.

NUREG/CR-4952: EXPERIMENTAL STUDY OF FILLET WELD UNDERCUT EFFECTS ON WELDED TUBING STRUCTURES UNDER CENTRIC AND ECCENTRIC CYCLIC LOADINGS. OKAILY,A.A. Calspan Corp. (subs. Arvin Industries/Franklin Research Center). June 1987. 85pp. 8708170121. 42171:131.

The effects of weld undercut defects on the structural behavior of twelve welded tubing test specimens subjected to centric and eccentric cyclic loadings were evaluated. The test program included the design, fabrication, and testing of one group of six fillet-welded tubing specimens and one group of six reinforced partial-penetration, groove-welded tubing specimens under centric and eccentric loading. Each group consisted of three centrically loaded specimens with 0, 1/32-, and 1/16-in undercuts and three eccentrically loaded specimens with 0, 1/32-, and 1/16-in undercuts. Permanent deflection, d, was recorded after 1, 5, and 10 load cycles at 10% increments of the calculated yield load, P(y), starting at 50% P(y), through the ultimate load, P(u). Permanent deflection ratios, d/do, of a defective specimen versus that of a similar, but sound specimen (i.e., no undercutting) at the same load level and number of cycles were then calculated. Analysis of test results led to several conclusions related to the effects of fillet weld undercut on permanent deflection ratio and ultimate load values of welded tubing structural members under centric and eccentric cyclic loading.

NUREG/CR-4953: CORRELATION OF RADIOIODINE RESUSPENSION WITH TEMPERATURE AT TMI-2. DANIEL,J.A.; DANIEL,E.A.; SETH,E.L. Daniel & Associates, Inc. July 1987. 137pp. 8708040231. DA-TR/8705. 42037:054.

This report addresses the observed long term behavior of radioiodine in specific locations in the TMI-2 Auxiliary and Fuel Handling Buildings, and provides data on the behavior of I-131 at relatively low temperatures, (50 degrees - 85 degrees F) and non- equilibrium conditions, since the building ventilation systems were in operation. This report also discusses the observed effect of changes in the daily concentration of radioiodine due to diurnal temperature cycles, and establishes a numerical relationship between radioiodine concentration and ambient temperature.

NUREG/CR-4954: LONG-TERM PERFORMANCE OF SPENT FUEL WASTE FORMS. MEANS,J.L.; MARKWORTH,A.J.; MCCOY,J.K.; et al. Battelle Memorial Institute, Columbus Laboratories. September 1987. 162pp. 8710090161. BMI-2154. 43009:058.

This report describes the results of a 2-year experimental program to investigate the leaching/dissolution characteristics of

spent fuel in simulated groundwaters. Experiments were conducted to evaluate the leaching of unirradiated UO(2) in simulated groundwaters both in the presence and absence of an external radiation field. Other experiments characterized the dissolution behavior of spent fuel under the same conditions as for UO(2). Leach data were interpreted in terms of rates and mechanisms of radionuclide release, thus providing insight on leach behavior under realistic repository conditions. Uranium release from both unirradiated and irradiated UO(2) appeared to be solubility controlled, with only minor leachate compositional effects on the uranium concentrations. When leached in the presence of a radiation field, unirradiated UO(2) produced elevated uranium concentrations only in the brine leachates. The leaching of both cesium and iodine from spent fuel was rapid and linear with the surface-area-to-volume (S/V) ratio. Similar to cesium and iodine, strontium release was strongly affected by leachate composition, with greater release associated with lower pH. A model describing spent fuel/UO(2) dissolution was developed, complementing the experimental studies.

NUREG/CR-4955: LONG-TERM PERFORMANCE OF CONTAINER MATERIALS FOR HIGH-LEVEL WASTE. MARKWORTH,A.J.; CIALONE,H.J.; MAJUMDAR,B.S.; et al. Battelle Memorial Institute, Columbus Laboratories. November 1987. 426pp. 8712280053. BMI-2155. 43821:037.

This report describes the results of experimental and analytical studies of high-level waste container degradation. Corrosion and hydrogen embrittlement tests were conducted on selected materials to identify environmental and metallurgical factors that promote material degradation, especially stress-corrosion cracking. A major emphasis on overpack materials focused on cast and wrought low-carbon steels. Results of the corrosion work show that, to more completely identify potential failure modes, exposure environments must be further defined. Predictions of pitting rates based on models utilizing nonreactive walls may lead to rejection of carbon steel as a viable overpack material when, on the basis of performance, it may perform satisfactorily. Hydrogen embrittlement was shown to be promoted in regions of microstructural change such as the weld heat-affected zone. These findings show that hydrogen embrittlement is important to container integrity. A small portion of this task was devoted to studying the possible internal corrosion of the canister. It was found that Type 304L stainless steel will likely contain high-level waste glass for the retrieval period and probably the thermal period. Modeling studies focused on general corrosion and pitting corrosion, with the models being extended to account for more realistic conditions. Results show that pit wall reactivity is an important consideration in predicting corrosion rates.

NUREG/CR-4956: SYSTEM PERFORMANCE OF HIGH-LEVEL WASTE PACKAGE COMPONENTS. NICOLOSI,S.L.; KURTH,R.E.; QUAYLE,S.F.; et al. Battelle Memorial Institute. September 1987. 279pp. 8710060258. BMI-2156. 42921:241.

This report presents results of analytical and experimental studies that can be integrated to describe the performance of nuclear waste packages at the system level. Several water chemistry modules were developed, and groundwater-radiolysis studies were performed to examine the production of radiolytic species at the outside surface of waste packages. Uncertainty analysis studies with a UO(2) water chemistry model assessed the importance of input parameters on the variability of calculated solubilities. A shell for a computer code was developed that provides an alternative method for examining the system performance of high-level waste packages. Integral experiments were conducted to examine interactions between components of the waste package and repository, bringing together elements of repository environments with components of spent-fuel waste forms. The system performance investigations showed that radiolysis effects are not significant when the gamma source is shielded by the overpack. It was observed that the production of oxalate is related to the bicarbonate concentration and dose rate. Other modeling studies showed that the dissolution of UO(2) may be influenced significantly by solution pe and chloride concentration.

NUREG/CR-4957: SURVEY OF GEOPHYSICAL TECHNIQUES FOR SITE CHARACTERIZATION IN BASALT,SALT AND TUFF. JONES,G.M.; BLACKEY,M.E.; RICE,J.E.; et al. Weston Geophysical Corp. July 1987. 149pp. 8709110098. 42609:161.

This report surveys various geophysical techniques that would have application during site characterization activities in connection with the high-level-waste storage project. Geophysical techniques may help determine the nature and extent of faulting in the target areas, along with structural information which would be relevant to questions concerning the future integrity of a high-level-waste repository. Following an introductory chapter which describes the variety of geophysical techniques available, subsequent chapters focus on particular geophysical applications to four rock types - basalt, bedded salt, domal salt and tuff - characteristic of the sites originally proposed for site characterization. No one geophysical method can adequately characterize the geological structure beneath any site. A combination of techniques is required. The seismic reflection method, which is generally considered to be the most incisive of the geophysical techniques, has to date provided only marginal information on structure at the depth of the proposed repository at the Hanford, Washington, site, and no useful results at all at the Yucca Mountain, Nevada, site. This result appears to be partially due to geological complexity beneath these sites, but may also be partially attributed to the use of inappropriate acquisition and processing parameters. It appears that to adequately characterize a site using geophysics, modifications will have to be made to standard techniques to emphasize structural details at the depths of interest.

NUREG/CR-4958: IMPACT OF PROPOSED FINANCIAL ASSURANCE REQUIREMENTS ON NUCLEAR MATERIALS LICENSEES. HENDRICKSON,P.; SCOTT,M.J.; MULLEN,M.F.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1987. 136pp. 8710070036. PNL-6233. 42965:018.

The NRC is considering a possible rulemaking that would require certain materials licensees to demonstrate financial ability to clean up accidental releases of radioactive materials. The rulemaking would potentially affect approximately 16,350 NRC and Agreement State licensees. This report was prepared to provide background information and analysis for the potential rulemaking. Specific topics examined in the report include: 1) characteristics of potentially affected licensees, 2) the availability and cost of various financial assurance mechanisms, 3) the financial impacts to licensees (including licensees classified as small businesses) of providing $2M of assurance per licensee and a sliding amount of assurance tied to risk, 4) the cost of administering a financial assurance rule, and 5) overall benefits and costs. Tabular information on past material licensee accidents and cleanup efforts is also included. The financial assurance mechanism that appears to be most suitable for providing the desired financial assurance is one or more newly-formed captive insurance companies. Potential benefits of the rulemaking include administrative time to secure such funds, and the possibility of more rapid cleanup with correspondingly reduced occupational and public exposure to radioactive materials.

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

The Health Physics Society Standing Committee (HPSSC) Working Group on Performance Testing of Extremity Dosimeters has issued a draft of a proposed standard for extremity dosimeters. The draft standard proposes methods to be used for testing extremity dosimetry systems and the performance criterion used to determine compliance. This study evaluates the draft standard's proposed performance criterion (absolute value of B + S less than or equal 0.35, where B is the bias and S is

the standard deviation) against the performance of extremity dosimeter processors. Twenty-one types of extremity dosimeters from 11 processors were irradiated by the Pacific Northwest Laboratory (PNL) to specific dose levels in one or more of seven categories. The processors evaluated the doses and returned the results to PNL for analysis. Approximately 60% of the dosimeters met the performance criterion. Two-thirds of the remaining dosimeters had large biases (ranging from 0.25 to 0.80) but small standard deviations (less than 0.15). Recommendations to improve the biases include providing calibrations (using appropriate irradiation standards and phantoms) followed by another set of performance tests, as well as visiting processors to identify other possible sources of error. It is further recommended that the draft standard be re-evaluated to ensure that it is appropriate for the performance testing of extremity dosimeters.

NUREG/CR-4962: METHODS FOR THE ELICITATION AND USE OF EXPERT OPINION IN RISK ASSESSMENT.Phase I - A Critical Evaluation And Directions For Future Research. MOSLEH,A.; BIER,V.M.; APOSTOLAKIS,G. Pickard, Lowe & Garrick, Inc. August 1987. 85pp. 8709090358. PLG-0533. 42566:268.

The purpose of this work is to critically review and evaluate the elicitation and use of expert opinion in probabilistic risk assessment (PRA) in light of the available empirical and theoretical results on expert opinion use. PRA practice is represented by five case studies selected to represent a variety of aspects of the problem: (1) Assessments of component failure rates and maintenance data, (2) Recent assessments of seismic hazard rates, (3) Assessments of containment phenomenology, (4) Assessments of human error rates, and (5) Accident precursor studies. The review has yielded mixed results. On the negative side, there appears to be little reliance on normative expertise in structuring the process of expert opinion elicitation and use; most applications instead rely primarily on the common sense of the experts involved in the analysis, which is not always an adequate guide. On the positive side, however, there is evidence that expert opinions can in fact be used well in practical settings. Suggestions are given for Phase II work to enhance the applicability and use of appropriate expert opinion methods.

NUREG/CR-4963: BORON CARBIDE - STEAM REACTIONS WITH CESIUM HYDROXIDE AND WITH CESIUM IODIDE AT 1270K IN AN INCONEL 600 SYSTEM. ELRICK,R.M.; SALLACH,R.A.; OUELLETTE,A.L.; et al. Sandia National Laboratories. August 1987. 82pp. 8712010140. SAND87-1491. 43486:017.

Three laboratory scale tests examined the effect of boron carbide, used for the control of some nuclear power reactors, on reactor accident conditions. The tests, conducted in steam at 1270K in an Inconel 600 system, examined the boron carbide - steam system and then that system with an addition of cesium hydroxide or cesium iodide vapor. Extensive reaction was demonstrated between the boron carbide and steam to produce B(2)O(3) and boric acids in a two step kinetic process. Average rates of production for the oxide and acids were calculated for the three tests. These rates are functions of time, temperature, B(4)C geometry, and partial pressure of the steam and the boric acid. Significant reactions were observed between the boric oxide formed on the B(4)C and both the CsOH vapor and the CsI vapor, dissociating the cesium and iodine in the latter case. The cesium reaction product in both cases was probably CsBO(2), with the iodine forming HI. Surface reaction rate constants were estimated for the CsOH and CsI reactions.

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

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

NUREG/CR-4965: TRAC-PF1/MOD1 US/JAPANESE PWR CONSERVATIVE LOCA PREDICTION. GRUEN,G.E.; FISHER,J.E. EG&G Idaho, Inc. (subs. of EG&G, Inc.). November 1987. 81pp. 8712240021. EGG-2513. 43794:197.

This report documents the results of a 200%, double-ended, cold- leg-break, loss-of-coolant-accident (LOCA) calculation using the TRAC- PF1/MOD1 computer code. The reactor system represented a typical United States/Japanese pressurized water reactor with a 15 x 15 fuel bundle arrangement 12-ft long, four loops, and cold-leg Emergency Core Cooling (ECC) Systems. Conservative boundary and initial conditions were used. Reactor power was 102% of the 3250 MWt rated power, decay heat was set to 120% of American Nuclear Society Standard 5.1, highest core lifetime values for power peaking and fuel stored energy were used, and the LOCA occurred simultaneously with a loss of offsite power. Best estimate assumptions were used for the break flow model, fuel rod heat transfer and metal-water reaction correlations, and steady-state fuel temperature profiles. A flow blockage model, having the capability to account for the effects of cladding ballooning or rupturing, was not used. Except for these best estimate assumptions, the boundary and initial conditions were consistent with those used in licensing calculations. Maximum fuel rod temperatures were 1380 K (2020 degrees F) and 1040 K (1410 degrees F) on the hottest evaluation model rod and hottest best estimate rod, respectively. The high reported values for fuel cladding temperature were a direct consequence of the conservative boundary and initial conditions used for the calculation, primarily the 2% overpower condition, the core decay heat assumption, and the degraded ECCS. The calculation demonstrated successful core reflooding before 1478 K (2200 degrees F) cladding temperature was exceeded on any fuel rod.

NUREG/CR-4966 V01: EVALUATION OF OPERATIONAL SAFETY AT BABCOCK AND WILCOX PLANTS.Volume 1 - Results Overview. HANSON,D.J.; MEYER,O.R.; BLACKMAN,H.S.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). October 1987. 198pp. 8712280204. EGG-2515. 43817:061.

A methodology was developed to assess the operational performance of nuclear power plants through an integration of thermal-hydraulic and human factors analysis techniques together with inputs from information used in the assessment of risk. This methodology was applied to evaluate the extent to which plant systems and/or operator actions are effective in lessening the severity of selected transients for Babcock and Wilcox (B&W) plants. Comparisons were also performed to assess differences in operational performance capabilities and limitations between selected Combustion Engineering, Westinghouse, and B&W plants. Detailed results from the methodology application are presented in two volumes. Volume 1 presents an overview

of the results with emphasis on the systems and operator performance. Volume 2 presents detailed results from thermal-hydraulic calculations.

NUREG/CR-4966 V02: EVALUATION OF OPERATIONAL SAFETY AT BABCOCK AND WILCOX PLANTS.Volume 2 - Thermal-Hydraulic Results. WHEATLEY,P.D.; DAVIS,C.B.; CALLOW,R.A.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). November 1987. 284pp. 8712310121. EGG-2515. 43869:088.

The Nuclear Regulatory Commission has initiated a research program to develop a methodology to assess the operational performance of Babcock and Wilcox plants and to apply this methodology on a trial basis. The methodology developed for analyzing Babcock and Wilcox plants integrated methods used in both thermal-hydraulics and human factors and compared results with information used n the assessment of risk. The integrated methodology involved an evaluation of a selected plant for each pressurized water reactor vendor during a limited number of transients. A plant was selected to represent each vendor, and the three transients were identified for analysis. The plants were Oconee Unit 1 for Babcock and Wilcox, H.B. Robinson Unit 2 for Westinghouse, and Calvert Cliffs Unit 1 for Combustion Engineering. The three transients were a complete loss of all feedwater, a small- break loss-of-coolant accident, and a steam-generator overfill with auxiliary feedwater. Included in the integrated methodology was an assessment of thermal-hydraulic behavior, including event timing, of the plants during the three transients. Thermal-hydraulic results are presented in this volume (Volume 2) of the report. An overview of all results, with emphasis on operator and hardware performance, is presented in Volume 1.

NUREG/CR-4968: EVALUATION OF UNCERTAINTIES IN IRRADIATED HARDWARE CHARACTERIZATION. BEDORE,N.; LEVIN,A.; TUITE,P. Waste Management Group, Inc. October 1987. 102pp. 8711030139. 43242:062.

This report evaluates the techniques used by industry to characterize and classify irradiated hardware components for disposal in accordance with the requirements of 10CFR Part 61 and the uncertainties associated with application of these techniques. It describes current methods and practices, identifies the uncertainties associated with the methods and practices considered, and recommends areas for improvement which could reduce uncertainty. Two different characterization methods are described. The first uses a combination of gamma scanning, direct sampling, underwater radiation profiling and radiochemical analysis to determine radionuclide content, while the second uses a form of activation analysis in conjunction with underwater radiation profiling. Both methods employ two distinct steps: (1) the determination of Cobalt-60 content and (2) the determination of scaling factors for hard-to-detect Part 61 radionuclides. Current uncertainties in Cobalt 60 determination can be reduced by improving underwater radiation profiling equipment and techniques. The calculational techniques used for activation analysis can also be refined to reduce the uncertainties with Cobalt-60 determination. Improved radiochemical analysis techniques, and the development of a statistically significant set of component composition data would eliminate some of the conservatism in current scaling factor determination.

NUREG/CR-4969: AXISYMMETRIC ANALYSIS OF 1:6-SCALE REINFORCED CONCRETE CONTAINMENT BUILDING USING A DISTRIBUTED CRACKING MODEL FOR THE CONCRETE. WEATHERBY,J.R. Sandia National Laboratories. September 1987. 80pp. 8712280239. SAND87-1670. 43820:204.

Results of axisymmetric structural analyses of a 1:6-scale model of a reinforced concrete nuclear containment building are presented. Both a finite element shell analysis and a simplified membrane analysis were made to predict the structural response and ultimate pressure capacity of the model. Analytical results indicate that the model will fail at an internal pressure of 187 psig when the stress level in the hoop reinforcement at the midsection of the cylinder exceeds the ultimate strength of the bar splices.

NUREG/CR-4970: STAINLESS STEEL ROUND ROBIN TEST.Centrifugally Cast Stainless Steel Screening Phase. BATES,D.J.; DOCTOR,S.R.; HEASLER,P.G.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. October 1987. 264pp. 8710290281. PNL-6266. 43216:260.

This report presents the results of the Centrifugally Cast Stainless Steel Round Robin Test (CCSSRRT). The CCSSRRT is the first phase of an effort to investigate and improve the capability and reliability of NDE inspections of CCSS. This phase was a screening test to identify the most promising procedures presently available for CCSS. The next phase will be a more in-depth program. Fifteen centrifugally cast stainless steel pipe sections containing welds and laboratory-grown thermal fatigue cracks in both columnar and equiaxed base material were used. These pipe specimens were inspected by a total of 18 teams from Europe and the United States using a variety of NDE techniques, mostly ultrasonic. It was anticipated that the anisotropic and coarse-grained structure of the specimens would present the inspecting teams with the problem of not only detecting the cracks but also of properly classifying uncracked material. Thus, measurement of team performance included results in both cracked and uncracked material. Some procedures showed promise; however, because of the high number of false crack calls reported by many of the inspecting teams it was difficult to demonstrate that some of the procedures could effectively discriminate between thermal fatigue cracks and uncracked CCSS. The results of the CCSSRRT make it apparent that a more detailed study on the capability and reliability of procedures to inspect stainless steel materials is needed to better understand the specific material and flaw properties and how they affect the outcome of an inspection.

NUREG/CR-4972: TWO-PHASE FLOW REGIME TRANSITION CRITERIA IN POST-DRYOUT REGION BASED ON FLOW VISUALIZATION EXPERIMENTS. OBOT,N.T.; ISHII,M. Argonne National Laboratory. June 1987. 56pp. 8709030565. ANL-87-27. 42470:131.

A visual study of film boiling using photographic and high speed motion-picture methods was carried out to determine the flow regime transition criteria in the post CHF region. An idealized inverted annular flow was obtained by introducing a liquid jet of Freon 113 through a nozzle, precisionly centered with respect to the internal diameter of the test section, with an annular gas flow. The respective ranges for liquid and gas exit velocities were 0.05-0.5 and 0.03-7.9 m/s. Nitrogen and helium were used in the study. For the present configuration, there are four basic flow regimes. Beginning from the nozzle exit, there is smooth, inverted annular flow section with liquid in the core and gas in the annulus; followed by the rough wavy section with an intact liquid core, the agitated and the dispersed flow regimes. For a given liquid jet velocity, the axial extent of each flow regime decreases with increasing gas velocity through the annulus. Generalized transition criteria and simplified correlations for the axial limits of the various flow regimes have been developed, by extending the results of previous studies on adiabatic inverted annular flow.

NUREG/CR-4973: INTRAPLATE SEISMICITY OUTSIDE OF THE UNITED STATES. SCHWEITZER,J.; GLOVER,L. Virginia Polytechnic Institute & State Univ., Blacksburg, VA. July 1987. 88pp. 8708040243. 42036:327.

A survey of intraplate seismicity in eastern Canada, Australia, and western Europe was undertaken. Eastern Canada has been divided into four seismic zones, namely, LaMalbaie (Quebec), western Quebec, New Brunswick, and Baffin Island. Australia, a relatively aseismic continent surrounded by very seismically active areas, has been divided into three seismic zones; western Australia, south Australia, and eastern Australia. Intraplate seismicity in western Europe is concentrated in central and

northern Europe. Based on this survey, several general observations about intraplate seismicity can be made. Much of the seismic activity in these intraplate regions is relatively shallow and is frequently localized in basement rocks. Seismic events are frequently attributed to movement along pre-existing faults, although surface faulting rarely accompanies these earthquakes. Paleo-rift zones seem to be particularly susceptible to reactivation. Plate interactions and asthenosphere drag both play a role in development of a regional horizontal maximum principal compressive stress, but the actual triggering mechanism and localization cause for this fault reactivation in intraplate regions is still only poorly understood.

NUREG/CR-4974: INTRAPLATE SEISMICITY IN THE EASTERN UNITED STATES. BOLLINGER,G.A.; EHLERS,E.G.; MOSES,M.J. Virginia Polytechnic Institute & State Univ., Blacksburg, VA. July 1987. 28pp. 8708040340. 42036:001.

Causes for intraplate earthquakes in the eastern United States are indeed complex. On a regional scale, earthquakes are spatially and temporally stationary, at least on a scale of decades. Spatially, the seismicity occurs in distinct zones superposed on a less active regional background level. Historically, moderate to large earthquakes have occurred within the seismic zones defined by microseismicity. Recent paleoseismicity studies suggest that these larger earthquakes can reasonably be expected to occur into the future. Earthquake focal depths in the eastern U.S. tend to be within the upper crust with little relation to surficial features. Because most eastern U.S. seismic zones are buried in the subsurface, geophysical methods are used to infer the presence of possible causal geologic structures at depth. Gravity data are useful in locating density (lithologic) changes within the crust and seismic activity tends to concentrate within gravity saddles formed at the intersection of northeast and northwest trending gravity anomalies. Seismically important plutons and Triassic basins which are associated with seismic activity have been located by their magnetic signatures. The eastern U.S. is interpreted to be composed of a mosaic of allochthonous suspect terranes. This hypothesis suggests that block boundaries are zones of crustal weakness.

NUREG/CR-4975: A REVIEW OF THE RESOLVING POWER OF REFLECTION SEISMOLOGY METHODS TO DETECT SUBSURFACE FAULTS AND/OR CHANGES IN LAYER THICKNESS. COSTAIN,J.K.; CORUH,C. Virginia Polytechnic Institute & State Univ., Blacksburg, VA. July 1987. 37pp. 8708050045. 42062:110.

Aspects of the resolving power of seismic reflection data in detecting subsurface faults are reviewed. Resolution by reflection seismology of displacements across subsurface faults is coupled with the detection and resolution of thin beds. Limits of detail that can be derived from seismic data are greatly affected by data acquisition and processing parameters. Temporal and spatial bandwidths as keys for seismic resolution are determined by source, recording system, absorption, patterns, temporal frequency range, group interval, temporal and spatial noise. Optimum time wavelets are important for maximum resolution in time which also affects resolution in the spatial domain. Response of vibrators and the use of force control units are reviewed along with the effects of source and receiver arrays used during data acquisition. Phase changes due to decoupling and phase compensation are included in the discussion. The importance of balancing the effects of intrinsic damping before stack is discussed, as are the effects of data windowing in time and space. Parameters affecting resolution along time and spatial axes are given to show that the Fresnel zone introduces an important limitation on the detail that can be derived from seismic data.

NUREG/CR-4976: PLANT RHIZOSPHERE PROCESSES INFLUENCING RADIONUCLIDE MOBILITY IN SOIL. CATALDO,D.A.; COWAN,C.E., MCFADDEN,K.M.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. October 1987. 76pp. 8711160184. PNL-6277. 43371:242.

Native vegetation associated with commercial low-level waste disposal sites has the potential for modifying the soil chemical environment over the long term and, consequently, the mobility of radionuclides. These effects were assessed for coniferous and hardwood tree species by using plants grown in lysimeter systems and examining their influence on soil solution chemistry using advanced analytical and geochemical modeling techniques. The study demonstrated formation of highly mobile anionic radionuclide complexes with amino acids, peptides, and organic acids originating from plant leaf litter and roots. The production of complexing agents was related to season and tree species, suggesting that vegetation management and exclusion may be appropriate after a site is closed. This research provides a basis for focusing on key complexing agents in future studies to measure critical affinity constants and to incorporate this information into mathematical models describing biological effects on radionuclide mobility.

NUREG/CR-4978: THE COOLDOWN ASPECTS OF THE TMI-2 ACCIDENT. THEOFANOUS,T.G. California, Univ. of, Santa Barbara, CA. August 1987. 63pp. 8709150313. 42678:239.

The cooldown of the TMI-2 reactor vessel due to high pressure injection that occurred at 200 minutes into the accident is re- examined. Flow regimes and condensation heat transfer in the cold legs and downcomer are considered. The presence of noncondensibles (hydrogen) and a mechanism leading to its accumulation around the condensation interfaces lead to conclusions that are materially different from those of a previous study that did not consider these effects.

NUREG/CR-4981: A SAFETY ASSESSMENT OF THE USE OF GRAPHITE IN NUCLEAR REACTORS LICENSED BY THE U.S. NRC. SCHWEITZER,D.G.; GURINSKY,D.H.; KAPLAN,E.; et al. Brookhaven National Laboratory. September 1987. 42pp. 8710[illegible]0161. BNL-NUREG-52092. 42921:096.

This report reviews existing literature and knowledge on graphite burning and on stored energy accumulation and releases in order to assess what role, if any, a stored energy release can have in initiating or contributing to hypothetical graphite burning scenarios in research reactors. It also addresses the question of graphite ignition and self-sustained combustion in the event of a loss-of- coolant accident (LOCA). The conditions necessary to initiate and maintain graphite burning are summarized and discussed. From analyses of existing information it is concluded that only stored energy accumulations and releases below the burning temperature (650 degrees C) are pertinent. It is shown that there is no evidence from the Chernobyl event that stored energy releases played a role either initiating or contributing to this accident. The conclusions from these analyses are that the potential to initiate or maintain a graphite burning incident is essentially independent of the stored energy in the graphite, and depends on other factors that are unique for each research reactor and for Fort St. Vrain. There is no new evidence associated with either the Windscale Accident or the Chernobyl Accident that indicates a credible potential for a graphite burning accident in any of the reactors considered in this review.

NUREG/CR-4982: SEVERE ACCIDENTS IN SPENT FUEL POOLS IN SUPPORT OF GENERIC SAFETY ISSUE 82. SAILOR,V.L.; PERKINS,K.R.; WEEKS,J.R.; et al. Brookhaven National Laboratory. July 1987. 156pp. 8708120084. BNL-NUREG-52093. 42128:097.

This investigation provides an assessment of the likelihood and consequences of a severe accident in a spent fuel storage pool - the complete draining of the pool. Potential mechanisms and conditions for failure of the spent fuel, and the subsequent release of the fission products are identified. Two older PWR and BWR spent fuel storage pool designs are considered based on a preliminary screening study which tried to identify vulnerabilities. Internal and external events and accidents are assessed. Conditions which could lead to failure of the spent fuel

Zircaloy cladding as a result of cladding rupture or as a result of a self-sustaining oxidation reaction are presented. Propagation of a cladding fire to older stored fuel assemblies is evaluated. Spent fuel pool fission product inventory is estimated and the releases and consequences for the various cladding failure scenarios are provided. Possible preventive or mitigative measures are qualitatively evaluated. The uncertainties in the risk estimate are large, and areas where additional evaluations are needed to reduce uncertainty are identified.

NUREG/CR-4983: THE SEALING PERFORMANCE OF BENTONITE/CRUSHED BASALT BOREHOLE PLUGS. WILLIAMS,J.R.; DAEMEN,J.J. Arizona, Univ. of, Tucson, AZ. November 1987. 273pp. 8712240101. 43795:321.

Mixtures of crushed rock and bentonite are considered for back- filling and sealing high-level nuclear waste repositories. Many variables affect the hydraulic conductivity of such mixtures, including the size and shape of the rock particles, method of mixing and emplacement, water content and density of the clay, and the weight ratio of rock to clay. Mixtures of crushed basalt and bentonite have been tested in two types of permeameters, 20 cm (8 in) diameter stainless steel permeameters and 10 cm (4 in) diameter PVC permeameters. Plugs were installed as a single lift or in many lifts; the water content of the clay ranged from air-dry (in most cases) to as high as 200%. Preliminary results show that mixture of 75% crushed basalt and 25% bentonite has a hydraulic conductivity between 1 x 10(- 9) cm/s and 2.5 x 10(-8) cm/s. In some cases, preferential flow paths have developed (possibly as a result of the montmorillonite washing out of the crushed rock matrix), giving hydraulic conductivities as high as 1 x 10(-4) cm/s. Other ratios of rock to clay have similar bimodal results. The probability of failure is decreased by including a higher percentage of clay in the plug, crushing the rock finer, and evenly mixing the crushed rock and clay.

NUREG/CR-4985: INDIAN POINT 2 REACTOR COOLANT PUMP SEAL EVALUATIONS. SUBUDHI,M.; TAYLOR,J.H. Brookhaven National Laboratory. August 1987. 96pp. 8712150083. BNL-NUREG-52095. 43729:184.

This report summarizes the findings on Westinghouse Reactor Coolant Pump Seal Performance at the Indian Point 2 Nuclear Power Station. The study was conducted under the joint sponsorship of Consolidated Edison of New York and the U.S. Nuclear Regulatory Commission Office of Research. The conclusions and recommendations herein are based on the review of the plant operational and maintenance data on seals, consultation with Westinghouse and Utilities, review of prior studies, and visual as well as in-depth examination of service exposed seals received from the plant.

NUREG/CR-4986: RADIATION DOSE ESTIMATES AND HAZARD EVALUATIONS FOR INHALED AIRBORNE RADIONUCLIDES.Final Report. MEWHINNEY,J.A. Lovelace Biomed & Environmental Research Institute. September 1987. 105pp. 8710050547. 42905:122.

This is the final report for a project whose objective was to conduct confirmatory research on physical chemical characteristics of aerosols produced during manufacture of mixed plutonium and uranium oxide nuclear fuel, to determine the radiation dose distribution in tissues of animals after inhalation exposure to representative aerosols of these materials, and to provide estimates of the relationship of radiation dose and biological response in animals after such inhalation exposure. This report is divided into three chapters which summarize the results of these investigations. The first report chapter summarizes the physical chemical characterization of samples of aerosols collected from gloveboxes at industrial facilities during normal operations. This chapter provides insights into key aerosol characteristics which are of potential importance in determining the biological fate of specific radionuclides contained in the particulate that would be inhaled by humans following accidental release. The second chapter describes the spatial and temporal distribution of radiation dose in tissues of three species of animals exposed to representative aerosols collected from the industrial facilities. These inhalation studies provide a basis for comparison of the influence of physical chemical form of the inhaled particulates and the variability among species of animal in the radiation dose to tissue. The third chapter details the relationship between radiation dose and biological response in rats exposed to two aerosol forms each at three levels of initial pulmonary burden. This study, conducted over the lifespan of the rats and assuming results to be applicable to humans, indicates that the hazard to health due to inhalation of these industrial aerosols is not different than previously determined for laboratory produced aerosol of PuO(2).

NUREG/CR-4987: SIMULATED SEISMIC TESTS ON 1/42- AND 1/14-SCALE CATEGORY I, AUXILIARY BUILDING. BENNETT,J.G.; DOVE,R.C.; DUNWOODY,W.E.; et al. Los Alamos National Laboratory. October 1987. 42pp. 8712020016. LA-11093-MS. 43521:138.

Two scale-model structures representing an idealized auxiliary building were seismically tested. The scales (1/42, 1/14) were chosen so that both structures were models of the prototype, and 1/42-scale model was a 1/3-scale model of the 1/14-scale structure. Both models were constructed out of microconcrete. The 1/42-scale used wire mesh to simulate reinforcing, and the 1/14 scale used model deformed bars. The general result verified previous test experience in this program: the frequency response of these structures, when subjected to seismic design loads, will be below that predicted from the structural analysis. The implication of this result for equipment & piping is under investigation. The recommendation of this program, based on testing thus far, is to verify the conclusions on larger real concrete structures of a geometry that will be agreed upon by the technical review group for this program.

NUREG/CR-4988: A TRAC-PF1/MOD1 ANALYSIS OF THE GINNA TUBE-RUPTURE EVENT ON JANUARY 25,1982. LIME,J.F.; JENKS,R.P. Los Alamos National Laboratory. October 1987. 61pp. 87120202uf-. LA-11094. 43521:186.

The R.E. Ginna steam-generator tube-rupture (SGTR) event on January 25, 1982, was simulated using the Transient Reactor Analysis Code TRAC-PF1/MOD1, version 12.3. A complete TRAC model of the Ginna plant was developed and the SGTR event was calculated to 5000 s. The overall plant behavior was simulated, with excellent to reasonable agreement obtained between calculated results and measured data. A primary objective of calculating and matching the actual reactor trip time was achieved. The analysis demonstrates that TRAC can accurately simulate full-scale plant SGTR transients.

NUREG/CR-4990: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Five Year Plan.FY 1986-1990. PUGH,C.E. Oak Ridge National Laboratory. September 1987. 189pp. 8711230334. ORNL/TM-10526. 43444:315.

The fourth in an annual series of five-year program-plan documents is presented for the Heavy-Section Steel Technology Program. The program is carried out by the Oak Ridge National Laboratory for the Materials Engineering Branch, Division of Engineering, Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission. The program is aimed at advancing the understanding and validation of materials and structures behavior as they relate to light water reactor pressure-vessel integrity. The program has eleven technical tasks and a management function. A background statement and a plan- of-action is given for each. The eleven technical tasks address fracture methodology and analysis, materials characterization, crack growth, crack arrest, irradiation effects, cladding evaluations, intermediate-vessel testing, thermal-shock testing, pressurized thermal-shock experiments, pressure vessel research users' facility, and shipping-cask material evaluations.

NUREG/CR-4995: EXPERIMENTAL ASSESSMENT OF THE SEALING PERFORMANCE OF BENTONITE BOREHOLE PLUGS. SAWYER,W.D.; DAEMEN,J.J. Arizona, Univ. of, Tucson, AZ. October 1987. 307pp. 8711100270. 43343:010.

Bentonite, which contains the clay mineral montmorillonite, is being considered as part of a multi-component engineered barrier which will seal geological repositories of high-level radioactive wastes. Four phases of laboratory testing were conducted to quantify bentonite strength and sealing properties. Chemical analysis revealed a wide range of compositions for commercial well-sealing grade bentonites such that bentonite quality control or specifications may be necessary. Reference testing of seven bentonite samples revealed serious testing problems that make bentonite characterization difficult using standard tests. Permeability testing compared the water flow through bentonite to water flow through intact rock. Compacted bentonite seals exhibited hydraulic conductivities in the 10(-9) to 10(-11) cm/s range, which is one to two orders of magnitude greater than that of some intact Columbia plateau basalt specimens. Mechanical testing included direct shear strength and swelling pressure tests. Direct shearing of compacted bentonites revealed that for both unsaturated and saturated test conditions, the bentonite/basalt interface will fail in shear before the intact bentonite plug itself fails. Swelling pressures generated by hydrating bentonites depend upon dry density, mixing and saturating fluid, and method of pressure measurement. Swelling pressures in excess of 1.7 MPa (250 psi) were measured. All results, especially numerical ones, depend on numerous installation and test parameters. The sensitivity can be quite high. Variations of several orders of magnitude in hydraulic conductivity can be expected for relatively small variations in some installation or test procedure aspects.

NUREG/CR-5002: METHODS FOR RECURRING LOSS TESTS. JOHNSTON,J.W.; LITTLEFIELD,J.; KINNISON,R.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1987. 58pp. 8710080312. PNL-6249. 42990:261.

Development of a program to use material control and accounting data for detecting recurring losses of nuclear material is described and some potential statistical methods are discussed. A moving sum test, including a moving variation for false alarm control, is described. Three non-parametric tests are discussed because of simplicity of use and because an estimation of the variance of the process difference is not required. Considerations for a computer system to collect and process recurring loss test data are identified and discussed.

NUREG/CR-5003: DESIGN OF A MATERIAL CONTROL AND ACCOUNTING SYSTEM TO PROTECT AGAINST CONCEALMENT OF DIVERSION BY FALSIFICATION AND COLLUSION. SMITH,B.W.; LEWIS,P.M.; BYERS,K.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. October 1987. 69pp. 8710270061. PNL-6308. 43188:309.

This report provides guidance for incorporating checks and balances in a nuclear material control and accounting (MC&A) system to control and detect falsification of data and reports that could conceal diversion of nuclear material. Data falsifications may be difficult to distinguish from mistakes in the data. The elements of control for data falsification are similar to existing security and accounting internal controls. Potential mechanisms for data falsification are identified and internal controls are discussed. Process monitoring systems supplement traditional MC&A by providing more timely detection of diversion and improved localization of losses, thereby providing additional controls against data falsification. However, there is a tradeoff in that personnel who have access to material are generating data for MC&A. Possibilities for collusion involving a person having both physical protection and MC&A responsibilities and another person having MC&A responsibilities are identified. Methods for protecting the system against a collusion threat are discussed.

NUREG/CR-5006: PRA APPLICATION PROGRAM FOR INSPECTION AT OCONEE UNIT 3. GORE,B.F.; VO,T.V.; HARRIS,M.S. Battelle Memorial Institute, Pacific Northwest Laboratories. October 1987. 65pp. 8710290130. PNL-6291. 43220:101.

The extensive Oconee-3 PRA performed by EPRI has been analyzed to identify plant systems and components important to minimizing public risk, and to identify the primary failure modes of these components. This information has been tabulated, and correlated with inspection modules from the NRC Inspection and Enforcement Manual. The report presents a series of tables, organized by system and prioritized by public risk (in person-rem per year), which identify components associated with 98% of the inspectable risk due to plant operation. External events (earthquakes, tornadoes, fires and floods) are not addressed because inspections cannot directly minimize the risks from these events, however, flooding caused by the breach of internal systems is addressed. The systems addressed, in descending order of risk importance, are: Reactor Building Spray, R B Cooling, Condenser Circulating Water, Safety Relief Valves, Low Pressure Injection, Standby Shutdown Facility-High Pressure Injection, Low-Pressure Service Water, and Emergency Feedwater. This ranking is based on the Fussel-Vesely measure of risk importance, i.e., the fraction of the total risk which involves failures of the system of interest.

NUREG/CR-5007: PREDICTION AND MITIGATION OF EROSIVE-CORROSIVE WEAR IN SECONDARY PIPING SYSTEMS OF NUCLEAR POWER PLANTS. KECK,R.G.; GRIFFITH,P. Massachusetts Institute of Technology, Cambridge, MA. September 1987. 46pp. 8710050583. 42911:163.

This report presents equations and parametric curves to predict erosive-corrosive wear rates in low carbon steel secondary piping systems of nuclear power plants. Two mechanisms causing the wear are identified and analytic expressions to predict the wear are presented. One mechanism, wear due to oxide dissolution, is present in single and two-phase flow lines; the other, wear by droplet impact induced fatigue, is only applicable for two-phase flows. Parametric curves are presented to predict the dissolution wear as a function of temperature, pH and characteristic (friction) velocity. An algebraic expression is used for droplet impact wear predictions as a function of superficial gas velocity and thermodynamic quality. Four mitigation strategies incorporated into the models (alloying, pH increase, moisture removal and velocity decrease) are introduced and the effectiveness of each is discussed. The models wear predictions are not expected to be quantitatively exact, but should yield accurate qualitative rankings of systems susceptible to erosive-corrosive damage.

NUREG/CR-5008: DEVELOPMENT OF A TESTING AND ANALYSIS METHODOLOGY TO DETERMINE THE FUNCTIONAL CONDITION OF SOLENOID OPERATED VALVES. MEININGER,R.D.; WEIR,T.J. Pentek, Inc. September 1987. 35pp. 8710060130. 42921:060.

This report describes the Phase I SBIR research conducted to develop and test a technique to determine the functionality of solenoid operated valves (SOVs) in a nuclear plant based on the measurement and analysis of inrush current. The technique developed uses a clip-on current probe, thus enabling all measurements to be made from outside the reactor building without disturbing any electrical connections. The inrush current to the solenoid operated valve is analyzed in real time using a personal computer and Fast Fourier Transform (FFT) techniques.

NUREG/CR-5010: RELAP5/MOD2 ASSESSMENT USING SEMISCALE EXPERIMENTS S-NH-1 AND S-LH-2. YUANN,R.; LIANG,K.; JACOBSON,J.L. EG&G Idaho, Inc. (subs. of EG&G, Inc.). October 1987. 73pp. 8711230401. EGG-2520. 43425:231.

This report presents the results of the RELAP5/MOD2 post-test assessment utilizing two small break loss-of-coolant accident (LOCA) tests (S-NH-1 and S-LH-2) which were performed in the Semiscale Mod-2C facility. Test S-NH-1 was a 0.5%

small break LOCA where the high- pressure injection system (HPIS) was inoperable throughout the transient. Test S-LH-2 was a 5% small break LOCA involving a relatively high upper-head-to-downcomer initial bypass flow and nominal emergency core cooling. Through comparisons between data and best-estimate RELAP5 calculations, the capabilities of RELAP5 to calculate the transient phenomena are assessed. For S-NH-1, emphasis was placed on the capability of the code to calculate various operator actions to initiate core heatup in the absence of HPIS. For S-LH-2, the capability of the code to calculate basic small break system response, such as vessel level during loop seal formation and clearing, break uncovery, and primary pressure response following accumulator injection, was assessed.

NUREG/CR-5011: APPLICATION OF THE ADAPTIVE-PREDICTIVE CONTROLLERS TO PLANT SAFETY SURVEILLANCE UTILIZING ON-LINE PLANT ANALYZER. FABIC,S. Dynatrek, Inc. October 1987. 141pp. 8711250160. 43461:002.

The plant safety surveillance concept described herein aims at continuously providing information which is not accessible through measurements and which, in conjunction with existing plant data, offers promise for a thorough understanding of the plant status and of the causes of upsets when they occur. The concept is based on using the Plant Analyzer Controller, which works in tandem with the Plant Analyzer, to control inputs to the Plant Analyzer so that its outputs continuously track the selected plant measurements. The basic premise is that, if the key plant measurements are being well reproduced (tracked) by the Plant Analyzer, the information which is being continuously computed (most of which cannot be measured), can be used to ascertain the plant status. Research results presented in this report show that the continuous tracking uncovers upset conditions which are, otherwise, either not amendable to measurement or eludes timely detection using conventional methods. The Plant Analyzer (a systems safety code capable of running in real time on a plant computer), and the Plant Analyzer Controller, would be used "on-line," i.e., connected to the plant computer which provides information on plant measurements. Their outputs would be graphically displayed in a comprehensive and easily understood manner. The version of the Plant Analyzer Controller described in this report is based on the Adaptive- Predictive, digital control theory.

NUREG/CR-5014: PARAMETRIC STUDY OF PIPE WHIP ANALYSIS. CHUN,R.C.; CHUANG,T.Y. Lawrence Livermore National Laboratory. October 1987. 50pp. 8711250003. UCRL-21177. 43449:327.

In the Energy Balance Analysis Model (Standard Review Plan (USNRC, 1981), Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping,") time dependence is not considered, and a constant blowdown thrust force is assumed. This force includes an amplification factor of 1.1 to account for potential effects of rebound. Many of the assumptions used in establishing the acceptance criteria, as stated in the Standard Review Plan, were based on engineering judgement and logic intended to assure upper bound design rather than on a mechanistic assessment of actual pipe rupture phenomena and their effects. As a result of the current practice an exceedingly conservative design may be introduced. This report represents a parametric study of the amplification factor to account for rebound effects in the Energy Balance Method. Of the 71 distinct cases we chose for our parametric study, the amplification factor of 1.1 seems sufficient except in five cases where the carbon steel pipes are small or have very small gaps between the pipes and the pipe whip restraints. We conclude that the amplification factor generally decreases as the parameters gap size, hinge-to-break distance and overhang increase.

NUREG/CR-5017: LIMESTONE CONCRETE AEROSOL EXPERIMENTS IN STEAM-AIR ATMOSPHERES.NSPP Tests 521,522,And 531,Data Record Report. TOBIAS,M.L.; ADAMS,R.E. Oak Ridge National Laboratory. October 1987. 64pp. 8712080385. ORNL/TM-10587. 43590:131.

This data record report summarizes the results from two tests involving limestone concrete 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-5025: EXPERIMENTAL STUDIES OF THE EARLY EFFECTS OF INHALED BETA-EMITTING RADIONUCLIDES FOR NUCLEAR ACCIDENT RISK ASSESSMENT.Phase II Report. SCOTT,B.R.; HAHN,F.F.; NEWTON,G.J.; et al. Inhalation Toxicology Research Institute. November 1987. 119pp. 8712170204. 43748:256.

This report summarizes a series of experiments concerning the effect of linear energy transfer and temporal radiation dose pattern to the lung from inhaled beta-emitting radionuclides. The results were used to test the validity of a hazard-function mathematical model for predicting death from radiation pneumonitis. Both morbidity and mortality within 18 months after exposure were examined in rats exposed to beta emitting radionuclides, giving brief or protracted irradiation of the lung or having weak or strong beta emissions. Protraction of the radiation dose to the lung from a lung half time of less than 3 days to a lung half time of 150 days has a sparing effect with a factor of 1.7. Low energy beta emissions have a similar effectiveness in producing lethal injury as high energy beta emissions. The hazard function model for predicting death adequately predicted the median lethal doses for rats.

NUREG/CR-5028: ON THE MELTING AND REFREEZING OF POROUS MEDIA. DOSANJH,S.S. Sandia National Laboratories. October 1987. 80pp. 8712280280. SAND87-1976. 43818:269.

A model of melt formation and relocation in a one-dimensional core rubble bed is developed. The analysis includes: mass conservation equations for the species of interest (UO2 and ZrO2); a momentum equation for which represents a balance among drag, capillary and gravity forces; and an energy equation which incorporates the effects of convection by the melt, radiation and conduction through the bed, and the energy release associated with fission product decay. An equilibrium UO2-ZrO2 phase diagram is incorporated utilizing a temperature-dependent thermal conductivity. A parametric study is conducted in which the height of the bed, the average particle diameter, the initial composition, the initial temperature profile, and aspects of the phase diagram are varied. Results from this study are then used to evaluate the DEBRIS module in the MELPROG computer code. Particular attention is given to the semi-empirical melt relocation models in DEBRIS. Experiments needed both to validate the current rubble bed model and to help guide future theoretical work are discussed.

NUREG/CR-5034: AN UPDATE OF MIDCONTINENT TECTONISM. HINZE,W.J.; BRAILE,L.W.; KELLER,G.R.; et al. Purdue Univ., West Lafayette, IN. October 1987. 62pp. 8711180060. 43394:001.

The nature of the tectonism and geodynamical processes of the midcontinent region of the United States is poorly understood because of the masking effect of relatively undisturbed Phanerozoic sediments, the short historical record of seismicity and the long earthquake recurrence interval. However, during the past decade, progress has been made in attacking these problems as a result of an improved data base which permits integration of seismicity data with ancient crustal structures and current stress orientations. Surface geophysical data together with seismicity, stress determinations and data from a few critically located deep drill holes have led to a much improved knowledge of the geological history of the crust and its physical attributes. Combining our knowledge of crustal structures, stresses, and seismicity, it is possible to relate current seismicity and tectonism to one of two models. Increasing evidence from across the midcontinent supports reactivation of pre-existing zones of crustal weakness that are appropriately oriented with respect to the stress field - the 'zone of weakness model' - as the model for the dominant contemporary tectonism and the 'local basement inhomogeneity model' as the mechanism for minor, low energy release earthquake activity.

NUREG/CR-5035: DATA BASE OF SYSTEM-AVERAGE DOSE RATES AT NUCLEAR POWER PLANTS.Final Report. BEAL,S.K.; BRITZ,W.L.; COHEN,S.C.; et al. Science & Engineering Associates, Inc. October 1987. 79pp. 8711060148. 43309:011.

Radiation exposure to workers is one of the indirect costs that must be considered in a regulatory analysis. Radiation exposure may be estimated from the product of the estimated man-hours to perform a job and the dose rate in the area in which the job is performed. In previous work, a methodology has been developed for estimating the man-hours required to perform jobs in operating nuclear power plants. The methodology uses the Energy Economic Data Base (EEDB) as a starting point. In this work a data base is derived of area dose rates for systems and components listed in the EEDB. The data base is derived from area surveys obtained during outages at four boiling water reactors (BWRs) at three stations and eight pressurized water reactors (PWRs) at four stations. Separate tables are given for BWRs and PWRs. These tables may be combined with estimates of labor hours to provide order-of-magnitude estimates of exposure for purposes of regulatory analysis. Caution should be exercised in the wanton use of these tables, however. They are only valid for work involving entire systems or components. The estimates of labor hours used in conjunction with the dose rates to estimate exposure must be adjusted to account for in-field time. Finally, the dose rates given in the data base do not reflect ALARA considerations.

NUREG/CR-5036: COST ANALYSIS FOR IMPLEMENTATION OF PROPOSED CHANGES IN VITAL AREA REQUIREMENTS. CLAIBORNE,E.; WATLINGTON,B.; SIMION,G.; et al. Science & Engineering Associates, Inc. October 1987. 122pp. 8711100260. 43342:060.

The Vital Area Committee (VAC) recently recommended changes to vital area requirements of nuclear power plants. Vital areas are those areas which contain vital equipment and therefore must be protected against radiological sabotage. In general, these recommendations are more relaxed than the current criteria contained in Review Guide 17 (RG-17). Before the VAC recommendations can be implemented, the NRC must perform a regulatory analysis. This study is a cost analysis of the RG-17 criteria and VAC recommendations to serve as one input to the regulatory analysis. The approach was to apportion the nuclear power industry into groups of similar plants, perform the cost analysis on selected plants from the groups, and extrapolate the results to the entire industry.

NUREG/CR-5037: FIRE ENVIRONMENT DETERMINATION IN THE LASALLE NUCLEAR POWER PLANT CONTROL ROOM. USHER,J.L.; BOCCIO,J.L. Brookhaven National Laboratory. October 1987. 31pp. 8712090097. BNL-NUREG-52106. 43631:272.

One of the objectives of the Fire Protection Research Program (FPRP) of the U.S. NRC is to improve the modeling of environments caused by fires in typical nuclear power plant enclosures. A three- dimensional fluid dynamics computer code (PHOENICS) has been adapted as a field-model fire code (SAFFIRE) for this purpose. The model has been applied to simulate two distinct fires in the control room of the LaSalle County power plant. The environments determined illustrate hazardous potential for both personnel and equipment.

NUREG/CR-5040: RADIONUCLIDE MIGRATION AROUND URANIUM ORE BODIES - ANALOGUE OF RADIOACTIVE WASTE REPOSITORIES.Annual Report For 1984-1985. AIREY,P.L.; DUERDEN,P.; ROMAN,D.; et al. Australian Atomic Energy Commission. November 1987. 217pp. 8711180064. AAEC/C55. 43394:065.

Evaluation of the uranium ore bodies in the Northern Territory of Australia as natural analogues of high-level waste repositories continues to contribute to the scientific basis for the long-term prediction of radionuclide migration, to assess the assumption underlying geochemical transport codes and to identify those processes which are significant for long-term prediction, but are not observed over laboratory timescales. An open system model has been used to calculate the rate of redistribution of uranium within the secondary mineralization, and radionuclide transport equations have been used to calculate retarded uranium travel times in the Koongarra ore body. A wide range of techniques, such as high gradient magnetic separation and sequential extraction procedures were used to study drill core and soil samples; other techniques such as fission and alpha recoil track analysis, proton induced X-ray and gamma-ray emmission and electron microprobe analysis have been used to examine thin sections of the drill core. Considerable effort has been made in the last year to extend the analogue to the study of transuranics and fission products. Supplies of drill core and groundwater colloids were collected in the field for I-129 and Pu-239 analyses and a comprehensive model for the production of I-129 in the uranium ore body was developed.

NUREG/CR-5041 V01: RECOMMENDATIONS TO THE NRC FOR REVIEW CRITERIA FOR ALTERNATIVE METHODS OF LOW-LEVEL RADIOACTIVE WASTE DISPOSAL. Task 2a:Below-Ground Vaults. DENSON,R.H.; BENNETT,R.D.; WAMSLEY,R.M.; et al. Army, Dept. of, Army Engineer Waterways Experiment Station. December 1987. 162pp. 8801070150. 43944:190.

Recommendations are provided for general design criteria and specific design review criteria covering the design, construction and operation of the below-ground vault (BGV) alternative method of low- level radioactive waste (LLW) disposal. A BGV is a reinforced concrete vault for LLW disposal that would be placed underground, below the frost line and above the water table, surrounded by filter and drainage systems and covered over with a low permeability earth layer. Eight major review criteria categories are identified in the report. The categories include (1) the loads and load combinations to be used in design, (2) structural design and analytical methods, (3) construction material quality and durability, (4) construction and operations, (5) quality assurance, (6) structural performance monitoring, (7) filter and drainage systems, and (8) waste cover system.

NUREG/CR-5042: EVALUATION OF EXTERNAL HAZARDS TO NUCLEAR POWER PLANTS IN THE UNITED STATES. KIMURA,C.Y. Lawrence Livermore National Laboratory. BUDNITZ,R.J. Future Resources Associates, Inc. December 1987. 231pp. 8712240077. UCID-21223. 43795:091.

As part of the research program supporting the implementation of the NRC Policy Statement on Severe Accidents, the Lawrence Livermore National Laboratory (LLNL) has performed a study of the risk of core damage to nuclear power plants in the United States due to externally initiated events. The broad objective has been to gain an understanding of whether or not each external initiator is among the major potential accident initiators that may pose a threat of severe reactor core damage or of large radioactive release to the environment from the reactor. Four external hazards were investigated in this report. These external hazards are internal fires, high winds/tornadoes, external floods, and transportation accidents. Analysis was based on two figures-of-merit, one based on core damage frequency and the other based on the frequency of large radioactive releases. Using these two figures-of-merit as evaluation criteria, it has been feasible to ascertain whether the risk from externally initiated accidents is, or is not, an important contributor to overall risk for the U.S. nuclear power plants studied. This has been accomplished for each initiator separately.

NUREG/CR-5059: VOID FRACTION MEASUREMENT LIQUID LEVEL DETECTION CONCEPT ASSESSMENT AND DEVELOPMENT. MOHR,C.L.; REICH,F.R.; BERRETT,M.K.; et al. Mohr & Associates. November 1987. 86pp. 8712240093. 43809:242.

The Coolant Inventory Monitor System for measuring the amount and the distribution of coolant in the primary system under accident conditions of a nuclear power plant is described in this report. This system is based on measuring the local void fraction of the coolant at selected points in the vessel and piping system and then estimating the coolant inventory. This report summarizes the development of the measurement instrument, the supporting tests and the development of the supporting software. Accuracies of plus minus 2% as compared with delta pressure cells can be obtained over void fractions ranging from 0 to 90%. The system can provide local measurements of void fraction and uses probes as small as 0.015 X 1.0 X 0.3 in (0.038 X 2.54 X 0.76 cm). Experimental test data are shown for temperatures up to 570 degrees F (572K). The system can be used up to the critical point. The graphics display shows a real time display of void fraction for up to fifteen channels and gives the instantaneous as well as a running average for each channel. This instrument is believed to meet the intent of the NRC be being able to measure the amount of water above the reactor core under all types of accidents.

NUREG/CR-5060: AN APPROACH TO THE MATHEMATICAL MODELLING OF THE URANIUM SERIES REDISTRIBUTION WITHIN ORE BODIES. AIREY,P.L.; GOLIAN,C.; LEVER,D.A. Australian Atomic Energy Commission. December 1987. 116pp. 8801110206. AAEC/C49. 43975:041.

The distributions of the uranium series isotopes in the Koongarra ore body were "predicted" by a simple one-dimensional model of isotopic migration parameters. Controlling the "predictions" were the retardation factors and the groundwater flow rates. The conditions modelled were a groundwater flow rate of 1 meter per year, retardation factors of 1 x 10(4) for 238U and of 1.2 x 10(4) for 234U, and 230Th is immobile. For this set of conditions, an exceptionally good agreement was observed between field-measured isotope ratios and "predicted" isotope ratios. A multiphase model adequately describes the redistribution of uranium and daughter radionuclides in the weathered zone of the ore bodies. It redistributes U, Th and Ra isotopes among an aqueous phase, an amorphous iron phase, a crystalline phase and a "phase" of clay and quartz.

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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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.

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

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- NUREG/CR-2000 V06 N5: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of May 1987.
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This index lists, in alphabetical order, the countries and performing organizations that prepared the NUREG/IA reports listed in this compilation. Listed below each country and performing organization are the NUREG/IA numbers and titles of their reports. If further information is needed, refer to the main citation by the NUREG/IA number.

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

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